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Vol. 1

Final Programmatic **Environmental Impact Statement** related to decontamination and disposal of radioactive wastes resulting from March 28, 1979, accident Three Mile Island Nuclear Station, Unit 2

Docket No. 50-320

Metropolitan Edison Company
Jersey Central Power and Light Company
Pennsylvania Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1981

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A Final Programmatic Environmental Impact Statement (PEIS) related to the decontamination and disposal of radioactive wastes resulting from the March 28, 1979, accident at Three Mile Island Nuclear Station, Unit 2 (Docket No. 50-320) has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission in response to a directive issued by the Commission on November 21, 1979. This statement is an overall study of the activities necessary for decontamination of the facility, defueling, and disposition of the radioactive wastes. The available alternatives considered ranged from implementation of full cleanup to no action other than continuing to maintain the reactor in a safe shutdown condition. Also included are comments of governmental agencies, other organizations, and the general public on the Draft PEIS on this project, and staff responses to these comments.

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March 1981

COVER SHEET AND ABSTRACT

1. Proposed Action and Location:

   DECONTAMINATION AND DISPOSAL OF RADIOACTIVE WASTES RESULTING FROM THE MARCH 28, 1979, ACCIDENT AT THREE MILE ISLAND NUCLEAR STATION, UNIT 2, LOCATED IN LONDONDERRY TOWNSHIP, DAUPHIN COUNTY, PENNSYLVANIA.

2. Messrs. Oliver Lynch and Paul Leech are the Project Managers for this statement. They may be contacted at the Three Mile Island Program Office, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 or at 301-492-7258.

3. A Final Programmatic Environmental Impact Statement (PEIS) related to the decontamination and disposal of radioactive wastes resulting from the March 28, 1979, accident at Three Mile Island Nuclear Station, Unit 2 (Docket No. 50-320) has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission in response to a directive issued by the Commission on November 21, 1979. This statement is an overall study of the activities necessary for decontamination of the facility, defueling, and disposition of the radioactive wastes. The available alternatives considered ranged from implementation of full cleanup to no action other than continuing to maintain the reactor in a safe shutdown condition. Also included are comments of governmental agencies, other organizations, and the general public on the Draft PEIS on this project, and staff responses to these comments.

SUMMARY

This programmatic environmental impact statement (PEIS) by the staff of the U.S. Nuclear Regulatory Commission (NRC) is an overall study of the activities necessary for decontamination of the facility, defueling, and disposition of the radioactive wastes which resulted from the accident on March 28, 1979, at Unit 2 of the Three Mile Island Nuclear Station (TMI-2). The following summary has been prepared by the staff for those who prefer to follow the main themes of the statement without referring to the technical descriptions, calculations, data, and other details that provide a basis for assessing the cleanup alternatives and their impacts.

In response to a directive issued by the Commission on November 21, 1979, to prepare this PEIS, the staff has reviewed the status of the contaminated facilities and their surroundings, surveyed the methods available to carry out the cleanup operations, and analyzed the impacts of the cleanup activities on the environment, members of the public, and plant workers. In summary, the staff has reached the following major conclusions and findings (see Section 12 for a complete listing):

- Full cleanup of the TMI-2 facilities should proceed as expeditiously as reasonably possible to reduce the potential for uncontrolled releases of radioactive materials to the environment.

- Existing methods are adequate, or can be suitably modified, to perform virtually all of the necessary operations without incurring environmental impacts that exceed acceptable limits; where special tools or methods are found necessary for operations such as defueling, engineering expertise is available to cope with such requirements.

- An early decision to decommission TMI-2 will have very little effect on the choice of alternatives for the cleanup tasks because most of the same tasks must be performed in order to remove and dispose of the damaged fuel.

- The time needed to complete the cleanup will be 5 to 9 years from the time of the accident.

- The most significant environmental impact associated with the cleanup will result from the radiation doses received by the entire work force from cleanup activities. These doses are estimated to be in the range from 2000 to 8000 person-rem.

- It is predicted that less than one additional cancer death attributable to exposure to radiation will occur among the entire work force engaged in cleaning up TMI-2. (The death rate from cancer among the U.S. population averages approximately 200 deaths per 1000 people.) Not more than two additional genetic defects are expected in descendents of exposed workers. (Among the U.S. population, approximately 60 genetic defects can be expected per 1000 people.)

- Throughout the cleanup, any anticipated releases to the environment must be controlled by the licensee in accordance with the staff's proposed effluent criteria to conform to the individual dose design objectives listed in 10 CFR Part 50, Appendix I, as mandatory limits. The total-body dose design objectives are 15 mrem/year from airborne particulate releases and 3 mrem/year from liquid releases. Implementation of the criteria in this manner is more stringent than for normally operating plants in recognition of the condition of TMI-2.

- Assuming the cleanup is conducted in accordance with the staff's proposed effluent criteria, the staff estimates that, for the entire cleanup, the cumulative total body dose to the maximum exposed individual offsite will range from 0.8 to 2.3 mrem for gaseous effluents.

- An individual offsite receiving the maximum estimated dose resulting from atmospheric releases during the entire cleanup (0.8 to 2.3 mrem) would incur an estimated increased risk of dying from cancer of between 1 in 2 million and 1 in 600,000, and an increased risk of a genetic effect to offspring over the next five generations of between 1 in 300,000 and 1 in 100,000.

- Assuming the cleanup is conducted in accordance with the staff's proposed effluent criteria, the total cumulative dose received by the entire population within a 50-mile radius of TMI-2 due to both gaseous and liquid releases would range from 10 to 30 person-rem for the entire cleanup. This is a small fraction (about .01%) of the background radiation dose received annually by the population from causes other than releases from TMI (annual population background radiation dose = 116 mrem/yr × 2.2 × 10^6 people = 255,000 person-rems).

- The psychological distress caused by the accident and operations necessary to proceed with the cleanup has declined, but there is a potential for temporary increases in distress as various cleanup activities are undertaken.

- The contaminated accident-generated water in the reactor building basement (sump) and in the reactor primary system cannot be left in its present condition and location if the cleanup effort is to proceed. Removal of this contaminated accident water will reduce the airborne and direct radiation levels in the building sufficiently to permit other cleanup operations to be accomplished with greater safety.

- Treatment of the contaminated accident water will transform the entrained radioactivity from its current mobile state to a more manageable form by concentrating and immobilizing the activity by an appropriate process. The cleanup activity will eliminate the risks associated with leaving the contaminated accident water radionuclide inventory in the mobile unprocessed state.

- A decision on the ultimate disposal of the processed water can be deferred until after the water has been processed. Then, the concentration of radionuclides remaining in the water will be low enough for the water to be stored safely onsite until the disposal decision is made. Processing the water to immobilize most of the radionuclides and storage of the processed water will not foreclose any reasonable options for disposition of the water or concentrated wastes.

- The staff regards the transfer of high-specific-activity waste to facilities operated by the Department of Energy to be the most appropriate course of action for processing and final disposal of this material. In the interim, radioactive fuel and high-specific-activity wastes from TMI-2 must be packaged and will have to be stored at the site temporarily until a suitable disposal site is established elsewhere. No significant environmental effects are expected from these activities.

- The staff has concluded that TMI should not become a permanent radioactive waste disposal site. If the damaged fuel and radioactive wastes are not removed, the Island would, in effect, become a permanent waste disposal site. The location, geology, and hydrology of Three Mile Island are among the factors that do not meet current criteria for a safe long-term waste disposal facility. Removing the damaged fuel and radioactive waste to suitable storage sites is the only reliable means for eliminating the risk of widespread uncontrolled contamination of the environment by the accident wastes.

The staff has based its analysis on the licensee's plans,* where they are available, as well as on alternatives the staff has independently developed and assessed. The alternatives considered are, in general, dependent upon radiological and technological conditions encountered. Because the precise conditions of the reactor core and other parts of the system are not known, the staff has described and assessed probable or bounding situations. When more information becomes available, appropriate supplements to the PEIS will be issued if the affected operations are found to be significantly beyond the scope of these assessments.

The ultimate disposition of TMI-2 is of interest to the Federal, State, and local governments, as well as to the licensee and the public. However, the disposition of the facility--whether to decommission or restore it to a condition acceptable for licensed operation--is not within the scope of this PEIS. The March 28, 1979, accident and its associated environmental impacts also are not within the scope of this PEIS.

*The term "licensee" or "Met-Ed" in this document refers to Metropolitan Edison Company, the principal owner (50 percent) and operator of the plant, Jersey Central Power and Light Company, Pennsylvania Electric Company, each of which owns 25 percent.

S.1 The Situation

During the accident at TMI-2, the reactor coolant water level dropped, uncovering the upper portion of the reactor core. This produced temperatures in the core in excess of 2500°F, which may have had the following consequences:

- Reaction of possibly 50 percent of the Zircaloy fuel cladding tubes (in the uncooled upper core region) with the water vapor and steam, thereby causing the tubes to fail and exposing uranium oxide fuel pellets containing fission products.
- Possible melting and fusing together of various stainless steel parts on adjacent fuel assemblies, such as the top end fittings and spacer grids that are located along the fuel assembly.
- Cracking and crumbling and possibly melting of uranium oxide fuel pellets in the overheated section of the core.
- Possible damage, caused by overheating, of other reactor parts. It is possible that the overheating produced local distortions and warping of some of these components.

Small pieces of fuel and other radioactive material may have been carried from the core by the flow of coolant. Larger fragments may have settled out in parts of the primary coolant system, smaller particles may be in suspension, and some will be dissolved in the cooling water. Radioactive material also plated out, forming a thin layer on the inside surfaces of the coolant system components. Although the total quantity of radioactivity in these various forms is not known with any precision, the upper limit on total radioactivity currently in the reactor coolant system (exclusive of fuel) is estimated at about 140,000 Ci.

Some of the radioactive gases leaked out of the reactor coolant system along with a large amount of water. Some of these gases escaped to the environment, but substantial amounts of radioactive gases remained in the reactor building. Shortly after the accident, xenon and iodine gases accounted for most of the radioactivity in the reactor building atmosphere, but these decayed rapidly to nonradioactive forms. The radioactivity remaining in the reactor building atmosphere up to June 27, 1980, consisted almost entirely of an estimated 57,000 Ci of krypton (Kr-85) gas.** Following authorization by the Commission, the gas was purged to the outside atmosphere during the period June 28 to July 11, 1980. Subsequently, the building has been purged several times. The release of Kr-85 has not exceeded 100 Ci for any purge; as of December 1980, the amount purged was less than 15 Ci per month and is decreasing. Some of the purges were made in conjunction with entries into the reactor building.

Several hundred thousand gallons of highly contaminated water were released from the primary system when the reactor pressurizer relief valve stuck open early in the accident and the coolant overflow tank ruptured. Additionally, primary system coolant leaked from the letdown and makeup system into the auxiliary and fuel handling building (AFHB) contaminating the floors, walls, and storage tanks. About 700,000 gallons of contaminated water (termed sump water) are standing about 8 feet deep in the reactor building basement. This sump water contains about 500,000 Ci of radioactivity. There also are about 100,000 gallons of water containing an estimated 20,000 Ci still circulating in the reactor coolant system. At the present time, heat from the reactor is lost to the building and ultimately to the environment. Backup cooling systems are available if needed. Tritiated water and dissolved radionuclides of cesium and strontium are the dominant radioactive materials of concern in the accident water. The other radionuclides are in low concentrations.

Because the tanks then available for storing contaminated water were rapidly being filled, the necessity for decontaminating the radioactive water in the AFHB tanks and sumps was recognized soon after the March 28 accident. The use of a demineralizer system, designated as EPICOR II, was authorized for this purpose by the NRC on October 16, 1979, and cleanup of the water in the AFHB has been completed. This processed water is being stored on the site in accordance with direction of the Commission and an agreement among the NRC, the licensee, and the City of Lancaster. The processed water still contains tritium, which is not removed by the EPICOR II system.

**Analysis after completion of the purging showed that, if instrument errors and uncertainties in the building free volume are considered, the actual amount of Kr-85 purged was approximately 44,000 Ci.

A demineralizer system designed for decontaminating water containing higher levels of radioactivity than EPICOR II is under construction by the licensee for processing the water in the reactor building. This system, known as the submerged demineralizer system (SDS), and alternatives to SDS are evaluated in Section 7.1 of the PEIS. Approval by the NRC would be required before any of these systems could be placed in operation.

Exposed interior surfaces and equipment in the AFHB and the reactor building were contaminated during the accident. The AFHB was contaminated by primary coolant leakage from the makeup and letdown system, and the reactor building was contaminated by hot water and steam carrying radionuclides released to the building under pressure. The interior exposed surfaces of both buildings were coated with thin deposits (known as plateout) of radioactive material. Removal of the plateout in the AFHB began in April 1979 and about two-thirds of the interior surfaces had been decontaminated by September 1980. Very little has been accomplished since then because of the licensee's limited funds. The largest portion of the radioactive contamination in the AFHB was deposited in the sludge in the sump and several tanks in the auxiliary building. These radiation sources have not yet been removed.

Five entries into the reactor building for radiation mapping and damage assessment had been made by January 1981, but work on decontaminating the reactor building had not yet started.

No significant impacts have been identified as a result of the low-activity solid waste handling and shipment operations to date. Wastes shipped by truck to the commercial low-level disposal facility near Richland, Washington, have consisted of immobilized decontamination solutions, compacted trash and noncompactible solid materials. As of February 5, 1981, 2013 drums and 273 LSA boxes of low-level waste had been transferred off the island in 36 truck shipments.

An interim radwaste storage facility has been constructed onsite to store temporarily some of the higher activity wastes, such as the spent demineralizer beds from the water treatment systems. This storage facility will be used until the evaluation of alternatives for offsite disposal of these wastes has been completed and an appropriate one is selected.

S.2 Reasons for Cleanup

The cleanup operations will remove sources of potential radiation exposure that currently pose risks to the health and safety of station workers and the public. Radiation sources are present in the form of airborne contamination, wastewater contaminated by radioactive materials during the accident, plateout of radioactive material on building and equipment surfaces, contaminated sludge, contaminated filter cartridges and demineralizer resins, and damaged fuel. As long as water with radioactive substances in it is allowed to occupy sumps and tanks, there exists a small probability of leakage into the groundwater and subsequently into the Susquehanna River. The contaminated water is also a source of direct radiation to workers requiring access to the building to perform critical maintenance (e.g., repair of nuclear instrumentation) or other repair to maintain the reactor in safe shutdown condition.

The reactor has been in a safe shutdown state since April 1979. The primary system temperature is about 120°F, and the small and decreasing amount of decay heat still being generated is being lost to the building. A new forced circulation system for the primary coolant, the mini-decay heat-removal-system (MDHRS), has been installed but is not in use because loss of heat to the building has been shown to be adequate. As long as the damaged fuel in the reactor core is cooled and remains relatively undisturbed and surrounded by boron-rich coolant, there is essentially no chance that the nuclear chain reaction, which was abruptly stopped at the time of the accident, could start again. But, the staff believes that as time passes, there will be an increasing potential for failure of essential equipment. Even though improbable, if the core were accidentally to begin a chain reaction once more, radioactivity could be released to the reactor building. The amount of radioactivity released during an accidental recriticality would be much less than that released in the initial accident. Even so, timely removal of the damaged fuel to safe storage is a paramount objective of the cleanup of TMI-2.

The feasibility of partial cleanup alternatives in which the reactor building would be sealed with some or all of the radioactive sources left in place was examined. It was found that all such alternatives, including taking no action other than maintaining the reactor in safe shutdown condition, either would not eliminate the potential risks or would convert part of the TMI-2 site into a long-term or permanent waste repository.

The staff concluded that all of the cleanup operations must be performed whether TMI-2 is decommissioned or refurbished to generate electricity in the future. The environmental impacts would be essentially the same regardless of whether the cleanup alternatives were chosen on the assumption that the plant would be decommissioned or on the assumption that the plant would be refurbished and restarted. The differences are less than the uncertainties in the best estimates that can be made for these impacts on the basis of the information presently available.

Cleanup of the facility should proceed in a timely manner, not only to mitigate any risks to the physical health of workers and nearby residents, but also to complete those activities which can cause psychological distress for residents in the area. The sooner the cleanup process is completed, the sooner the sources of concern will cease to exist.

S.3 The Cleanup Operations

The cleanup comprises four fundamental activities: building and equipment decontamination; fuel removal and decontamination of the primary coolant system; treatment of radioactive liquids; and packaging, handling, storage and transportation of radioactive wastes. The current schedule for conducting these activities is indicated in Figure 1. Figure 2 illustrates how the wastes resulting from the cleanup activities, and from the accident itself, would be separated for eventual packaging and disposal.

The removal of unwanted radioactive contamination from materials and equipment is a familiar and routine operation for reducing radiation levels. Decontaminations of various types have been conducted since the 1940s and a considerable amount of experience and technology is available. These experiences illustrate that available techniques can be modified to suit the conditions at TMI-2. Applicable experience in removing damaged fuel and core components is limited, hence development of specific techniques will be required.

Building and Equipment Decontamination

Cleanup of the AFHB started with the general areas where contamination was relatively light and is proceeding to rooms (cubicles) containing tanks and other equipment which are more heavily contaminated. The methods in use are not essentially different from those used in the process of removing dirt from any surface except that care must be taken to protect workers from radioactive contamination, and more stringent methods must be used to remove most of the contamination. The methods used for decontaminating building and fixed equipment surfaces include washing with a high-pressure water jet, wet and dry vacuuming, and manual wiping. Clean surfaces are often protected by applying strippable coatings which are easily removed if the surface becomes recontaminated. Small demountable equipment items can be cleaned by electrochemical or ultrasonic techniques. The methods considered to be most practicable for removing the sludge that accumulated in the sump, pipes, tanks and other vessels containing water involve resuspension of the sludge in water by agitation to form a slurry which is pumped out and filtered.

As of September 1, 1980, the amount of labor by workers directly involved in the AFHB decontamination effort was about 500,000 person-hours. The average exposure rate for these workers generally allowed normal shift operation. Shielding is used to protect workers from the ambient radiation fields, and only a fraction of a worker's time on the job is actually spent in the radiation field. The staff estimates that a total of about 750,000 person-hours of work effort will be needed to decontaminate the AFHB.

The methods used for decontaminating the reactor building will be similar to those used in the AFHB, although the strategies will be different because the reactor building consists primarily of large open spaces, while the AFHB is divided into many small cubicles. Surface decontamination may be easier in the reactor building than in the AFHB because most of the reactor building surfaces are painted, whereas most of the surfaces in the AFHB are untreated concrete. The major tasks that must be coordinated and carried out are removal of the contaminated water from the basement, removal of the sludge and debris, and removal of the plateout from the building and equipment surfaces.

The staff estimates that decontamination of the reactor building (excluding any additional decontamination that might be required in connection with decommissioning or refurbishing

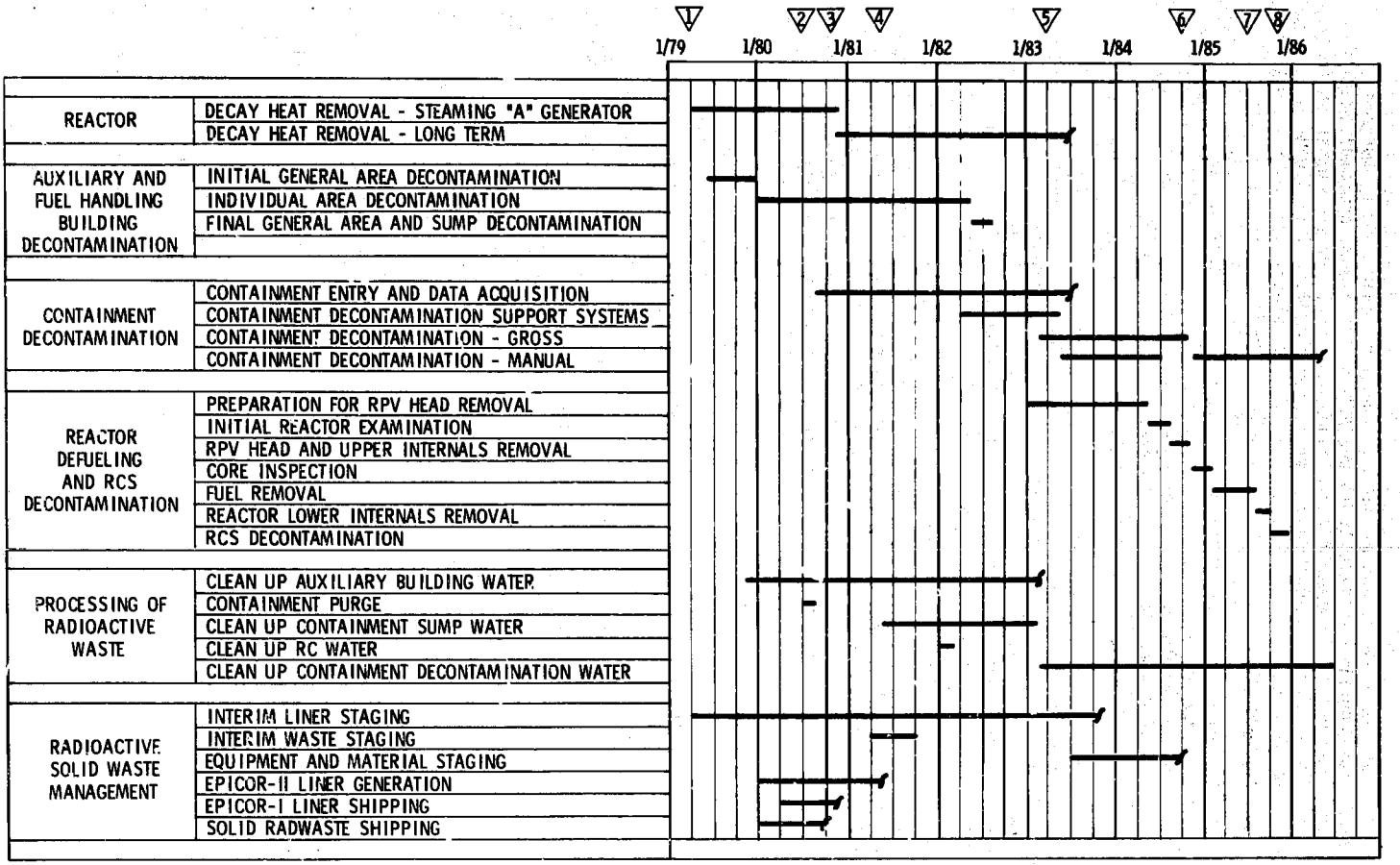

Figure 1. Licensee's Planning Schedule for TMI-2 Cleanup, Phase I and II. (Construction of support facilities not included.)

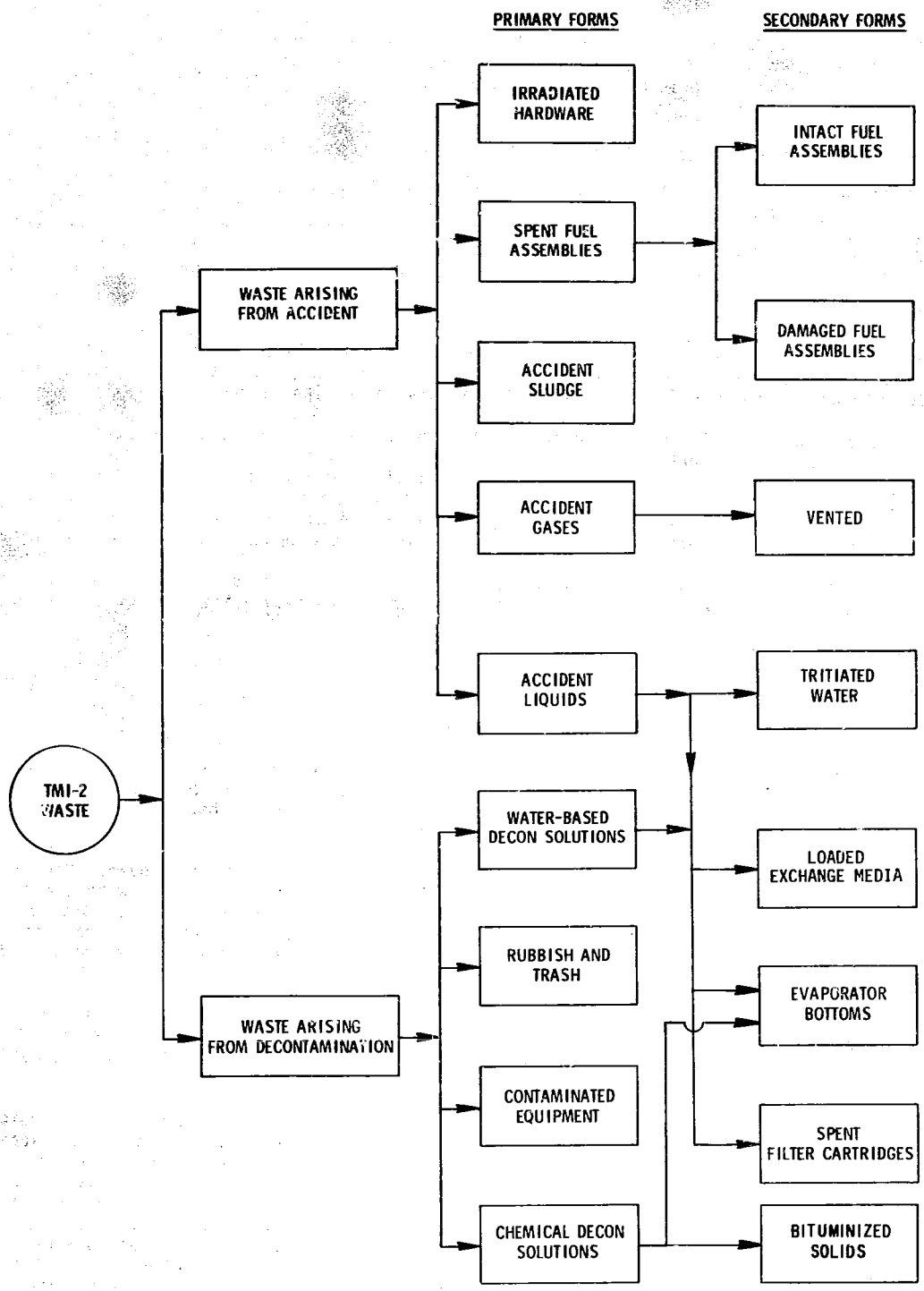

Figure 2. Characterization of TMI-2 Radioactive Waste.

operations) would require a work effort in the range of 300,000 to 900,000 person-hours. This includes both in-building and out-of-building time for workers with assignments that require entry into the building; it does not include support work by others with assignments that do not require entry.

Fuel Removal and Primary System Decontamination

The ultimate objective of the reactor defueling and primary system decontamination is to remove all fuel, damaged reactor parts, and radioactive plateout in the coolant system. Because the exact condition of the reactor core and some of the other reactor parts will not be known until thorough inspections have been performed, it is impossible to predict or plan the defueling and cleanup operations in complete detail at this time. However, the major steps and the order in which they will be conducted are reasonably certain.

The coolant system must first be connected to a cleanup system so that the coolant can be decontaminated. It also will be necessary to remove any additional radioactive materials released to the water during fuel removal operations. This could be done by continuously running the water through a cleanup system consisting of filters and demineralizers.

Obtaining access to the fuel requires removal of the reactor vessel head and components above the fuel. These tasks can be performed in much the same manner as during normal reactor refueling operations, namely by direct contact methods. If minimal warpage or mechanical damage occurred during the accident, these operations should proceed with relative ease. However, if warpage or mechanical damage is extensive, considerable difficulty could be encountered in fuel removal operations, and underwater cutting and machining operations might be needed.

The fuel will be moved under water to the spent fuel pool for interim storage. Some of the fuel assemblies may not be all in one piece. Operations necessary to remove the fuel assemblies are:

- Detailed inspection of the core.
- Removal of loose debris.
- Removal of fuel assemblies using special equipment.

During defueling, it is very important to maintain the boron concentration in the circulating water at the proper level in order to prevent reactor recriticality. Since some of the water treatment processes remove boron along with the radionuclides, boron may have to be added to the water during the defueling.

After the fuel has been removed, the support structure for the fuel must be removed. Normally, removal of the support structure is a straightforward procedure requiring hook up with the crane, lifting it out of the pressure vessel, and moving it to the fuel transfer canal. Because of the possibility that overheating has caused distortion and warping of the support structure, removing it may not be easy. Accordingly, planning allows for the contingency of having to remove this structure by cutting it into smaller pieces while still under water.

After removal of all the fuel from the reactor pressure vessel (hence removing of any further source of radioactivity which could recontaminate the system), the final step will be to clean out the residual radioactivity from the system. This would be accomplished by a method quite analogous to flushing the cooling system of an automobile.

The staff assumes that decontamination of the reactor building will be largely completed before fuel removal activities begin. Thus, during these activities the contribution to worker exposure from building background radiation should be small. The major contribution to worker exposure will be the general background of 2 to 3 mR/hr at the surface of the transfer canal water during underwater operations to disassemble the reactor. In addition, it will be necessary to work in radiation fields as high as 150 to 200 mR/hr in performing some of the hands-on activities. The staff has estimated the time-averaged field for a typical worker during the defueling activities over a work shift to be 10 mR/hr. While some persons will be working in higher average fields and others in lower fields, it is the staff's judgment that this average value is appropriate for estimating radiation exposures for workers performing the defueling and primary cooling system cleanup activities. A total of about 100,000 to 300,000 person-hours of effort will be required, depending upon the conditions found during inspection.

Treatment of Liquid Waste

Liquids involved in the TMI-2 decontamination will require further processing to permit their safe disposal in accordance with the staff's proposed use of the effluent criterion in 10 CFR 50 Appendix I as discussed in Section 1.6.3.2; these liquids include those directly generated during the March 28, 1979, accident (accident water) as well as liquids contaminated during the cleanup operations.

The accident-generated water in the reactor building sump and the primary system cannot be left in its present condition and location if the cleanup effort is to proceed. Some of the alternatives considered for disposition of this water involve its cleanup through the use of filtration, ion exchange, evaporation and bitumenization techniques. Others include transfer from its present location to onsite storage facilities or processing the water for transport and disposal at a low-level radioactive waste disposal facility. As decontamination solutions are generated, they too must be either cleaned up, stored, or processed and shipped offsite. The alternatives considered for accident-generated water and decontamination solutions are discussed below.

Long-term onsite storage of the unprocessed accident water involves transfer from its present locations in the reactor building and primary system to storage tanks. This water could be transferred to tanks within the reactor or auxiliary fuel handling buildings, if available, or to newly constructed exterior tanks. In either case, the storage tanks would have to be heavily shielded to reduce radiation levels in areas near the tanks. Storage of the accident water onsite would defer cleanup and complicate the cleanup operation without contributing to its end goals. For these reasons, long-term onsite storage of unprocessed accident water is not considered a reasonable alternative.

Direct immobilization involves mixing unprocessed accident water with a binder materal such as Portland cement or vinyl ester styrene for either temporary onsite storage or offsite shipment to a commercial shallow land burial facility. Immobilization of accident water with cement would take about 5 years, produce about 7400 cubic yards of concrete, and require about 1900 shielded shipments from the TMI-2 site. This was not considered suitable for unprocessed accident water but may be used for the relatively small quantitives of decontamination liquids.

Several processes and systems are available for treating the liquid to remove the contaminants. The following processes were considered: (1) filtration, (2) ion exchange, (3) evaporation, and (4) bitumenization.

Filtration is applicable to TMI-2 liquid wastes as an initial step in a process. It is not a appropriate treatment process by itself because much of the radioactivity is in solution and thus is not removed by filtering.

Ion exchange, the same process used in household water softeners, involves the removal of ionic species from an aqueous phase. The ion-exchange media considered for use at TMI-2 include inorganic zeolites and other minerals and organic resins. Ion exchange is appropriate for accident water and some decontamination solutions. It is not appropriate for treatment of chemical decontamination solutions because the chemical nature of these liquids would lead to rapid breakdown and plugging of ion-exchange media.

Evaporation would separate the water from the non-volatile radionuclides and other impurities dissolved in the liquid waste. Most of the contaminants are retained in the concentrated solution (or bottoms) while the relatively clean water vapor is condensed to liquid which requires further processing in an ion-exchange system. Additionally, the concentrated solutions would have to be immobilized in a solidification system. Evaporation is only appropriate for treatment of TMI-2 liquid wastes with low to moderate concentrations of dissolved solids and low radionuclide concentrations.

Bitumenization combines evaporation and immobilization in one step. The radionuclides are immobilized in an asphalt-like material (bitumen) and the vaporized water removed from the liquid waste is condensed for further treatment in an ion-exchange system. Bitumenization is only appropriate for TMI-2 liquids of low to moderate radioactivity concentration with at least 5 percent solids content by weight.

The basic processes described above were combined for purposes of analysis into nine different treatment systems. The zeolite-based ion-exchange systems are best suited for cleanup of accident water and water used for decontamination which would contain a high concentration of radionuclides. Evaporator/resin and bitumen/resin systems are considered suitable only for cleanup of chemical decontamination solutions due to the complex chemical nature of these liquids, their solids content, and moderate radionuclide concentrations. There are no practical systems for removal of the tritium present in the accident water.

Ten alternatives were evaluated for disposition of processed accident water containing tritium. These alternatives consisted of:

- Long-term onsite storage either as bulk liquid or as cement slabs. Since it would take about 200 years for the contained radioactivity to decay to innocuous levels, and storage merely defers ultimate disposition, these alternatives were not considered suitable.

- Onsite disposal of cement slabs by placement in shallow land burial trenches or injection of liquid water deep underground. These alternatives would present novel technical and regulatory issues and would also convert the site to a permanent repository. For these reasons, onsite disposal was not considered suitable.

- Shipment offsite for disposal in a shallow land burial facility, injection deep underground or disposal in the ocean. There are no regulatory obstacles to transport of bulk liquids containing low-level radioactivity but special permits would have to be obtained to dispose of the liquid in a deep injection well or the ocean. Immobilization of the liquid for shallow land burial is within the scope of current regulations but states may prevent the use of burial sites, due to space limitations, for such a large quantity of material with such a low level of radioactivity.

- Controlled release as liquid to the Susquehanna River, or as a vapor to the atmosphere or simultaneous releases to the river and to the atmosphere. Processed water can be diluted with available cooling tower blowdown water and discharged to the river at rates controlled to satisfy 10 CFR Part 20 and 10 CFR 50 Appendix I criteria. Placement of the water in a lined pond on the Island where it would evaporate would reduce the tritium concentration to levels approaching background in the river over a relatively short time (3-5 years); the remaining water could be retained in the pond for the lifetime of the plant. The water could also be injected into the mechanical draft cooling tower where some of the water would be vaporized and released to the atmosphere and the rest released to the river via the cooling tower blowdown.

Packaging, Handling, Storage, Transportation and Disposal of Radioactive Waste and Fuel

The wastes resulting from the accident and from decontamination activities are not all in a form acceptable for onsite storage or offsite disposal. It will therefore be necessary to treat some of these wastes. The treatment alternatives considered depend on the physical form of the material. Combustible trash can be incinerated to reduce volume by factors of 80 to 100, but incineration leads to the generation of ash that must be immobilized prior to disposal. The addition of an immobilizing agent results in a net effective volume reduction of 40 to 50. Noncombustible trash can be compacted to reduce volumes by a factor of about 5. Contaminated equipment and hardware can be disassembled and mechanically sectioned for volume reduction.

Treatment of contaminated liquids by filtration, ion-exchange, and evaporation leads to the generation of spent filter cartridge assemblies, contaminated ion-exchange media, and evaporator bottoms or sludges. At present, filters, ion-exchange media and sludges are accepted in the dewatered condition at some commercial low-level waste disposal facilities, but some others require that these materials be in a solidified form. Evaporator bottoms also must be solidified for disposal at all sites. An NRC order currently requires that contaminated ion-exchange resins from EPICOR II be solidified before shipment to disposal facilities. However, there are other alternatives to solidification that immobilize the waste, such as shipping the high-specific-activity portion of these and other wastes generated in the future to Department of Energy facilities for processing and disposal.

The wastes will be packaged prior to onsite storage or shipment to a waste disposal facility. The disposable containers considered for packaging include 55-gallon drums, wooden boxes for

low-activity material, and large steel containers with capacities as great as 200 ft^3. Special casks are required for fuel assemblies and could be required for certain spent filter cartridges, expended ion-exchange media, and sludges. The package handling techniques depend on the radiation level, size, and weight of the disposable container. Low-activity materials in drums or wooden boxes are handled manually and are transferred within the facility using forklifts, motorized pallets, or other package-handling equipment. For higher-activity disposable containers, semiremote and remote handling systems are used so as to reduce radiation exposure to personnel.

In general, all waste will be transferred to an onsite staging/storage facility prior to shipment offsite. These facilities may be shielded enclosures. The facilities are divided by container type and surface radiation level.

Shielded packages will be used for shipping spent fuel, contaminated hardware, certain spent filter cartridges, expended ion-exchange media, and other higher radiation level wastes. The number of shielded waste shipments is estimated to be between 220 and 690, and the number of unshielded shipments is estimated to range from 130 to 310. The range reflects the uncertainties inherent in the estimates as a result of the fact that the actual volume of waste generated will depend on which decontamination and waste treatment alternatives are implemented.

Disposal options considered include:

- Disposal of high-level wastes, high-specific-activity wastes and transuranic wastes in a special offsite waste repository,
- Disposal of the lower range of the high-specific-activity wastes by intermediate depth burial; and,
- Disposal of the low-activity wastes in shallow land burial.
- Storage of irradiated fuel either onsite or offsite pending final disposal or reprocessing.

S.4 Environmental Impacts of the Cleanup

The principal environmental impacts that can be expected to occur as a consequence of the cleanup activities at TMI-2 are indicated below.

Occupational Doses and Health Effects

Decontamination workers at the plant will receive a total cumulative radiation dose estimated at between 2000 and 8000 person-rem for the whole cleanup program. Using the NRC staff's health effect risk estimators, the health effect estimates corresponding to these doses range from 0.3 to 1 additional deaths among these workers due to cancer and from 0.5 to 2 additional genetic effects among their descendents. A summary of the minimum and maximum estimates by the staff is given below in Table S-1. These ranges are broad because of uncertainties of the plant conditions and the amount of work that will be needed to decontaminate the reactor building and its contents. The occupational dose to each worker will be limited to 3 rem/quarter in accordance with 10 CFR Part 20; however, the exact dose to any one individual cannot be predicted because work assignments have not been made.

Offsite Doses and Health Effects from Projected Cleanup Activities

The total-body dose to the individual that may receive the maximum exposure offsite from gaseous and liquid releases from the cleanup operations during any year will not exceed about 15 mrem (Appendix R). The increased risk that this dose would cause a fatal cancer in the individual who received it is about one chance in 100,000. The increased chance of that dose causing genetic effects to offspring of the maximum exposed individual is about one chance in 20,000. These risks are small compared to the current normal incidence rates for fatal cancers and genetic effects among the poulation. Public health statistics indicate that in the United States, one person in five probably will die of cancer and that the normal occurrence of hereditary disease in offspring is about 1 in 17. For the general population within 50 miles of the plant, the

Table S-1. Estimated Occupational Doses and Resulting Health Effects as a Result of TMI-2 Cleanup Operations

Major Cleanup Operation	Cumulative Occupational Dose (person-rem)		Potential Fatal Cancers		Potential Genetic Effects in Offspring	
	Minimum	Maximum	Minimum	Maximum	Minimum	Maximum
Building and equipment decontamination	1,000	3,600	0.13	0.5	0.3	0.9
Fuel removal and primary system decontamination	900	4,100	0.1	0.5	0.2	1.1
Treatment of radioactive liquids	40	120	0.006	0.016	0.01	0.03
Packaging, handling, storage and transportation of radioactive wastes	75	520	0.009	0.07	0.02	0.14
Total[a]	2,000	8,000	0.3	1	0.5	2

[a]Totals are rounded to one significant digit.

total cumulative dose from expected releases would be in the range of 10 to 30 person-rem; this is less than 0.01 percent of the 255,000 person-rem annual radiation dose to the same population from background.

An individual onlooker who spent three minutes at an average distance of 3 ft from a truck loaded with radioactive waste might receive a dose of up to 1.3 mrem. The probability that this dose would cause an increased chance in fatal cancer in the individual who received the dose is about 1 in 1 million. The added probability of genetic effects to offspring of the exposed individual is about 1 in 200,000. The estimated 700,000 persons who reside along the longest (2750-mile) route from TMI to the farthest disposal site might receive a cumulative population dose within the range of 20 to 50 person-rem for all TMI decontamination waste and fuel shipments.

The potential for effects on aquatic organisms in the Susquehanna River and in Chesapeake Bay has been evaluated for both controlled and accidental discharges of processed and unprocessed water to the river. Low concentrations of Cs-137 could persist in sediments in both the river and the Bay for some years following an accidental discharge of unprocessed water from TMI-2, but the levels would be so low as to have no radiation effects on aquatic species or on man and would not be detectable in sediments beyond the immediate site vicinity. For controlled releases of processed water there would be no significant effects on man or aquatic species.

Postulated Accidents

The accidents considered possible by the staff range from the more likely but low consequence failure of a HEPA filter to the extremely unlikely occurrence of a leak in the reactor vessel that leads to uncovering and overheating of the core with the reactor vessel head removed.

Building ventilation systems are equipped with two or more HEPA filters in series to remove small particles of dust from the air being exhausted to the atmosphere. Failure of a HEPA filter would allow some of the radioactive dust to escape until the failure was detected and the ventilation system shut down. An accident of this nature in the AFHB could result in a total-body dose of 0.0084 mrem to the maximum exposed individual offsite.

Although extremely unlikely, the unprocessed reactor building sump water might leak out of the reactor building. In the improbable event that all the water reaches the river, the concentration of radionuclides would be orders of magnitude below maximum permissible concentrations for unrestricted areas. Percolation through the ground to the river is calculated to take more than one year, and monitoring wells around the reactor building will provide early indication of groundwater contamination, thus giving sufficient time to take action to mitigate the consequences. Only tritium would reach the river within 1 year, whereas other isotopes, such as Sr-90 and Cs-137, would be greatly delayed because of adsorption onto the soil.

While the staff could not hypothesize a credible mechanism for a leak in the reactor vessel, the consequences of such an occurrence were evaluated. Since from many hours to several days would be available to take corrective action should loss of cooling be developing, the leakage from the reactor vessel into the reactor building could be limited to a small fraction of the current inventory of radioactivity. The worst conceivable situation would occur if all the Kr-85 (45,000 Ci) and most of the cesium (470,000 Ci) still in the core were released to the reactor building. The resulting activity levels inside the reactor building would then be comparable to, or less than, the levels inside the reactor building following the accident on March 28, 1979. The releases and offsite doses for this incredible bounding event would be comparable to those which have been estimated for cleanup of the reactor building following the accident (i.e., overall releases and offsite doses for the reactor building cleanup would be doubled).

Potential Releases Caused by External Events

The potential for the release of radioactive materials as a result of flood, tornado, or aircraft impact was evaluated. A flood somewhat larger than the Hurricane Agnes flood of June 1972 would overtop the levee surrounding the plant, but waste storage facilities are protected by thick concrete, and it is not very likely that they would be breached. Nor are these facilities likely to be affected by tornado or impact of an aircraft; however, the exposed processed water storage tanks would be vulnerable if these low-probability events occurred. Release of this water containing less than 0.6 μCi/mL of tritium would not present a public health hazard if the water reached the Susquehanna River because maximum permissible concentration limits (10 CFR Part 20) are not likely to be exceeded at the nearest public drinking water supply.

Psychological-Socioeconomic Effects

The level of psychological distress among some members of the communities surrounding the plant increased immediately following the accident but had considerably diminished by midsummer 1979. Low levels of distress will probably continue during the cleanup process, but no long-term psychological effects on the great majority of the community are predicted. Nevertheless, the long-term nature of the cleanup program presents the potential for increasing psychological distress for some people; consequently, completing the cleanup as expeditiously as safety will permit is desirable.

Social impacts during the cleanup could include possible resistance to consumption of agricultural and fishery products that the public may think are radioactively contaminated. Those who make all or part of their living from seafood taken from Chesapeake Bay or those involved in agricultural production are likely to be affected to the largest degree; but losses, if any, should be of short duration.

The disposition of processed accident water is of concern to many individuals. The potential economic impact on Chesapeake Bay activities (sea food industry and recreational uses) if the processed accident water were diluted and a controlled release made to the Susquehanna River is a matter of special concern. The State of Maryland is undertaking a study of the possible economic impacts on the marketing of seafood from the Chesapeake Bay if the public were to perceive a health hazard from controlled release of the processed accident water. The NRC staff is of the opinion that until such a study is completed, no decision should be made regarding the disposition of the processed accident water (unless an emergency arises which requires early disposition). Adequate storage capacity exists for interim retention of the processed accident water onsite.

Although the number of truck shipments necessary to carry solid radioactive wastes to disposal sites will be large (ranging from 350 to 1000), the shipments will be made over a long period of time and reduction of the marketability of residential property near the route through Middletown, Pennsylvania, should be temporary.

Radiological Environmental Monitoring Program

Monitoring around the TMI site and in nearby communities during decontamination of TMI is being performed by (1) the U.S. Environmental Protection Agency as the lead federal agency, (2) the Commonwealth of Pennsylvania, (3) the U.S. Department of Energy, (4) the Nuclear Regulatory Commission, (5) the State of Maryland, and (6) Metropolitan Edison Company (the licensee).

In addition to their own direct monitoring, the Department of Energy and the Commonwealth of Pennsylvania have sponsored a Community Radiation Monitoring Program that involved people in 12 communities within about five miles of TMI.

The very comprehensive radiological monitoring program provided by the cooperation of the above organizations consists of direct and indirect measurements of exposure rates and sampling of all appropriate media at numerous locations in the offsite area within 21 miles of TMI-2. Exposure rate measurements are made using recording and nonrecording rate meters and thermoluminescent dosimeters (TLDs). Samples of air, soil, vegetation, milk, fish, aquatic plants, sediments, and water are collected and analyzed for specific radionuclides and for gross beta and gamma emissions.

The results of all monitoring programs are reported to the EPA, which is responsible for coordinating offsite monitoring around TMI-2 and for compilation and dissemination of the resulting data to the public.

Economic Costs for Cleanup of TMI-2

Estimates of the relative cost of alternatives considered for each phase of the cleanup of TMI-2 have been developed, excluding costs not directly associated with the actual decontamination work. Examples of the costs not included are the costs of the support activities essential to the whole process, cost of replacement power, interest charges and inflation. The estimates also are in constant 1980 dollars and do not include the effect of inflation. These cost estimates were made for the purpose of comparing the various alternatives that were considered feasible. A total cost estimate for the cleanup of TMI-2 has not been made by the staff.

PREFACE

The NRC staff appreciates the many helpful comments received from the public and government agencies on the draft Programmatic Environmental Impact Statement (PEIS) which was issued in August 1980 for public comment. Our efforts to be responsive to these comments have resulted in a variety of changes in the PEIS, particularly in Chapters 5 through 8 which some persons found confusing. While the organization of those chapters in the draft statement enabled the reader to follow the expected chronological sequence of the cleanup activities, it also had the disadvantage of scattering information on particular subjects, such as processing contaminated water from several plant locations, through various parts of the document. We have therefore reorganized the material in these chapters so that discussions of similar activities in this final PEIS are grouped together.

To further aid the reader we have simplified some of the technical descriptions, updated information, and included additional illustrations. Relative economic cost estimates of the alternative methods of performing the cleanup of TMI-2 have also been included, as promised when the draft statement was issued. However, we do not regard the addition of cost information (or the other modifications mentioned above) as a substantial change in the content of the document. Our primary responsibility is to ensure that the cleanup activities are conducted in accordance with NRC's mandate to ensure the health and safety of the public and to protect the environment.

Bernard J. Snyder, Program Director
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Office of Nuclear Reactor Regulation

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FOREWORD

This programmatic environmental impact statement was prepared by the U.S. Nuclear Regulatory Commission, TMI Program Office, Office of Nuclear Reactor Regulation (the staff), pursuant to the Commission's November 21, 1979, Statement of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement and the requirements of the National Environmental Policy Act of 1969 (NEPA).

The NEPA states, among other things, that it is the continuing responsibility of the Federal Government to use all practicable means, consistent with other essential considerations of national policy, to improve and coordinate Federal plans, functions, programs, and resources to the end that the Nation may:

- Fulfill the responsibilities of each generation as trustee of the environment for succeeding generations.
- Assure for all Americans safe, healthful, productive, and esthetically and culturally pleasing surroundings.
- Attain the widest range of beneficial uses of the environment without degradation, risk to health or safety, or other undesirable and unintended consequences.
- Preserve important historic, cultural, and natural aspects of our national heritage, and maintain, wherever possible, an environment which supports diversity and variety of individual choice.
- Achieve a balance between population and resource use which will permit high standards of living and a wide sharing of life's amenities.
- Enhance the quality of renewable resources and approach the maximum attainable recycling of depletable resources.

Further, with respect to every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, Section 102(2)(C) of the NEPA calls for preparation of a detailed statement on:

(i) the environmental impact of the proposed action;

(ii) any adverse environmental effects which cannot be avoided should the proposal be implemented;

(iii) alternatives to the proposed action;

(iv) the relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity; and,

(v) any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented.

Information for this statement was obtained from the licensee's Environmental Report and Final Safety Analysis Report or from the staff's Final Environmental Statement for the Operating License, dated December 1976, and from new information provided by the licensee or independently developed by the staff. This information is available to the public. Any comments by interested persons received on this information have been considered by the staff. In conducting the required NEPA review, the staff met with the licensee to discuss items of information provided, to seek new information from the licensee that might be needed for an adequate assessment, and generally to ensure that the staff had a thorough understanding of the proposed cleanup operations. In addition, the staff sought information from other sources that would assist in the

evaluation and visited and inspected the project site and surrounding vicinity. Members of the staff met with State and local officials charged with protecting State and local interests and held scoping meetings with the public in Harrisburg, Pennsylvania, on January 29, 1980, in Middletown, Pennsylvania, on February 12, 1980, and in Baltimore, Maryland, on February 15 and March 20, 1980.

On the basis of the foregoing and other such activities or inquiries as were deemed useful and appropriate, the staff made an independent evaluation of the TMI-2 cleanup plans and operations. This evaluation led to the publication of a draft programmatic environmental impact statement, prepared by the Office of Nuclear Reactor Regulation. It was circulated to Federal, State, and local governmental agencies for comment, and a summary notice of the availability of the draft environmental statement was published in the Federal Register. Interested persons were invited to comment on the draft statement. In addition, the staff held a total of 31 meetings with the public, local officials, and interested organizations during the comment period on the draft statement in order to explain the staff's assessments and to solicit verbal comments from individual members of the public and to respond to these comments at that time. Verbatim transcripts were made at ten of the major public meetings from which the staff, after review, obtained further comments which were expressed verbally by the public. These verbal comments were received as if they were written submittals.

After receipt and consideration of comments on the draft statement, the staff prepared this final environmental statement, which includes a discussion of comments generated on the draft statement and the responses thereto. The comment letters received on the draft statement appear in Appendix A; responses to the comments are contained in Section 13.

Single copies of this statement may be obtained as indicated on the inside front cover.

Messrs. Paul Leech and Oliver Lynch are the Environmental Project Managers for this statement. Should there be any questions regarding its contents, they may be contacted at the following address:

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1. INTRODUCTION

1.1 THE PURPOSE AND SCOPE OF THIS STATEMENT

On November 21, 1979, the Nuclear Regulatory Commission announced its decision to prepare a programmatic environmental impact statement (PEIS) on the decontamination and disposition of radioactive wastes resulting from the March 28, 1979, accident at Three Mile Island Nuclear Station Unit 2 (see Appendix B). The PEIS is intended to provide an overall evaluation of the environmental impacts that could result from these activities, beginning when the plant conditions were stabilized after the accident and continuing through completion of the cleanup from the accident. For purposes of this document, the term "cleanup" is used to mean decontaminating and defueling the plant, and disposition of radioactive wastes.

As stated in the Commission's notice, "An overall study of the decontamination and disposal process will assist the Commission in carrying out its regulatory responsibilities under the Atomic Energy Act to protect the public health and safety as decontamination progresses. It will also be in keeping with the purposes of the National Environmental Policy Act to engage the public in the Commission's decision-making processes, and to focus on environmental issues and alternatives before commitments to specific cleanup choices are made."

The Commission directed the staff to include in the PEIS an overall description of the activities and a schedule for their completion, along with a discussion of alternatives considered and the rationale for choices made. This information has been included to the extent it is presently available from the licensee.* However, there are many areas of uncertainty regarding the cleanup operation, as the Commission recognized (Appendix B). For example, the precise condition of the reactor core will not be known until the reactor vessel has been opened. Where such is the case, the staff has described and assessed the probable or bounding situations. If, when more information becomes available, proposed activities are found to be significantly beyond the scope of these assessments, appropriate supplements to the PEIS will be issued.

The proposed scope of the statement was discussed with representatives of the President's Council on Environmental Quality, the licensee, and several state agencies. Early in the process of developing the draft PEIS, scoping sessions were also held with the public in Harrisburg and Middletown, Pennsylvania, and in Baltimore, Maryland. The comments received were particularly helpful to the staff in understanding public concerns (see Sec. 1.4) about certain methods of decontamination and waste disposal being proposed by the licensee. Similar comments and many more specific to the Draft PEIS (see Appendix A) were received during the public comment period which followed the issuance of that document in August 1980. After publication of the draft PEIS for comment, the staff held 31 meetings with the public, local officials, and interested organizations to obtain first-hand the comments and concerns of participants at these meetings and to have an interchange of ideas.

One major issue, the ultimate disposition of TMI-2, is of interest to the federal, state, and local governments as well as the licensee and the public. However, the disposition of the facility--whether to decommission or restore it to a condition acceptable for licensed operation--is not within the scope of this statement. If a decision to decommission Unit 2 is made before decontamination of the reactor building is far along, it is possible that different methods and chemicals would be used to speed the cleanup of some portions of the plant, as opposed to those that might be chosen to minimize damage to the equipment. Such possibilities are considered in Section 2.2. However, a decision to either restore or decommission the facility would probably not occur until a detailed inspection and engineering assessment is made of the nuclear steam supply system and further information is known about its condition. To make this inspection

*The term "licensee" or "Met-Ed" in this document refers to Metropolitan Edison Company, the principal owner (50%) and operator of the plant; Jersey Central Power and Light Company, and Pennsylvania Electric Company, each of which own 25%.

requires that the core be removed and the reactor cooling system be decontaminated to a level that will not cause excessive radiation exposure to people either onsite or offsite. However, the staff has assessed in Section 2.2 the alternative of decommissioning even in the event a decision to decommission is made early in the cleanup.

Decontamination of the auxiliary and fuel handling buildings began prior to the Commission's decision to issue the PEIS. For this reason, some of the discussion in Section 5 is historical, and no alternative methods are presented for work already done. With regard to decontamination efforts not yet undertaken, the staff has based its analysis on the licensee's plans where they are available, as well as alternatives that the staff has independently developed and assessed. Current estimates of economic costs for the proposed activities and alternatives were not available from Met-Ed or other sources in time for inclusion in the draft statement. Cost estimates are provided in this final statement.

To enable preparation of this document, specific cutoff dates were selected upon which to base the assessments. Changes in plant conditions that may have occurred since those dates do not materially affect the evaluations.

1.2 HISTORY OF THE PLANT AND ITS PRESENT STATUS

On November 4, 1969, following a public hearing, a construction permit for TMI-2 was issued by the Atomic Energy Commission. An operating license for Unit 2 was issued on February 8, 1978.*

Between issuance of its operating license and March 28, 1979, TMI-2 had operated for about 95 effective full-power days (or the equivalent of 3165 MW-days per metric ton of low-enriched uranium oxide). Prior to the accident, the unit had been operating without interruption since March 7, 1979. At full power, the TMI-2 system would supply 890 MW (2770 MW thermal) of electric power to the utility's transmission system; together, the two units had a 1700-MW electric generating capacity. TMI-2 was operating at 97% power when the accident occurred.

TMI-2 has a pressurized-water reactor with the nuclear steam supply system (NSSS) provided by Babcock & Wilcox Company. Under normal reactor operation, the primary coolant water inside the reactor vessel is maintained at around 575°F and 2200 psi. Heat generated by the fission process within the reactor core is removed by means of the primary coolant to two steam generators where steam is produced to operate a turbine.

The reactor vessel and components (Fig. 1.1) that are exposed to radioactivity are located within the reactor building (also referred to as the containment building). The turbine generator, feedwater system, and electrical generation equipment are housed inside the turbine building. The remaining support systems, e.g. the high-pressure injection pumps and the makeup and let-down systems, are located in the auxiliary building adjacent to the reactor building (Figs. 1.2 and 1.3).

At about 4:00 a.m., March 28, 1979, a series of feedwater pumps in the turbine building 'tripped' (stopped operating), resulting in a turbine-generator trip and automatic shutdown of the reactor. An increase in pressure within the pressurizer activated the pilot operated relief valve (PORV), which served to prevent overpressure of the primary coolant system by releasing steam. However, after the primary system pressure was reduced to the PORV closure set point, the PORV failed to close. The light on the control room panel indicating the PORV status showed that electric current to the PORV solenoid had been terminated, which would normally indicate that closing of the valve had been accomplished. This inadequate and misrepresented information prompted the operators to wrongly assume, for more than two hours, that the PORV had been closed. During this time, the continual loss of primary coolant and reduction of pressure in the system, coupled with the operators' shutting off of the high pressure injection (HPI) system (which had been adding water to the primary system), led to a drop in the water level, uncovering some of the reactor core. Rupture of fuel rod cladding was indicated by the release of excessive radioactivity into the primary coolant as well as the accumulation in the reactor vessel of hydrogen gas produced from the chemical interaction between the fuel rod zircaloy cladding and steam. After initial uncertainties regarding the presence of a hydrogen bubble in the reactor vessel, and its implications, forced cooling was established at 8 p.m. on March 28, 1979. On April 3, 1979, the decay heat level was already down to 5 MW and the primary system was secured to operate at 281°F and 1050 psi with reactor coolant pump 1A. On April 7, 1979, the system pressure was further reduced

*TMI-1, located at the same site, was issued an operating license on April 12, 1974.

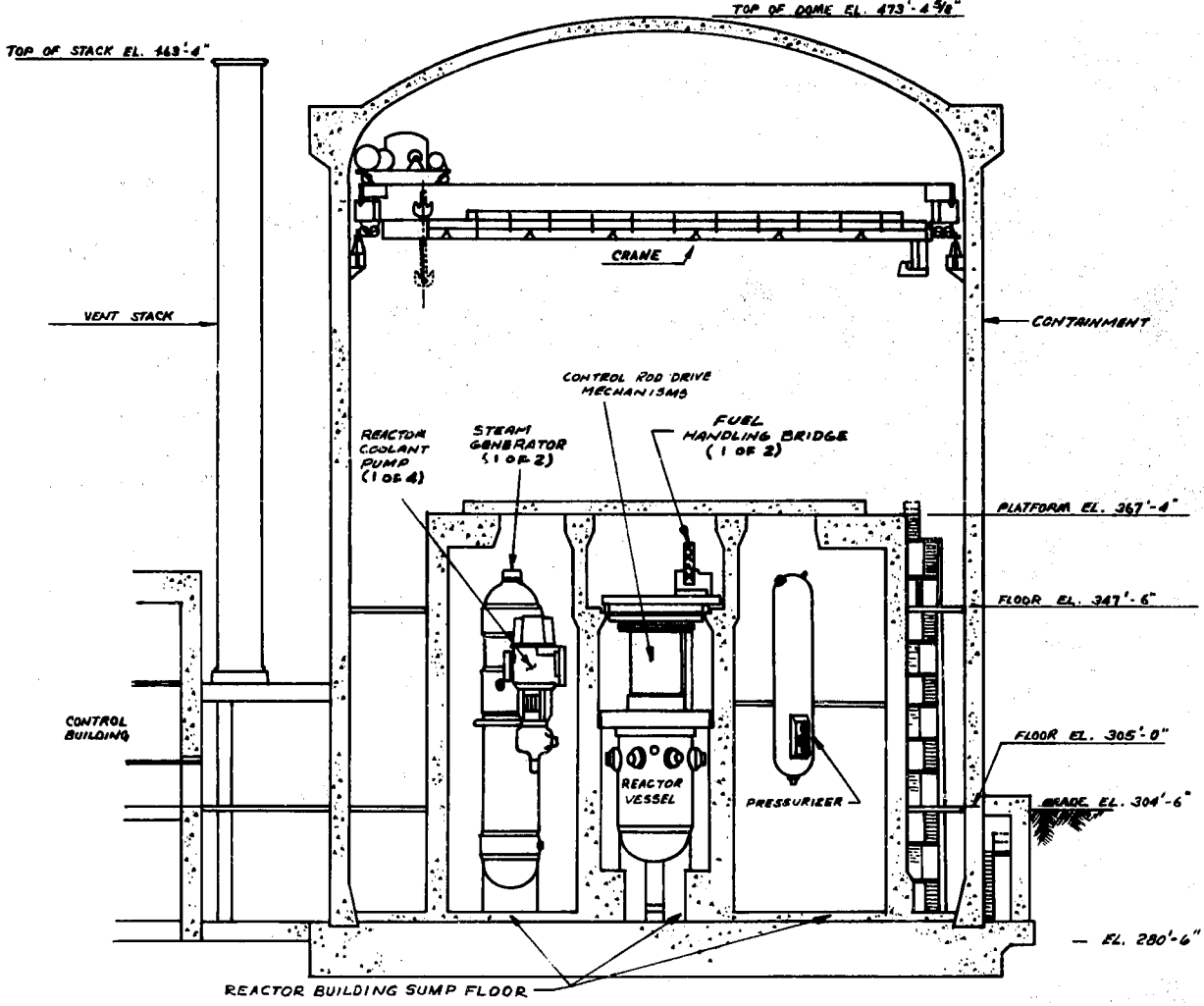

Figure 1.1. Reactor Building and Major Components of Primary System.

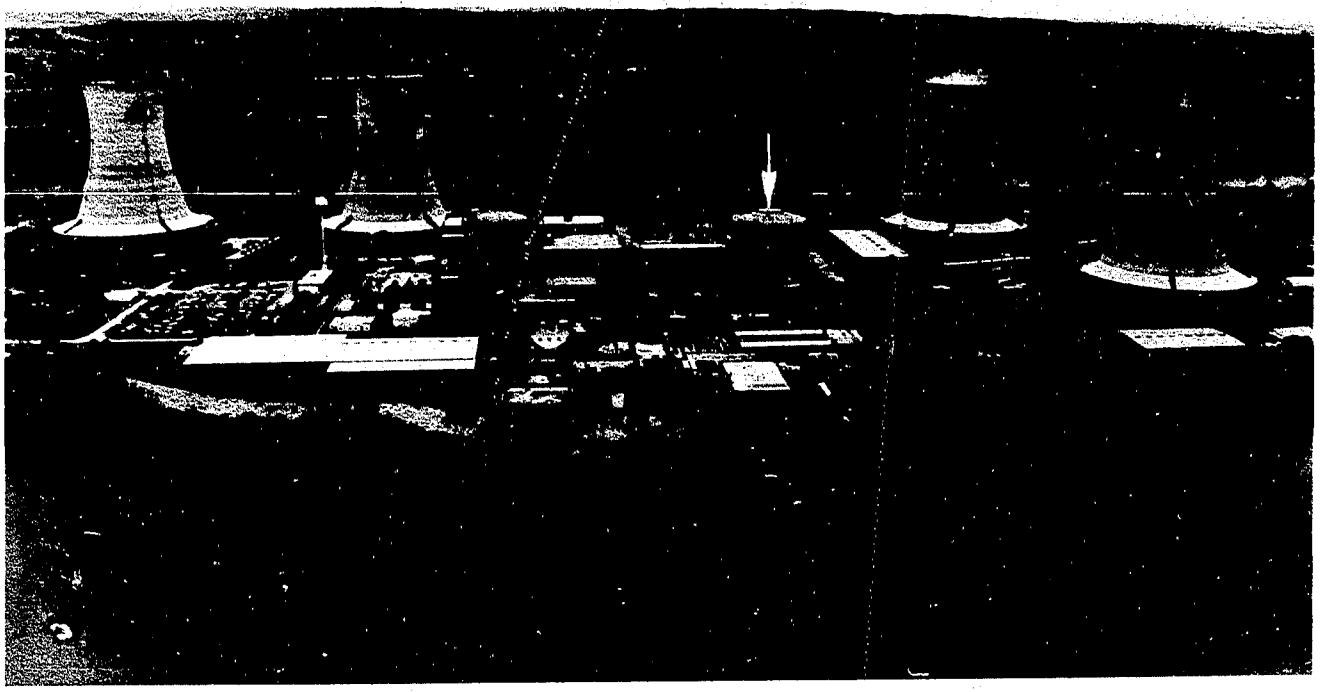

Figure 1.2. Aerial View of the Three Mile Island Station. Three Mile Island Nuclear Generating Station is located on Susquehanna River, about 10 miles south of Harrisburg, Pa. Damaged Unit-2 reactor is housed in the cylindrical building (marked by arrow) on the right of central plant complex.

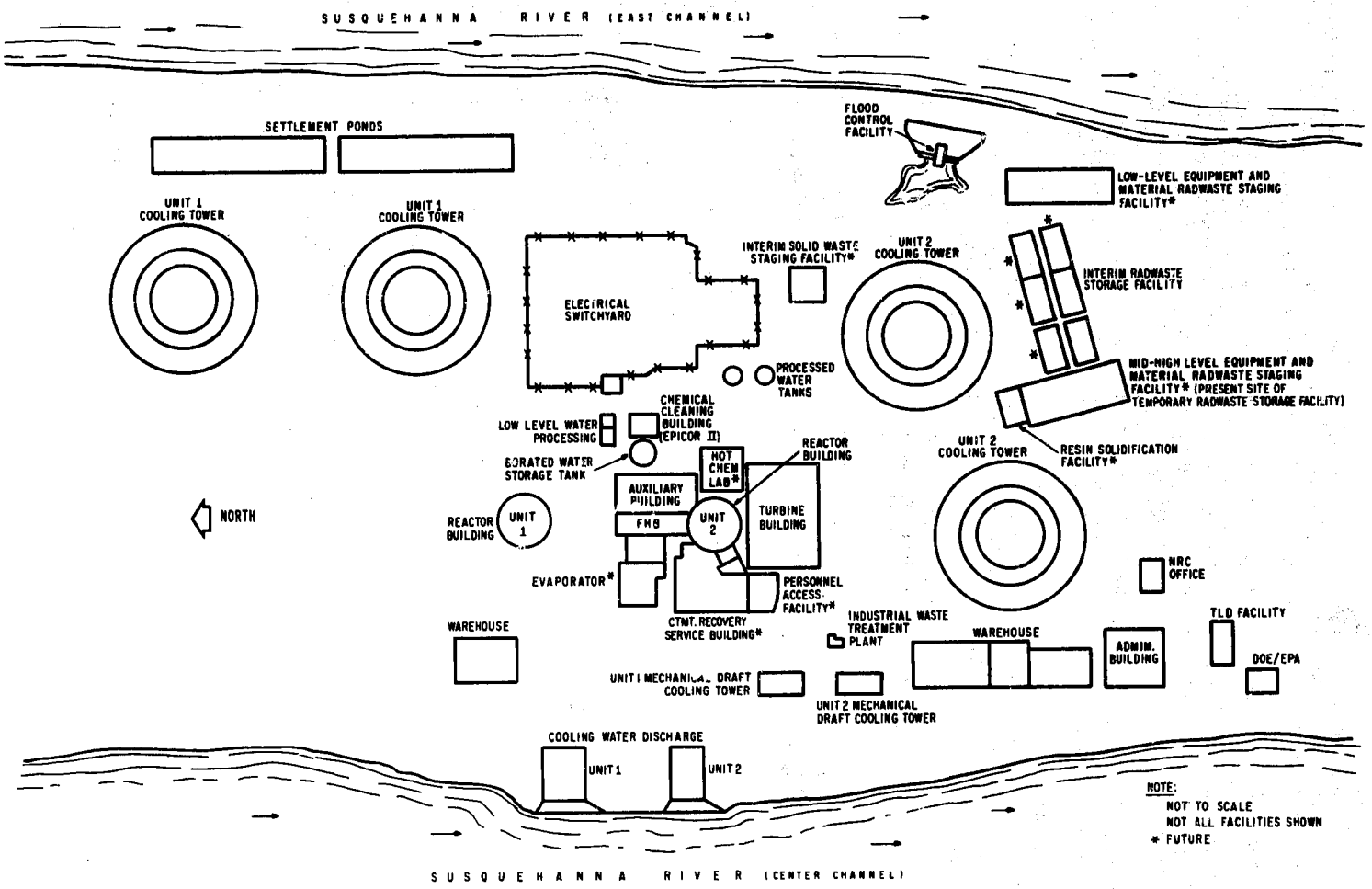

Figure 1.3. Schematic Plan of the TMI-2 Site. (Not all features shown.)

to 400 psi, and on April 27, 1979, cooling through natural recirculation was established without the need of primary system pumps for operation.

A more detailed description of the accident and its consequences between March 28 and April 3, 1979, can be found in the "Report of the President's Commission on the Accident at Three Mile Island,"[1] and "Three Mile Island: A Report to the Commissioners and the Public."[2]

As of December 13, 1980, TMI-2 had been shut down for 628 days and decay heat had decreased to about 70 kW. The primary system temperature was then about 120°F and pressure was about 94 psi. Decay heat was removed by natural recirculation, with the "A" steam generator steaming to the main condenser and the reactor coolant system loops releasing heat to the reactor building atmosphere. On January 5, 1981, the licensee stopped steaming the "A" steam generator by shutting a turbine bypass valve. This put the RCS in a "loss-to-ambient" mode of cooling, which is the transfer of reactor decay heat from both RCS loops to the reactor building ambient.[3] The reactor is subcritical, with a substantial shutdown margin below criticality provided by the reactor control rods and boron dissolved in primary coolant water.

Radioactive gases (primarily krypton-85) contaminated the reactor building atmosphere as a result of the accident. On June 12, 1980, the Nuclear Regulatory Commission approved the staff's recommendation that the licensee be allowed to decontaminate the building atmosphere by controlled purging. The purging operation began on June 28, 1980, and was completed on July 11, 1980, removing approximately 44,000 Ci of Kr-85. The first entry into the reactor building since the accident was made by two Met-Ed personnel on July 23, 1980, and by December 13, 1980, a total of five entries into the reactor building had been made to survey damage and acquire radiation data. This information is discussed further in Section 5.2.

The reactor building is flooded to a depth of about 8 ft with water containing radionuclides (about 500,000 Ci, primarily the cesium (Cs) isotopes 137 and 134); and most of the approximately 300,000 ft^2 of exposed surfaces are contaminated with a thin layer of radionuclides (of the order of 0.01 Ci/ft^2, primarily Cs-137 and Cs-134).

The auxiliary building was contaminated to a lesser extent by water and gas from the primary system and the reactor building. All of the "accident" water* in the auxiliary building has been processed, and the processing of water used in washing and flushing of contaminated equipment surfaces is continuing. As of December 1980, the surfaces of nearly all of the general areas in the auxiliary building had been decontaminated, and decontamination of the cubicles containing tanks and other equipment was about 80 percent complete. The level of atmospheric contamination was below the threshold for unrestricted worker access.

1.3 SUMMARY OF THE LICENSEE'S OBJECTIVES, PROPOSED ACTIONS, AND SCHEDULE

The major objectives of the licensee's TMI-2 decontamination and defueling plan[4] are to maintain the reactor in a safe state; decontaminate the plant; process and immobilize dispersed fission products; and remove and dispose of the reactor core, with maximum assurance of public health and safety. In December 1979, Met-Ed estimated that decontamination and defueling could be accomplished within a time span of about 2 to 2½ years from working entry into the reactor building, given no unusual technical, regulatory, political, or financial constraints,[4] at an approximate cost of $400 million.[5]

In November 1980, Met-Ed extended its schedule by 28 months[6] to reflect the impact of regulatory constraints and approval processes, and the company's diminished availability of funds for cleanup resulting from an order by the Pennsylvania Public Utility Commission to cease using operating revenues for this purpose.[7] In consideration of these developments, the utility company has also increased its estimate of the cleanup costs, including funds spent in 1979 and 1980, to $750 million (in 1980 dollars).[6]

Accordingly, the licensee has adjusted the cleanup efforts "to a level appropriate to the present and indicated situation while being careful not to adversely impact public health and safety." The objectives of its revised program[8] are to:

1. Maintain the plant in a safe condition with minimum but adequate operating personnel and site support staff.

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*"Accident" water is defined in Section 1.6.2.1.

2. Continue limited decontamination of the Auxiliary Building areas, lines, tanks.

3. Continue activities directed at cleanup of the reactor building water (sump and reactor coolant system).

4. Continue carefully selected planning, engineering and licensing activities aimed at Reactor Building decontamination, fuel removal, support of licensing submittals.

5. Support finalization of the Programmatic Environmental Impact Statement (PEIS).

6. Continue development of an appropriate Unit 2 Radiological Controls Program.

The revised schedule shown in Figure 1.4 represents the licensee's planning schedule, as of November 7, 1980, for completion of the first two major phases of the effort at TMI-2 by the spring of 1986. The licensee's third phase, reconstruction for operation, is not within the scope of this PEIS.

Phase I commenced shortly after the accident on March 28, 1979, with plant cooldown. The key events of Phase I include auxiliary building decontamination, accident water processing, krypton-85 purging from the reactor containment building, containment entry, construction of additional site support facilities, and reactor building decontamination. Several of these activities will extend in time past the start of Phase II.

Phase II will commence with preparation for reactor pressure vessel head removal. The primary milestones for Phase II are reactor pressure vessel head removal, fuel removal, and completion of the reactor coolant system decontamination.

The schedule will be significantly influenced by many factors that are not precisely known at this time. These include, but are not limited to, the exact radiation environment and core conditions encountered, continuing financial limitations and regulatory activities, craft labor and material availability, availability of offsite or onsite radwaste storage and processing, and availability of offsite disposal capability on a timely basis.

1.4 PUBLIC CONCERNS

In planning this environmental statement, members of the NRC staff held scoping meetings with public officials and interested members of the public to provide them an opportunity to identify subjects which should be addressed in the PEIS. These meetings were conducted at the locations listed below on the dates shown:

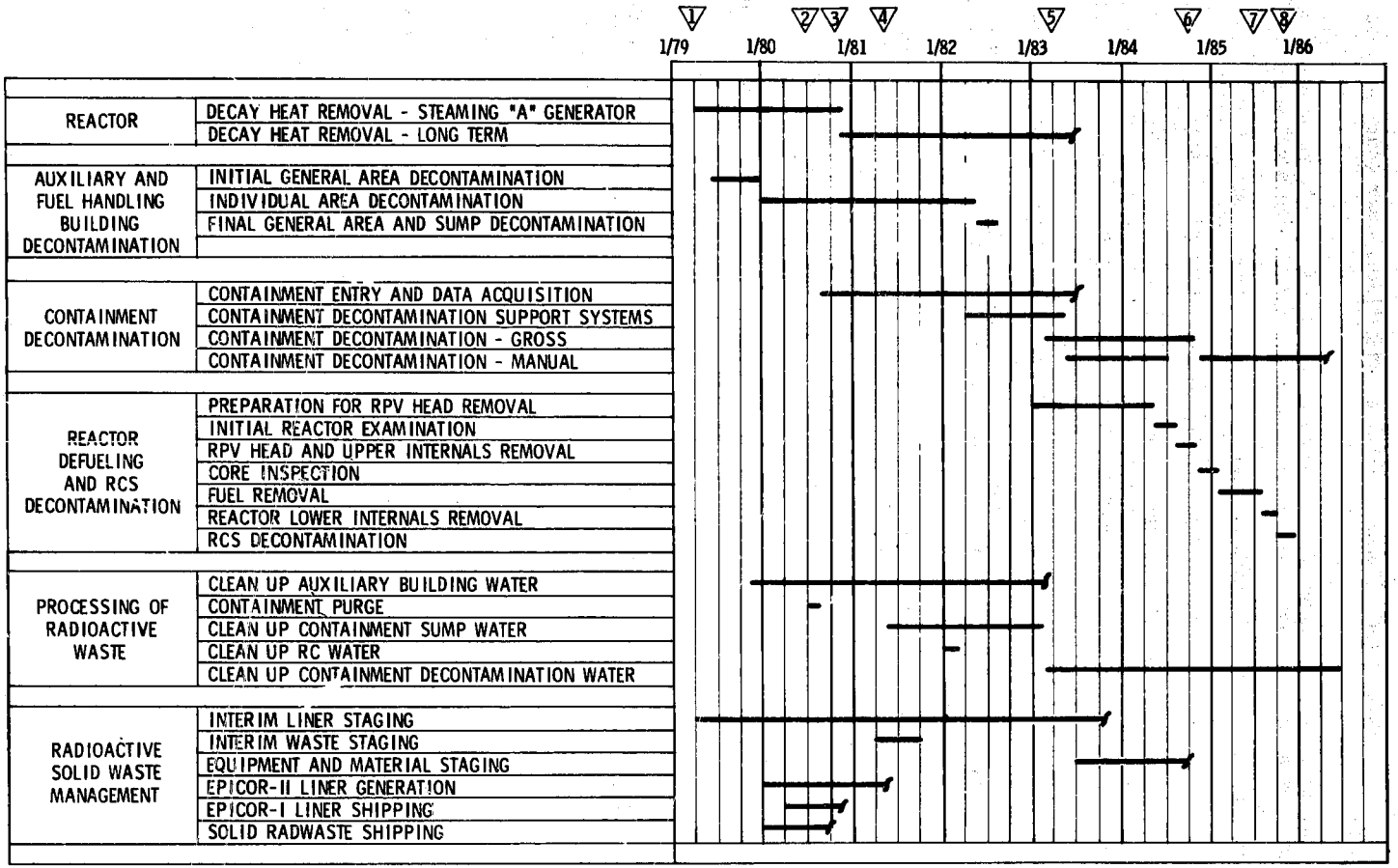
Harrisburg, Pennsylvania	January 29, 1980
Middletown, Pennsylvania	February 12, 1980
University of Maryland Baltimore Campus	February 15, 1980
Johns Hopkins University, Baltimore, Maryland	March 20, 1980

The comments expressed at these sessions clearly revealed a general concern about some of the possible decontamination methods and their potential impacts. Many people in the plant vicinity indicated apprehension about the relatively small releases of krypton gas from the reactor building which occasionally occur and stated their opposition to purging the gas. Other participants, notably those who attended the meetings in Baltimore, were more concerned about the possibility that releases of "accident water" to the river, even though largely decontaminated and greatly diluted, might eventually be authorized by the NRC. Possible adverse effects on drinking water supplies downstream and on the marketability of Chesapeake Bay seafood were the primary concerns relative to such releases of water from TMI-2. The availability of appropriate disposal sites, particularly for high-level waste, also was questioned and a number of comments indicated opposition to making TMI into a permanent waste repository.

Additional subjects for consideration in the PEIS were identified from transcripts of the meetings and subsequent letters as follows:

- The geographic area evaluated should include downwind and downstream areas to the extent that they will be affected by any releases.

- There should be a central agency to receive and correlate all air, water, and biotic monitoring data.



CONSTRUCTION, PROCESSING ACTIVITIES

ACTIVITY CONTINUED AS NEEDED

MILESTONE: 1 ACCIDENT, 2 CONTAINMENT PURGE, 3 INITIAL CONTAINMENT ENTRY, 4 PEIS ISSUED, 5 START CONTAINMENT DECONTAMINATION, 6 RPV HEAD REMOVED, 7 FUEL REMOVED, 8 RCS DECONTAMINATION COMPLETE.

Figure 1.4. Licensee's Planning Schedule for TMI-2 Cleanup, Phase I and Phase II. (Construction of support facilities not included.)

- Geology should be addressed both as relevant to hydrology and to storage of waste material at the site.
- Ecological studies should include food chains and possible cumulative effects on the biota of the affected areas, both terrestrial and marine.
- Independent sources should be considered in determining which of the various alternative methods should be used at various stages.
- The bibliography should include the methods, designs, and authorities considered.
- A separate chapter or appendix should deal with what are determined to be acceptable levels of radioactive or toxic emissions.
- The NRC should establish a citizens' advisory committee with funding for independent expert consultants to work with Argonne National Laboratory in preparation of the statement.
- The psychological stress issue needs to be considered.
- The impact of the restart of Unit 1 should be considered.
- The role of primary and secondary safety systems in the cleanup operation should be minimized.
- Releases that have occurred since the accident should be addressed.
- Section on offsite exposure and health effects should be expanded.
- There should be coherence in the overall decontamination plan.
- Acceptance by Department of Energy of all wastes until they become ready for shipment should be addressed.
- Specific radiological effluent release criteria for the cleanup process should be included.
- In case Met-Ed does not have sufficient funds to complete the cleanup, the PEIS should identify other sources of funds and the authorizations needed.
- Costs of alternative methods for decontamination should not be the basis for selection versus lowest environmental impact; the usual cost-benefit balance, in which the benefits of electrical energy generated usually outweigh the environmental costs, is not appropriate for this PEIS.

Most of the above subjects were considered by the staff in its preparation of the Draft PEIS, which was issued for public comment in August 1980, and in this Final PEIS. As stated in Section 1.1, a major concern regarding Unit 2--whether to restart or decommission it--is not within the scope of this document. However, decommissioning of TMI-2 and the resultant environmental impacts are briefly discussed in Section 2.2.

During the 90-day public comment period, which ended on November 20, 1980, a great variety of comments were received. They are reproduced in Appendix A. The staff's consideration of these comments is reflected in the revisions made for this Final PEIS and in specific responses which appear in Chapter 13.

To afford the public with increased opportunities to make their views known on the cleanup plans for TMI-2, the comment period was extended beyond the required 45 days to 90 days, and the staff conducted 31 meetings on the Draft PEIS during that period with various groups of people in Pennsylvania and Maryland. These meetings and their locations are listed in Appendix Y. A TMI Advisory Panel of citizens and public officials has also been established to provide communications directly to the NRC Commissioners throughout the cleanup of TMI-2.

1.5 DECONTAMINATION EXPERIENCE AT OTHER NUCLEAR FACILITIES

The removal of unwanted radioactive contamination from materials and equipment is a familiar and routine operation in the nuclear field for reducing radiation levels.[9] Decontaminations of

various types have been conducted since the 1940s and a vast expanse of experience and technology is available. Decontamination operations have been accomplished in the laboratory, in in-reactor and out-of-reactor test loops, on tools and building surfaces, and in entire nuclear reactor coolant systems. Many tests have been conducted to improve decontamination processes and techniques, and these improvements have been utilized on materials and equipment wherever radioactive contamination is found, for example, universities, hospitals, space satellites, nuclear laboratories, operating nuclear reactors, and fuel cycle facilities and laboratories.

The decontamination techniques employed can generally be classified into the broad categories of mechanical methods and chemical methods. The majority of cases usually require only the simpler mechanical techniques similar to those in ordinary housekeeping, i.e., wiping, brushing, scrubbing with or without detergents or scouring compounds, and wet or dry vacuuming. More complex mechanical techniques include water flushes, high-pressure water jets, surface-strippable films, sandblasting, ultrasonic removal, and even controlled removal of the actual surfaces of contaminated objects. While not strictly definable as techniques for decontamination, immobilizing contamination with coatings or covering with shielding materials are often employed for radiation level reduction.

Chemical decontamination techniques employ solvents to dissolve or suspend the radioactive contamination. The solvents can range from simple applications of water or water-containing detergents to complex formulations containing sequestering and chelating agents, surfactants, corrosion inhibitors, oxidizing or reducing agents, and inorganic or organic acids. Solvents have been formulated to remove specific types of radionuclides and for use on specific types of contaminated surfaces.

Decontamination experiences directly relevant to TMI-2 are described in this section to indicate the extent to which techniques and technologies are available and have been employed successfully (in many cases under conditions much more severe) for the various types of decontaminations that can be anticipated for TMI-2. The relevant experiences include:

- Entry into highly contaminated areas.
- Examples of building and equipment surface decontaminations applicable to anticipated decontaminations inside the TMI-2 containment.
- Cleanup and decontamination following fuel spills and core failures.
- Decontamination of entire reactor coolant systems to remove radioactive corrosion product films and absorbed fission product radioactivities.
- Post-decontamination operational history.

These experiences illustrate that the basic technologies for decontamination are well established and that available techniques can be modified to suit the conditions at TMI-2. Applicable experience in removing damaged fuel and core components is limited, and development of techniques specific to TMI-2 will be required. Some of the past experience relevant to the TMI-2 situation is summarized below.

1.5.1 Experience with Initial Entries into Highly Contaminated Facilities

The main objectives of the initial entry into a highly contaminated facility are to assess both the sources and magnitudes of the radiation fields present and to determine the physical condition of the facility and extent of damage so that the requirements needed for decontamination can be established. Thorough planning is required to effectively execute the initial entry.

1.5.1.1 Canadian NRX Reactor

A power surge in the NRX reactor occurred in 1952 causing failure of about 10% of the uranium fuel rods and releasing approximately 10,000 Ci of fission products to the cooling system, with subsequent spread throughout the reactor structure. The building basement acted as a reservoir for the highly radioactive water, and as the various wetted surfaces dried out, airborne contamination became a major problem. About one million gallons of radioactive water was generated.

Immediately after the incident, personnel were sent into the reactor building wearing masks and protective clothing to measure exposure rates. Rates of 10 roentgen per hour (R/hr) were

measured at the lower header room along concrete walls and floors and on the basement area walls and floors. Rates at the upper area ceilings, walls, and floors were 50 mR/hr, with local hot spots of up to 3 R/hr.

During decontamination operations, improved procedures for preparing personnel for entry into radiation field work areas evolved. These included background seminars describing the incident and delineating work areas and cleanup objectives; training in the use of protective clothing and equipment; use of mockups areas; use of closed-circuit TV (CCTV) to preview the contaminated area; and detailed descriptions of the work to be done, including procedures and estimated times required.[10-12]

1.5.1.2 Canadian NRU Reactor

In May 1958 during transfer of a failed fuel rod from the NRU reactor to a refueling flask, the rod stuck, coolant was lost and the fuel ignited. Portions of the rod fell into a maintenance pit where it set fire to materials in the pit. The portion of the rod that fell into the pit contained 200,000 curies (Ci) of fission products. Available monitoring instruments at the pit were offscale (> 1000 R/hr). Exposure rates elsewhere were 1 R/hr near the fuel flask, 1000 R/hr two ft above the reactor, 2.5 R/hr in the west reactor hall walls, 100 R/hr on the floor, and 30 mR/hr in offices, corridors, and change rooms. Airborne contamination was too high to be measured with accuracy. When the uranium began burning, the main exhaust fan on the roof was running but was shut off immediately to confine the contamination; nevertheless, some contamination was later found up to 5000 ft downwind.

A team of workers carrying buckets of wet sand buried the burning fuel in the pit within 15 minutes, and the highest dose received was 5.3 rem. A team of 35 workers working in one-minute shifts, using long-handled (24-ft) rakes, hoes and shovels, raked the fuel rod onto a tray, covered it with sand, and then removed it to a solid-waste burial ground. Despite the high radiation field (as much as 1000 R/hr), the average dose received was 1.4 rem and the maximum individual dose was 6.4 rem. The record indicates that extreme care was taken in timing each exposure and that the workers took advantage of all available shielding. The cleanup was done under emergency conditions without benefit of detailed preplanning. During the week after the incident a systematic attack on the decontamination problem was organized, and during the next two months some 600 workers took part in the cleanup.[13]

1.5.1.3 SL-1 Reactor

On January 3, 1961, a nuclear accident occurred at the SL-1, a military reactor at the National Reactor Testing Station (NRTS) in Idaho, resulting in fatal injuries to three reactor operators. During the period from January 3 to January 10, an NRTS disaster plan was in effect for the recovery of the three casualties and for the assessment of the reactor shutdown condition. The reactor core and pressure vessel were damaged beyond repair. The reactor building incurred only minor damage but was grossly contaminated. Only insignificant contamination was released to the environment. Initial operations started within minutes of the accident and were limited to extremely short entries (seconds) for determining radiation levels and casualty recovery. Health physicists wearing air packs measured exposure rates of 500 to 700 R/hr two ft inside the reactor room, 500 R/hr at the doorway to the operating floor, 10 R/hr in the control room, 250 mR/hr at the bottom of the outside steps, and 100 mR/hr at the guard house. Recovery teams recovered the first casualties within six days. Twenty-two persons received radiation exposures in the range of 3 to 27 R whole-body.

The second phase of the recovery operations was undertaken with the objectives of establishing the nuclear status of the reactor core, assuring that a renewed nuclear excursion would not occur, and planning for methods and equipment for the cleanup operations. This work included radiation surveys, the installation of neutron and gamma monitoring instruments, viewing the top of the pressure vessel head and the vessel interior to determine the condition of the core, and determining the water level in the pressure vessel. Because of the concern over the possibility of renewed nuclear excursions, all of these operations were performed by means remote to the reactor building interior. This phase was concluded in May 1961, with the determination that the pressure vessel contained no water and that subsequent nuclear excursions could be prevented by keeping the vessel dry.[14,15]

1.5.2 Experience with Decontamination of Building Interior Surfaces and Equipment

Methods and experience with routine low-level decontamination are well documented.[9,16,17] Items such as tools, equipment, walls, floors, glove boxes, and hot cells can be readily decontaminated in place. Most large nuclear facilities not only practice in-place decontamination, but also have special facilities to which smaller contaminated articles can be brought for decontamination.

Decontamination of small or completely contained but highly contaminated materials and equipment also is routinely conducted in the nuclear industry. However, experience is limited with high-level decontamination of building interior surfaces and equipment where the contamination has spread over large areas such as the entire interior surfaces of a reactor building.

1.5.2.1 Canadian NRX Reactor

Because of the large volumes of water involved (1 million gallons) in the NRX accident and the high activity (10,000 Ci), the water was stored in the reactor basement. The basement contained considerable equipment, including many instruments. The water was pumped through a pipeline to a disposal area 1¼ miles away and allowed to seep through the soil, which very effectively removed the activity.[12]

The containment surfaces were decontaminated by use of the following procedures:

- Flush with water, and pump water to disposal area.
- Flush with high-velocity hot water, sometimes with detergents.
- Remove all possible equipment for disposal or decontamination elsewhere.
- Flush a third time. Clean or shield local hot spots. (A "pinhole" camera using both light- and gamma ray-sensitive films was useful in finding hot spots.)
- Decontaminate concrete by removing surfaces by means of flame priming, chipping, sand-blasting and grinding, using a vacuum system to remove the dust. Seal in any remaining activity.
- Finish surface decontamination by scrubbing with cotton wipers using detergents and acids. (This transfer technique worked better than scrubbing with brushes and flushing.)
- Protect cleaned surfaces from recontamination by covering with paper.

During the course of the decontamination, serious airborne contamination occurred; this was corrected by improved ventilation, increased hygiene practice by the workers, and continuous cleaning and mopping of all working areas. Inhalation problems were solved by closer control of the use of respirators. Following decontamination the reactor was repaired and returned to service 14 months after the incident.[12]

1.5.2.2 SL-1 Reactor

For this operation, photographic, motion picture, pinhole-camera, and television pictures (many obtained remotely with a crane) were used extensively, together with the radiation measurements, to assess the reactor condition and plan the decontamination effort. Training with mockups was conducted, a control point established, equipment designed, and a new entry made to the fan room to permit better access. All operations were planned with the requirement of keeping water from entering the reactor. Manual and remote cleanup were used. Shielding of the reactor head was installed remotely, tanks were drained, and all loose items were removed. Next, vacuuming and sweeping were employed to reduce exposure rates to the level at which the reactor vessel could be removed to a hot cell for examination without incurring excessive worker doses. Equipment and wall sections were cut out and sent to a disposal area. Concrete support columns were removed using a bulldozer, and the steel reinforcing rods were cut. In slightly contaminated adjoining buildings, wiping, vacuuming and the use of surface-strippable films were sufficient for the decontamination.[14,15]

1.5.2.3 Surry

The Surry reactor steam generators, in Virginia, were removed in 1978 and 1979 because of multiple tube failures. Before carrying out the work, it was necessary to decontaminate the reactor building interior surfaces and the sump.

Initial radiation levels were 500 mR/hr on the interior building surfaces and 2 to 3 R/hr in the sump. The decontamination was carried out by hot-water scrubbing with household abrasive detergent. All decontamination solutions were collected and processed through the Surry plant waste treatment system.

At the end of two weeks the residual radiation levels were reduced to 2 to 3 mR/hr on the surfaces, and 500 mR/hr in the sump. These low levels allowed the programmed work on the steam generators and primary coolant piping to continue with a minimum exposure of personnel. The decontamination work resulted in a dose of 23 person-rem and involved 25 people. In addition, 400,000 pounds of lead was installed to provide temporary shielding in the heat exchanger cells to reduce exposure rates from 400 mR to about 40 mR/hr. Total cumulative dose required to instal l the shielding was 143 person-rem.[18]

1.5.3 Experience with Removal of Damaged Cores

While there is considerable experience with the removal of single failed fuel elements, experience is limited where significant core damage has been incurred.

1.5.3.1 Canadian NRX Reactor

After a considerable decontamination effort was completed in the reactor building, particularly in the area beneath the reactor, electrical and piping connections were removed in preparation for core removal.

All undamaged fuel rods were withdrawn from the reactor. Many others were fused together. An attempt to pull out the defective fuel rods was unsuccessful because of breakage. The tops of reactor concrete shields and top thermal shield were removed to provide greater access to the fuel rods. Three fuel rods were pulled using a cutting procedure. The third rod being removed in this fashion broke and allowed radioactive sludge to fall into the basement. The remainder of the fuel rods were grasped near the bottom and pulled out, with the work crew no longer trying to keep the lower portion of the calandria, or vessel, intact. All the fuel rods were pulled into a standard fuel rod carrying flask or fuel can and transported to a water trench leading to an extraction plant.

The next major operation was removal of the calandria. The tank was isolated from the cooling systems, rigged, lifted, and placed into a large canvas bag for containment. It then was placed on a skid and towed to a burial ground. There the calandria, with radiation levels measured to be about 700 R/hr, was covered with sand for shielding.

The whole calandria removal task was rehearsed using a nonradioactive "dummy" calandria. The trained removal crew was able to perform the difficult remote operation efficiently in about 30 minutes.[10,12]

1.5.3.2 Enrico Fermi I Reactor

On October 5, 1966, the Enrico Fermi I nuclear reactor, in Michigan, sustained partial meltdown of two fuel assemblies as the result of coolant-flow blockage. An estimated 10,000 Ci of fission products was released to the primary coolant and reactor cover gas. Special wedging tools of a chisel design were employed to break the bond between the two fused elements, and the elements were removed from the core for examination. Decontamination operations were successful and were climaxed by full-power operation on October 16, 1970.[19]

1.5.3.3 Canadian NRU Reactor

Following removal of the fuel rod fragments described in Section 1.5.1.2, a large cleanup effort was organized. An emergency headquarters was set up with CCTV and public address systems to the reactor hall, and an emergency change room was established. A vacuum system with a cyclone-type sand filter to prevent clogging was used to remove residual fuel debris. This operation reduced exposure rates to 1 R/hr per person and by utilizing 300 workers resulted in an average exposure

of 1.5 R per person. Wet mopping with detergent and wiping, with some additional vacuuming, provided the remainder of the cleanup.[13]

1.5.4 Experience with Decontamination of Reactor Coolant Systems

Chemical decontamination experience to remove fuel failure debris, including fuel fragments, is very limited. Only one reactor, the Plutonium Recycle Test Reactor (PRTR), at Hanford, Washington, has undergone such a decontamination.[9,20]

The PRTR reactor system was decontaminated in 1962 after the failure of an experimental fuel element. Radiation levels were up to 250 R/hr at low spots.

There has been little experience with removing fuel debris from large reactors; however, tests have been conducted on a number of loops and pilot plants.[21-27] Nitric acid was applied alternately with sodium hydroxide at the decontamination of the Homogeneous Reactor Experiment at Oak Ridge National Laboratory.[9] A variety of solvents have been applied to desorb and solubilize fission products from metal surfaces at the Idaho Chemical Processing Plant,[28] indicating that fuel removal technology is available for application at nuclear plants.

The removal of corrosion product oxide films and accumulated sludge has been a continuing concern of non-nuclear electrical generating plants and chemical process industries for many years. Mechanical and chemical cleaning processes have been used routinely, and systems are returned to active service after having been cleaned. These processes are generally applicable to cleaning nuclear plants. The principles and experience have been improved and applied in the nuclear power industry where similar problems exist plus the added complication of radioactive impurities incorporated into films and sludges. In many cases the removal of the radioactive impurities are the primary reason for the cleaning (decontamination) operation. The following case histories of decontamination and return to active service are pertinent to TMI-2.

1.5.4.1 PRTR

As a result of the reactor's experimental fuel element failure, the normal corrosion product oxide film absorbed a portion of the fission products. First, mechanical removal of the debris was accomplished by high-velocity flushing and filtration followed by draining and flushing dead-legs. Chemical removal of the oxides was then conducted using two applications of buffered oxalic-peroxide compounds. Finally, alkaline permanganate followed by inhibited oxalic acid was applied to remove residual fission product and corrosion product activities. The final exposure rates were near 5 mR/hr and the entire decontamination sequence took ten days. A second full-reactor decontamination was accomplished in three days using alkaline permanganate solution followed by a citric and oxalic acid mixture.[9,29-31]

1.5.4.2 Shippingport

The Shippingport, Pennsylvania, pressurized-water reactor (PWR) primary coolant system was chemically decontaminated in 1964 to remove corrosion product oxides and reduce radiation levels. A two-step, modified, low-concentration alkaline permanganate-citric acid treatment was used. The unique feature of this decontamination was the treatment of all chemical solutions, flushing water, and dilution water by ion exchange so that no liquid wastes were generated. Average decontamination factors of about 50 were achieved for the pipe walls and steam generators. As a result of the decontamination, modifications to the primary coolant piping were made with minimal radiation exposure to personnel. The Shippingport reactor was returned to active service and has operated to the present time.[9,32]

1.5.4.3 N Reactor

The N Reactor at Hanford, Washington, is a larger and more complex system than a commercial PWR. This system has been successfully decontaminated internally 11 times to remove the corrosion product oxides and reduce the radiation levels at the reactor face working areas. In each instance, an inhibited phosphoric acid was employed. Substantial reductions in radiation levels were achieved, and the decontamination procedure has become an annual routine that poses no serious problems to reactor operations.[33-35]

1.5.4.4 SGHWR Prototype

Both primary loops of the Steam Generating Heavy Water Reactor (SGHWR), Winfrith, England, were decontaminated in 1964 and 1971 using a citric and oxalic acid compound and since then have been decontaminated almost on an annual basis. In this case, a citric and oxalic acid compound was chosen because the primary concern was the removal of radioactive corrosion product oxides. In each case the total decontamination took six days. This included the chemical contacts, the initial preparations, and the rinses. More than 90 percent of the activity was removed in the first chemical contact. The decontamination factors of from 1.3 to 39 were achieved. The reactor system was returned to service with no subsequent problems.[36]

1.5.4.5 BR-3

A two-step decontamination process was carried out in October 1975 in the BR-3 reactor in Mol, Belgium, to reduce the radiation levels, facilitate steam generator inspection, and provide access for eventual repairs if required. The decontamination chemicals used were alkaline permanganate followed by a citric and oxalic acid mixture. A total of about 230 Ci of corrosion product activities was removed, and the steam generator radiation levels were reduced from an initial value of 10 R/hr to an average of 70 to 80 mR/hr.

1.5.4.6 Lingen Power Plant (KWL)

Several critical components of the primary coolant loop of the KWL reactor in the Federal Republic of Germany were removed and decontaminated in 1970. The decontamination solutions used were: (a) 0.25 percent oxalic acid + 0.25 percent citric acid, (b) 0.2 percent NaOH + 0.2 percent $KMnO_4$, followed by (c) 0.25 percent oxalic acid + 0.25 percent citric acid. Decontamination factors of 3 to 120 were reported, and most of the radioactivity removed was cobalt (Co)-60.[37]

1.5.4.7 SENA

The Societe d'energie Nucleaire Franco-Belge des Ardennes (SENA), Chooz, France, pressurized-water plant was decontaminated in 1968 to facilitate repairs, after 2400 equivalent full-power hours. Four steam generators and associated parts of their primary loops were decontaminated for the removal of radioactive corrosion products using a two-step process. The first step consisted of circulating alkaline permanganate solution. This was followed by a 6 percent solution of citric acid and oxalic acid. Decontamination factors ranging from 16 to 107 were achieved at the steam generator inlet and outlet.[38]

1.5.4.8 Douglas Point Reactor

A recent development in decontamination involves use of dilute solutions and regeneration of these solutions while in use with a strong acid cation exchanger in the H+ form. This process was used on the Douglas Point heavy-water-cooled PWR at Tiverton, Ontario, Canada. The decontamination solution used was a mixture of citric acid, oxalic acid and ethylenediaminetetraacetic acid (EDTA).[39,40] The decontamination factors realized ranged from 4 to 10, and it is estimated that 150 to 180 person-rem dosage was avoided during the maintenance work as a result of the decontamination.

More recently, the process has been used to decontaminate a portion of the reactor water cleanup system and let-down heat exchanger of the Vermont Yankee Reactor and Brunswick Reactor plants, located in Vermont and North Carolina, respectively.

1.5.5 Environmental Impact

1.5.5.1 Occupational Doses

The principal environmental impact of the decontamination operations discussed above has been the occupational doses incurred by the personnel conducting the decontaminations. For those decontaminations where significant quantities of radioactive materials escaped the reactor cooling system into the reactor building (i.e., NRU, NRX, and SL-1), the total occupational doses were 700, 250, and 998 person-rem, and involved 600, 200, and 463 people, respectively. The maximum total individual exposure to decontamination personnel was 19 rem.

1.5.5.2 Offsite Doses

During the subsequent decontamination efforts of even the more severe accidents reported in the preceding text, there were no measurable offsite doses. However, in some cases the offsite dose was not measured.

1.5.5.3 Nonradiological Effects

The most significant nonradiological environmental impact of the discussed decontaminations was the use of materials and energy to complete the work. In comparison to the facilities recovered and the occupational doses avoided, the use of the materials and energy was insignificant. Wastes generated were removed to existing radioactive waste disposal sites and the additional volume had insignificant environmental impact.

1.5.6 Comparison with TMI-2

Much of the experience on decontamination has been with systems smaller than TMI-2 in volume and surface area. However, gross contamination and radiation fields have been greater following the accidents at NRX, NRU, and SL-1 than expected at TMI-2 (based on analyses of samples of sump water and on the extent of surface contamination of small areas which were removed from the reactor building). From swipe samples taken during the reactor building entries to date, no particulate uranium or other fuel debris have been detected. The surfaces are contaminated with soluble fission products, primarily Cs-134, Cs-137, Sr-89, and Sr-90 which leached from the breached fuel into the water and were subsequently distributed throughout the containment. Past experience in decontaminating reactor buildings indicates that early removal of all loose objects, water washing, and scrubbing will significantly reduce this contamination. Exposure rates at the floor levels of the TMI-2 reactor building, excluding "hot spots", are estimated to range from 250 to 500 mR/hr. Decontamination of the Surry reactor containment in 1979 was performed at exposure rates of up to 500 mR/hr (see Sec. 1.5.2.3).

The first major difference at TMI-2 from past experience is that the reactor building at TMI-2 contained 44,000 Ci of Kr-85, which restricted access to the containment building for more than a year after the accident. At all the previous accidents, the workers began recovery operations immediately and got the job done quickly and effectively. With passage of time since the accident, there is concern that the surface contamination is becoming more difficult to remove. The second major difference is that 700,000 gallons of contaminated water remain at the bottom of the TMI-2 reactor building. The problem is further complicated by the large quantities of sodium in the TMI-2 water. Waste disposal requirements are much more restrictive and complex than for past experience.

Experience at other nuclear facilities in removing damaged fuel and core components has been rather limited, and much of the existing experience is not directly applicable to TMI-2. Perhaps the most applicable experience occurred at the NRX reactor in Canada. The NRX experience has shown that difficult defueling operations can be completed in a reasonable length of time with fairly low occupational radiation exposure. The important lesson from NRX is that detailed planning and the use of mockups for training can be very beneficial in reducing the occupational exposure.

Other plants have had fuel removed after severe damage; generally these incidents have involved only a single or a few fuel assemblies. For the most part these fuel assemblies have been constructed of stainless steel-clad uranium metal fuel. The TMI-2 fuel is quite different in that the fuel is uranium oxide pellets with zircaloy cladding, which is more susceptible to oxidation and embrittlement.

Decontamination of the reactor cooling system to remove residual fuel debris and fission products can be accomplished with well-established techniques. Based on PRTR experience, it probably will be necessary to remove the corrosion product films because the fission products may have been absorbed into them. Removing these films can be accomplished readily with existing solvents; in contrast, aged films are sometimes difficult to remove. For this special decontamination, it might be necessary to develop a special oxidizing solvent to condition the corrosion product films (replacing the conventional alkaline permanganate solvent) and at the same time dissolve

the residual uranium oxide fuel debris. Two major obstacles to the reactor coolant system decontamination are currently evident:

- If significant fuel cladding oxidation has occurred, insoluble Zirconium dioxide (ZrO_2) could be distributed throughout the reactor coolant system. Hydrofluoric acid, one of the few ZrO_2 solvents, is too corrosive to be used as a general decontaminant; therefore, the ZrO_2 will have to be removed by mechanical means.

- Commercial nuclear power plants are not designed with special considerations for large-scale decontamination operations.

1.6 REGULATORY REQUIREMENTS, OTHER CONSTRAINTS, AND FUTURE CRITERIA

The NRC is responsible for the regulation of TMI-2 cleanup operations to ensure protection of the health and safety of the public. In line with this responsibility, the staff has concluded that expeditious decontamination of TMI-2, including removal of the fuel from the accident-damaged reactor, and disposition of the radioactive wastes are necessary. For this reason, for all post-accident operations at TMI-2 the NRC has maintained the following regulatory objectives:

(a) Maintain reactor safety and reactor building integrity,

(b) Assure that environmental impacts are minimized, and that radiation exposures to workers, to the public, and to the environment are within regulatory limits and are as low as reasonably achievable (ALARA),

(c) Assure the safe defueling of the reactor and storage of the fuel, and

(d) Assure the safe collection, packaging, storage and/or disposal of radioactive wastes from the decontamination activities.

Implementing cleanup activities is the responsibility of the licensee. However, NRC's role in the decontamination operations may change if the licensee should go bankrupt or otherwise default on its obligation to decontaminate the TMI-2 facility. NRC's objectives in the cleanup operations would remain the same because of its mandate to protect public health, safety, and the environment.

In this section, current regulatory requirements, constraints, and criteria are discussed. Future criteria pertaining to the cleanup of TMI-2 are also proposed. These topics include radiological requirements (occupational exposure limits, radioactive effluent limitations, and radioactive waste disposal requirements) and nonradiological requirements (sewage and industrial wastewater discharge, and dredging permits).

1.6.1 Regulatory Requirements

Current regulatory requirements affecting the TMI-2 cleanup activities include the applicable provisions of 10 CFR Parts 20, 30, 40, 50, 70, and 71 of the Commission's Regulations; the terms and conditions of TMI-2 Operating License No. DPR-73; and the terms and conditions of several Commission orders and actions relevant to the postaccident stage of TMI-2 and reflecting the nonoperable condition of the facility.

1.6.1.1 10 CFR Part 20 and Part 50

10 CFR Part 20, Standards for Protection Against Radiation, establishes standards for permissible doses, levels, and concentrations of radiation, precautionary procedures, and waste disposal. Sections 20.101 through 20.104 relate to occupational exposure, and indicate the permissible doses of radiation and levels and concentrations of radioactive materials in air and water to which individual radiation workers may be exposed. Depending on conditions, these regulations allow maximum individual occupational doses of 3 rem per quarter to the whole body, head and trunk and active blood-forming organs, lens of eyes, or gonads; 18-3/4 rem per quarter to the hands and forearms or feet and ankles; and 7-1/2 rem per quarter to the skin of the whole body.

Limitations also are placed on the use of respiratory protection and the concentrations of radioactive materials in air and water in restricted areas where workers may be occupationally exposed. These concentrations for individual radionuclides are specified in Table I, Column 1 of Appendix B to 10 CFR Part 20. Additional restrictions on the concentrations of radioactive material

in effluents to unrestricted areas are contained in Table II of Appendix B to 10 CFR Part 20. Formulae and procedures for applying these concentrations to mixtures of radionuclides also are specified.

Sections 20.301 through 20.305 deal with disposal of solid, liquid, and airborne wastes. These sections establish standards and procedures governing disposal of wastes by (1) transfer to authorized recipients, (2) release to sanitary sewage systems, (3) burial in soil, and (4) other methods approved on a case-by-case basis. Specific approval is required for disposal at sea and incineration of wastes, as well as for other disposal activities that do not meet the generally prescribed criteria. Generally, wastes discharged to sewage systems must be soluble or dispersable in water and meet the concentration and quantity limits (e.g., 1 Ci/yr) specified in Section 20.303.

Appendix I to 10 CFR Part 50 provides numerical guides to meet the ALARA criterion of 10 CFR Part 20 for releases of radioactive materials in liquid and gaseous effluents to unrestricted areas. The numerical guidelines of Appendix I to 10 CFR Part 50 have been selected as the appropriate effluent discharge limits for the cleanup operations and are reflected in Appendix R.

1.6.1.2 10 CFR Parts 30, 40 and 70

Commercial land burial facilities presently are licensed on a case-by-case basis under 10 CFR Parts 30, 40, and 70, or equivalent regulations in the case of Agreement State licensing actions. Such facilities must be located on federal- or state-owned land under the provisions of 10 CFR Section 20.302(b).

NRC intends to allow the disposal of TMI-2 wastes at existing disposal sites provided that the risk to the public (both present and future generations) is similar to that presented by disposal of wastes routinely generated and disposed of at those disposal sites. Clearly, the existing disposal site requirements would need to be met.

At TMI-2, organic resin wastes will be generated. The upper levels of the bulk specific activity organic resin wastes routinely generated at other nuclear power plants have been on the order of 1 to 10 Ci/ft^3 of Cs-137 total activity. In this range of specific activities the resins will be exposed to approximately 10^8 rads after 300 years, or ten half-lives of Cs-137, assuming all of the energy is absorbed in the waste form. At exposures of 10^8 to 10^9 rads, organic resins could undergo significant decomposition, which could result in increased mobility of the collected radionuclides, gas evolution, and potential acid formation. Therefore, to ensure adequate radiation stability of the organic resin over ten half-lives, resins having bulk specific activities greater than 10 Ci/ft^3 will undergo specific evaluations to ensure that radionuclide mobility and other impacts within the waste container are minimized over the hazardous lifetime of the wastes.

Some of the TMI-2 wastes fail to fit established patterns for nuclear power plant wastes. Therefore, the proposed TMI waste forms need to be separately assessed by the NRC with respect to the objectives of the proposed 10 CFR Parts 60 and 61. The assessments of these nonroutinely generated wastes can best be performed on a case-by-case basis since there are many uncertainties regarding the characteristics of the TMI-2 wastes, and the processing and disposal options which are appropriate must be individually determined.

Until the uncertainties of the waste characteristics are resolved, non-routinely generated wastes would have to be retrievably stored. The waste container designs would be required to provide mechanical and structural stability over the intended storage period, during onsite and offsite handling, and throughout the transportation phase.

For wastes found to be unacceptable for routine disposal at the existing burial sites, more sophisticated disposal methods would be required. These methods might include additional container requirements, different waste processing modes, modifications to routine burial methods and procedures, or disposal in a high-level-waste repository.

The NRC staff intends to address the disposal criteria for nonroutine wastes on a case-by-case basis, giving consideration to the waste forms proposed by the licensee. This approach ensures a suitable basis on which to continue to protect the public health and safety and the environment and at the same time offers the licensee adequate flexibility in processing and package designs.

In addition to criteria placed on TMI-2 wastes based on the staff assessments, specific requirements based on Commission orders need to be complied with. As an example, the Commission has

ordered that EPICOR-II resins must be solidified prior to shipment offsite. This order is compatible with recently imposed disposal site license conditions at the Hanford, Washington, and Barnwell, South Carolina, burial sites requiring that after June 30, 1981, all resins having specific activities greater than 1 microcurie per milliliter (1 μCi/mL) (0.028 Ci/ft^3) for isotopes with half-lives greater than five years must be stabilized by solidification. The Barnwell site, however, will also consider disposal of dewatered resins in a high-integrity container. A high-integrity container is one which will maintain its mechanical and structural characteristics over the hazardous lifetime of the wastes.

1.6.1.3 10 CFR Part 71

NRC regulations on the packaging and transport of radioactive materials are contained in 10 CFR Part 71. This part also points out the applicability of regulations of the Department of Transportation (DOT) regarding radioactive material shipments. All shipments of radioactive materials--sources, samples, fuel, and wastes--will be made in accordance with this part and applicable DOT regulations (49 CFR Parts 170-189).

1.6.1.4 TMI-2 Operating License, Commission Statements and Orders Related to TMI-2 Cleanup

Several Commission Statements, Orders and amendments to the TMI-2 Facility Operating License (DPR-73) have been issued subsequent to the March 28, 1979, accident. The following list is a summary of these actions.

Date	Description
May 25, 1979	Statement - preparation of an Environmental Assessment for the operation of the EPICOR-2 system. Prohibited the discharge of accident-contaminated water.
July 20, 1979	Order for Modification of License - Authority to operate facility suspended, licensee to maintain facility in a shutdown condition in accordance with approved operating and contingency procedures.
October 16, 1980	Memorandum and Order-Commission approved operation of EPICOR-II system and provided its rationale for this decision.
October 18, 1979	Order for Modification of License - Authorized operation of EPICOR-II system, added technical specifications requiring monitoring of EPICOR-II discharge paths, required licensee to maintain tankage at TMI-1 for possible storage of waste water from TMI-2, and prohibited offsite shipment of spent resins until approved by the Director of NRR.
October 26, 1979	Clarifying Amendment to Order for Modification of License - Stated that a hearing is not necessary prior to operation of EPICOR-II.
November 21, 1979	Statement of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement - Stated that the Commission had decided to prepare a programmatic environmental impact statement on the decontamination and disposal of radioactive wastes resulting from the March 28, 1979, accident at Three Mile Island Unit 2.
February 11, 1980	Order - Issued new proposed Technical Specifications which superseded all pre-accident Appendix A Technical Specifications and Appendix B Technical Specifications 5.1, 5.2, and 5.3 which specified the licensee's pre-accident management organization.
March 12, 1981	License Amendment No. 10 - Formally amended license to add provisions of October 18, 1979, EPICOR-** Order for Modification of License.
April 7, 1980	Approved - Interim Criteria for Radiological Effluents from TMI-2 Data Gathering and Maintenance Operations.

Date	Description
June 12, 1980	Memorandum and Order - Commission approved decontamination of the reactor building atmosphere by controlled purging to the atmosphere and provided its rationale for this decision.
June 12, 1980	Order for Temporary Modification of License - For duration of reactor building purge, deleted certain sections of Appendix B Technical Specifications which limited gaseous effluent releases based upon Curies/second and imposed limits based upon exposure.
June 24, 1980	License Amendment No. 11 - Permitted bypassing of the interlocks from certain radiation monitors during purging of the reactor building.
August 11, 1980	Modification of Order - Deleted proposed Technical Specifications operability requirements (which has been imposed by the Order of February 11, 1980) for the balance of plant diesel generators and the 13.2 kv circuit from the Middletown Junction Substation.
September 26, 1980	Statement of Policy - Emphasized most strongly that all of our health, safety and environmental requirements applicable to Three Mile Island Unit 2 must be fully complied with by Met Ed. Also stated that in the event of any conflict, NRC health, safety and environmental requirements must supersede State agency requirements that result in a lesser degree of protection to the public.
November 14, 1980	Amendment of Order - Added proposed Technical Specification operability requirements for the Mini Decay Heat Removal System and deleted proposed technical Specification operability requirements for certain balance of plant equipment no longer required with addition of MDHRS.
January 29, 1981	License Amendment No. 12 - Revised license condition 2.E(3) so that tankage to store waste water would no longer be required to be reserved in TMI-1 but would rather be required to be reserved in TMI-2.

1.6.1.5 Order Regarding Proposed Technical Specifications

By Order of the Director, NRR, dated February 11, 1980, a new set of formal license requirements was imposed to reflect the postaccident condition of the facility and to ensure the continued maintenance of the current safe, stable, long-term cooling condition of the facility (45 F.R. 11282). The requirements:

(1) Define operating parameters for the current safe, stable, long-term cooling mode for the facility (defined as the recovery mode), and delete all other permissible operating modes so as to assure that operation of the facility in other than the stable shutdown condition of the recovery mode is precluded;

(2) Impose functional, operability, redundancy and surveillance requirements as well as safety limits and limiting conditions with regard to those structures, systems, equipment and components necessary to maintain the facility in the current safe, stable, shutdown condition and to cope with foreseeable off-normal conditions;

(3) Prohibit venting or purging or other treatment of the reactor building atmosphere, the discharge of water decontaminated by EPICOR-II system, and the treatment and disposal of high-level radioactively contaminated water in the reactor building, until each of these activities has been approved by the NRC,* consistent with the Commission's Statement of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement (44 F.R. 67738) (see Appendix B of this document).

*By Memorandum and Order, dated June 12, 1980, the Commission gave the approval contemplated by this restriction insofar as necessary for the licensee to conduct a purging of the TMI-2 reactor building in accordance with procedures approved by the NRC.

These requirements were set forth in a new set of proposed Technical Specifications contained in an attachment to the Order.

Section 6.8, Procedures, of these Technical Specifications requires that the licensee shall establish, implement and maintain written procedures covering the following activities:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Recovery Operations Plan implementation.

c. Surveillance and test activities of safety related equipment and radioactive waste management equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Radiation Protection Plan implementation.

g. Recovery Mode implementation. (Specifically Recovery Mode procedures which involve a reduction in the margin of safety, including those which:)

   1. Directly relate to core cooling.

   2. Could cause the magnitude of radiological releases to exceed limits established by the NRC.

   3. Could increase the likelihood of failures in systems important to nuclear safety and radioactive waste processing or storage.

   4. Alter the distribution or processing of significant quantities of stored radioactivity or radioactivity being released through known flow paths.

Pending approval by the NRC of written procedures required by a. through g. above, the licensee shall follow the previously approved procedures.

Each procedure above, and changes thereto, shall be reviewed by the Plant Operations Review Committee and approved by the Manager Site Operations prior to implementation and reviewed periodically as set forth in administrative procedures. Each procedure of b. and g. above, and changes thereto, shall be submitted to the NRC prior to implementation; these procedures, and changes thereto, shall also be subject to approval by the NRC prior to implementation. The Manager Site Operations shall have responsibility for determining which procedures are in the b. and g. category.

Temporary changes to procedures a., c., d., e. and f. above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the Plant Operations Review Committee and approved by the Manager Site Operations within 14 days of implementation.

Temporary changes to procedures of b. and g. above may be made provided the provisions above are satisfied and the change is submitted to the NRC for review within 72 hours following approval by the Manager Site Operations.

1.6.1.6 EPICOR II Waste Treatment System

On March 12, 1980, Amendment No. 10 to License No. DPR-73 was issued (page 1 of Ref. 41). This amendment revised the Technical Specifications contained in Appendix B to the license and established new requirements for radioactive waste treatment, storage, and shipment involving the EPICOR II waste treatment system.

This amendment also added the following license conditions:*

2.E.(2) The licensee shall promptly begin the process of decontaminating the intermediate-level waste water from TMI-2 by operating EPICOR-II. Prior to operation, the licensee shall consult the Director of (the Office of Nuclear Reactor Regulation) NRR for approval of the final operating procedures and design and construction details. In order to reduce the inherent risk from the contaminated water most expeditiously and prudently, the licensee should, to the extent possible, process all the water once through the EPICOR-II system.

2.E.(3) The licensee shall maintain suitable tankage at TMI-1 that could be used to store waste water from TMI-2 at an appropriate state of readiness, should additional storage become necessary.

2.E.(4) The licensee shall not ship spent resins offsite unless they have been solidified, and only then with the prior approval of the Director of NRR, provided however, that the licensee may ship nonsolidified but dewatered spent resins offsite if it determines, and the Director of NRR concurs, that such a shipment is required to assure continued operation of EPICOR-II or otherwise required to protect public health and safety. The licensee shall expeditiously construct a facility for solidification of the spent resins and shall use such facilities for resin solidification upon receiving the Director of NRR's concurrence with the design and operating procedures.

1.6.1.7 Interim Criteria for Radiological Effluents from Data Gathering and Maintenance Operations

On April 7, 1980, the NRC approved specific radiological effluent criteria for the purpose of data gathering and maintenance operations for TMI-2 during the period prior to issuance of the PEIS.[42] (Releases which are specifically not covered by these criteria are purging of the reactor building atmosphere, disposal of EPICOR-II water, and treatment and disposal of high-level radioactively contaminated water in the reactor building.)

In essence, these interim criteria established a modification of the 10 CFR Part 50, Appendix I, design criteria for specific TMI-2 activities as effluent limits. The criteria indicate what information the licensee must submit to the NRC for approval prior to performing these activities, and the type of review that the staff will perform in determining whether or not to approve each request. These criteria are as follows:

- The licensee must request approval from the NRC to perform data gathering and maintenance operations. In addition, separate procedures must be developed for each operation and submitted to the NRC for approval. These procedures must contain a description of the need for the operation, estimates of radioactivity that may be released, and estimates of onsite and offsite doses that may occur as a result of the operation. The procedures for each operation should be designed to conform to the existing NRC technical specifications as well as to the "As Low As Reasonably Achievable" (ALARA) concepts of 10 CFR Parts 20 and 50. The procedures developed by the licensee should not interfere with the applicability of other limitations, conditions, or agreements that the licensee may have regarding the releases of radioactive gaseous or liquid effluents with NRC, or with other federal, state, or local authorities.

- These procedures will be reviewed by the NRC to ensure that they meet the existing technical specifications, that the ALARA concepts of 10 CFR Part 20 and 10 CFR Part 50 are met, and that the existing Appendix I to 10 CFR Part 50 design objectives are conformed to, and that they conform to agreements to which the NRC is a party.

The Deputy Program Director, TMI Program Office, onsite now has the authority to permit weekly releases which result in offsite doses that are not greater than 5 percent of the annual Appendix I 10 CFR Part 50 design objectives normalized to a weekly rate (i.e., 0.05 times the annual design objective divided by 52). These permitted releases will allow the onsite TMI manager the flexibility to continue or authorize decontamination procedures while keeping

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*Page 2 of Reference 41.

releases at a small fraction of those evaluated in the FES of 1972 for Units 1 and 2 and the Supplement to the FES of 1976 for Unit 2.

The Director of the Office of Nuclear Reactor Regulation (NRR) now has the authority to permit weekly releases which result in offsite doses that are not greater than 50% of the annual Appendix I to 10 CFR Part 50 design objectives normalized to a weekly rate (i.e., 0.50 times the annual design objective divided by 52).

Releases which may result in offsite doses in excess of those described above require approval by the Commission.

Any impacts that will occur under the interim criteria are expected to be a small fraction of those described in the TMI Final Environmental Statement (FES) of 1972 and the Supplement to the FES in 1976. Nevertheless, recognizing the sensitivities of the local citizens to any activities at TMI, the NRC staff will inform the local governmental officials of any releases before allowing them to take place. The public will also be notified through appropriate press channels. This notification will be done in conjunction with established working relations with the Environmental Protection Agency. If faced with an emergency, the staff may find it necessary to make adjustments in these procedures and would use whatever means are available to keep the public informed.

1.6.2 Other Constraints

In addition to NRC regulatory requirements, there are other constraints that affect activities at TMI. These constraints include the Settlement Agreement with the City of Lancaster and various nonradiological constraints expressed in dredging permits and discharge permits issued by various Federal and Commonwealth of Pennsylvania agencies.

1.6.2.1 Settlement of the City of Lancaster Lawsuit

In the case of the City of Lancaster v. United States Nuclear Regulatory Commission, Civil Action No. 79-1368 before the U.S. District Court for the District of Columbia, a settlement was entered on February 27, 1980, between the parties to that action: City of Lancaster, City of Lancaster Authority, Albert B. Wohlsen, Jr., the United States Nuclear Regulatory Commission, and the Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company.* As a condition of the settlement agreement, the NRC and the utilities agreed, among other things, that "... no accident-generated wastewater will be discharged into the Susquehanna River from the date of this Settlement Agreement through December 31, 1981, or until the NRC completes its Programmatic Environmental Impact Statement .... or until the NRC completes such other environmental review referred to in its November 21, 1979, Statement .... regarding the discharge of accident-generated wastewater into the Susquehanna River, whichever is earlier."** However, nothing contained in the Settlement Agreement precludes either the NRC or the Director of the Office of Nuclear Reactor Regulation from authorizing whatever measures it or he deems necessary to cope with any emergency situation.†

The agreement provided specific definitions for "accident-generated water":**

(a) Water that existed in the TMI-2 auxiliary, fuel handling, and containment buildings including the primary system as of October 16, 1979, with the exception of water which as a result of decontamination operations becomes commingled with non-accident-generated water such that the commingled water has a tritium content of 0.025μ Ci/mL or less before processing;

(b) Water that has a total activity of greater than 1 μCi/mL prior to processing except where such water is originally non-accident water and becomes contaminated by use in cleanup;

(c) Water that contains greater than 0.025 μCi/mL of tritium before processing.

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*Settlement Agreement, p.1.

**Settlement Agreement, p. 2.

†Settlement Agreement, p. 3.

The agreement also established requirements for NRC notification of appropriate public authorities in the event a release occurred or was contemplated.

1.6.2.2 Dredging Permits

Two dredging permits presently are in effect for TMI, one issued by the U.S. Army Corps of Engineers (COE) (Permit NABOP-F/4 75-1048, with amendment) and one issued by the Pennsylvania Department of Environmental Resources (PDER) (Permit 2275724). The COE permit was issued on January 19, 1976, for maintenance dredging around the cooling water intakes for a period of ten years, expiring on December 31, 1986.[43] It was amended on March 15, 1976, to allow for relocation of spoil disposal sites on the island within the dike system surrounding the TMI facility.[44] The PDER dredging permit was issued on January 13, 1976, and expires on December 31, 1981 (Sec. 1, p. 8, item 10 of Ref. 45).

1.6.2.3 Discharge Permits

The following discharge permits regulating sewage and industrial wastewater are currently in effect for the TMI facility:

Pennsylvania Department of Environmental Resources Sewage Permit No. 2275419. This permit was issued on January 23, 1976, for the purpose of approving the construction of a sewage treatment facility and for the discharge of treated sewage to the Susquehanna River.[46] Soil and erosion control conditions were also made part of the permit. The permit was amended on August 10, 1977, to extend it to January 23, 1980.[47] This permit was again extended on August 23, 1979, to expire on January 23, 1982.[48] However, the TMI sewage plant currently is not in service.[49]

Industrial Waste Permits. Four industrial waste permits issued by the Pennsylvania Department of Environmental Resources currently are in effect for TMI (Sec. 1, p. 8, item 10 of Ref. 45): Permit 2270202 (issued June 8, 1973), Permit 2270204 (issued August 17, 1971), Permit 2270209 (issued February 15, 1977), and Permit 2277206 (issued February 22, 1978). These industrial waste permits cover thermal effluent criteria for discharge from mechanical draft cooling towers, treated industrial wastes, and other industrial wastewater to the Susquehanna River.

National Pollutant Discharge Elimination System (NPDES) Permits. The NPDES permit[50] issued by the USEPA for the TMI facility would have expired on January 30, 1980. However, a timely application for a new permit was made on August 15, 1979, to the Commonwealth of Pennsylvania, which now has NPDES permit authority.[51] The EPA permit will remain in effect while that application is still pending before the Pennsylvania Department of Environmental Resources. The NPDES permit is a comprehensive discharge permit covering a variety of discharge points from the TMI facility: surface water, sanitary, storm water transport system, surface impoundment, waste acceptance forms, evaporation, consumption, and other miscellaneous points. There are 27 discharge points indicated in the application for TMI (pp. 2-5, Ref. 45). Specific discharge limitations on heat, chemicals, oil and grease, and other pollutants are identified in the NPDES permit.

1.6.3 Future Criteria

Criteria affecting the TMI-2 cleanup proposed herein are expected to be developed over time as a result of issuance of this PEIS or otherwise. Future waste management and disposal criteria for low-level radioactive wastes and proposed radiological effluent limits for decontamination activities are presented. Decontamination criteria for removable radioactive contamination are also indicated. These decontamination criteria have been applied to the cleanup activities discussed in the body of this statement.

1.6.3.1 Future Radioactive Waste Disposal Criteria

Low-Level Waste

The staff currently is developing comprehensive regulations for low-level waste (LLW) disposal which will contain technical requirements and procedures for licensing LLW disposal facilities. A preliminary draft of these regulations, 10 CFR Part 61, was made available for public comment in the Federal Register (FR, Vol. 45, No. 41, February 28, 1980). An environmental impact statement (EIS) relative to 10 CFR Part 61 is being prepared.

Part 61 will establish the overall performance objectives to be achieved in the disposal of LLW, dealing with long-term protection of the groundwater, protection from an inadvertent intruder, and assurance of protection of public health and safety during site operations. Specific technical requirements will be established regarding (1) the form and content of waste that would be acceptable for disposal at a near-surface disposal facility, (2) the design, operation and closure of the facility, and (3) institutional controls on monitoring, surveillance and use of the site after closure. Procedures for the filing, content, and commission action on applications will also be established. The technical requirements on waste form and content, facility design and operations will address specific waste disposal concerns currently being considered in analyzing disposal of particular TMI wastes.

It is scheduled for 10 CFR Part 61 to be issued as a proposed rule during 1981 and as a final rule in 1982. In the interim, the staff would evaluate on a case-by-case basis the acceptability of wastes for disposal and applications for new disposal sites following the objectives and considerations set out in the draft of Part 61.

High-Level Waste

The NRC is vested with licensing and regulatory authority over certain U.S. Department of Energy (DOE) facilities by Sections 202(3) and (4) of the Energy Reorganization Act of 1974. These sections refer to: (1) facilities used primarily for the receipt and storage (including disposal) of high-level radioactive wastes (HLW) and (2) retrievable surface storage and other facilities authorized for the express purpose of long-term storage of HLW.

Geologic repositories would not be licensed as "production" or "utilization" facilities. Rather, they would be licensed under those provisions of the Atomic Energy Act dealing with receipt and possession of "byproduct" and "special nuclear" materials.

The NRC has the responsibility to evaluate all aspects of repository performance which could affect public health and safety. As a first step in carrying out its responsibility, the NRC staff is developing the regulations under which a geologic repository will be licensed. This regulation will be codified in 10 CFR Part 60 - Disposal of High-Level Radioactive Wastes in Geologic Repositories.

In 10 CFR Part 60, which is currently waiting approval before the Commission, high-level wastes are defined as:

(1) irradiated reactor fuel,

(2) liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel, and

(3) solids into which such liquid wastes have been converted.

The overall performance objective for the high-level waste repository (i.e., the maximum allowable release of radionuclides to the biosphere) will be established by the U.S. Environmental Protection Agency (EPA) in their Environmental Radiation Protection Standards. The NRC staff will implement these requirements in 10 CFR Part 60. The procedural part of 10 CFR Part 60 is scheduled to be noticed in the "Federal Register" as a final rule in early 1981. The technical part was noticed in the "Federal Register" as an advanced notice of proposed rulemaking in May 1980 and is scheduled to be noticed as a final rule in late 1981. The technical criteria of 10 CFR Part 60 will apply to all wastes emplaced in a geologic repository for high-level wastes and, therefore, would apply to TMI wastes disposed of in this manner.

Until very recently the designs of deep geologic repositories have placed almost total reliance for containment of the radionuclides on the site characteristics and surrounding geology. As a result, little attention has been given to waste forms, containers, and other engineered barriers to significantly contribute to the containment and isolation of the radionuclides for extensive periods of time. For example, credit was not normally given for the waste form and its associated packaging being able to contain the radionuclides for more than a few tens of years.

In order to compensate for the uncertainty in predicting the behavior of geologic systems over long periods of time, the technical criteria of 10 CFR Part 60 are based on a conservative

multi-barrier approach. In this approach, the repository consists of three major barriers: (1) the waste package, (2) the underground facility, and (3) the geologic setting. The NRC will establish minimum performance objectives for each of these major barriers.

The major performance objectives for the engineered system are (1) a waste package that is designed to contain wastes for at least 1000 years; (2) after package failure, the system is designed to limit the annual rate of release of radioactive material to less than one part in one hundred thousand of the maximum amount of each radionuclide calculated to be present in the underground facility at any time after 1000 years; and (3) the repository must be designed so that all the waste could be removed starting at any time up to 50 years after the termination of waste emplacement. In addition to these performance objectives, the NRC staff is developing guidance for the technical community regarding how to meet these performance objectives.

1.6.3.2 Proposed Criteria for Radiological Effluents from Decontamination Activities

The NRC staff proposes modifications of the TMI-2 technical specifications relating to radiological effluents resulting from the decontamination program. These modifications are discussed below:

- The TMI-2 Technical Specifications will be supplemented to require that radioactive effluents from the cleanup and decontamination operation of TMI-2 be limited so that the numerical design objectives of Appendix I to 10 CFR Part 50 will not be exceeded. Additionally, the licensee will be required to maintain radioactive effleunts as far below the Appendix I objectives as practicable. The licensee will have to maintain cumulative estimates of offsite population doses and maximum individual doses.

- Quarterly, the licensee must submit to NRC dose estimates based on all actual releases which occurred over the previous three months, including accidental and routine releases. Data and methods used to make these estimates should be provided or appropriately referenced. Actual environmental conditions should be used in making these estimates. Also, calculations as described above should be submitted to NRC within 30 days after each accidental release.

These proposed modifications to the technical specification are indicated in Appendix R. These modifications are proposed for the purpose of implementing the requirements of Appendix I to 10 CFR Part 50 and to assure that the offsite doses that may occur are as low as reasonably achievable, while at the same time, do not exceed the numerical design objectives of Appendix I. It is necessary to evaluate each release for conformance to the ALARA principles of 10 CFR Part 50, Appendix I, for two reasons. First, the numerical design objectives of 10 CFR Part 50, Appendix I, were designed for routine operation of a light water reactor and not for a cleanup and decontamination operation. They were based on the evaluation of years of reactor operational data. Since there is much less experience in large-scale decontamination activities there is little basis on which to establish numerical design objectives. Thus, it is necessary to evaluate each release for consistency with the "as low as reasonably achievable" (ALARA) principle. Second, doses that have occurred offsite during the calendar year must be taken into consideration. If accidents such as those described in the PEIS occur, doses offsite could be significantly larger than those described for routine decontamination operation, and could even exceed the numerical design objectives of 10 CFR Part 50, Appendix I, for normal operations of a light water reactor. If such a case occurs, the NRC may require a temporary termination of the cleanup operation or other action as deemed necessary. The basis used in approving latter programs will depend in part on the success of the earlier ones in achieving ALARA releases.

Requiring that the numerical design objectives of Appendix I to 10 CFR Part 50 are met will assure that the radiation dose received by the public during the cleanup operation is sufficient to protect the public and is equivalent to or below that of a normal operating reactor. These doses, even when added to the doses which occurred during the TMI-2 accident (maximum dose less than 100 mrem) are likely to have negligible health effects to individuals of the population. The background radiation in the area amounts to about 116 mrem per year, 36 percent of which comes from cosmic radiation, 39 percent from terrestrial radiation, and 24 percent from internal radiation (mainly K-40 deposited in the body). On the basis of comparison of the doses calculated here to those of natural background radiation, the health effects through the completion of the cleanup operation will be non-existent, especially in consideration of the fact that natural background radiation in the United States varies from one location to another within a range of about 70 to 310 mrem per year.

Adherence to the proposed modifications to the existing technical specifications will assure that the cleanup operation will proceed in a manner that provides a wide margin of public safety. The total doses offsite to the maximum exposed individual or to the population throughout the course of the cleanup operation should be a small fraction of that received due to background radiation. These requirements will also assure that any new releases are as low as reasonably achievable. The accident reporting requirements will provide the staff with the information necessary to be able to make an informed decision should it be necessary to terminate or modify an operation.

1.6.3.3 Surface Contamination Criteria

Guidance for acceptable surface contamination levels is provided in Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" (June 1971); and in Regulatory Guide 8.21, "Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants" (May 1978). Table 1 of Regulatory Guide 1.86 provides specific guidance for acceptable removable surface contamination levels for unrestricted areas. These levels are shown in Table 1.1.

Table 2 of Regulatory Guide 8.21 provides similar (but slightly less stringent) guidance for unrestricted areas.[52] These limits are considered compatible in level of safety with those for release of facilities and equipment for unrestricted use, as given in Regulatory Guide 1.86. In accordance with this guidance, the staff has developed this programmatic environmental impact statement using the levels indicated in Table 1.1 as those levels at which decontamination would be considered complete for unrestricted access or unrestricted release of decontaminated equipment or facilities.

Fixed contamination (e.g., radioactivity that is not detected by smear surveys for loose surface contamination) also must be limited. Consequently, in addition to Table 2 criteria, the average fixed contamination will be limited to less than 0.1 mrad/hr.

Table 1.1. Acceptable Removable Surface Contamination Levels[a] for Unrestricted Access

Radionuclide[b]	dpm/100 cm^2
Alpha radiation from U-natural, U-235, U-238, and associated decay products	1000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	20
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	200
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above	1000

[a]Reg. Guide 1.86, Table 1, p. 1.86-5.

[b]Where surface contamination by both alpha- and beta-gamma emitting nuclides exists, the limits established for alpha- and beta-gamma emitting nuclides should apply independently.

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2. MAJOR DECONTAMINATION AND WASTE DISPOSAL ALTERNATIVES

The alternatives addressed in this statement for dealing with the aftermath of the accident at TMI-2 are primarily concerned with decontamination, core removal, and waste disposal. The broad-scope, postaccident alternatives range from implementation of full cleanup to no action beyond maintaining the facility in safe shutdown condition, and may be grouped as follows:

1. Full cleanup, salvage and decontaminate usable equipment,
2. Full cleanup, remove equipment with minimal or no decontamination,
3. Partial cleanup with defueling,
4. Partial cleanup, fix core in place,
5. No action--maintain safe shutdown.

The impacts that would result from subsequent use of the site are beyond the primary scope of this statement; however, the staff recognizes that there may be a relation between the cleanup decision and decisions on future use of the site. Because an early decision could be made to decommission Unit 2, the staff has examined the impacts associated with various decommissioning alternatives. The results of this effort indicate that an early decision to decommission would not significantly affect the need for decontamination activities nor the resulting impacts discussed in this impact statement.

The possible options for future use of the site include (a) decommission the TMI-2 plant, (b) refurbish, requalify and use the existing nuclear power system, (c) use the TMI-2 buildings and site as permanent waste repositories, or (d) use the TMI-2 buildings and site as temporary waste repositories until a final decision is made. The decommissioning alternatives include (a) removal and dismantling (DECON), (b) safe storage with deferred decontamination (SAFSTOR), and (c) entombment (ENTOMB).[1-3] The decommissioning alternatives all depend upon prior cleanup and defueling and are discussed in Section 2.2.

To determine the effect that different future-use options might have on cleanup operations, it is important to consider separately the two operational phases identified in Section 1.3: Phase I, the decontamination phase that extends from shortly after the accident up to removal of the reactor pressure vessel head (RPVH); and Phase II, the defueling phase, which starts with removal of the RPVH and extends through removal of the fuel and decontamination of the reactor coolant system (RCS).

The critical factor in determining the effect of an early decision regarding future use of the site on cleanup operations is whether or not the fuel must be removed from the reactor pressure vessel and stored elsewhere. For reasons discussed below, the staff concludes that defueling and decontamination of the RCS are essential operations that should not be omitted regardless of the outcome of a decision on future use of the site.

A consequence of this position is that the distinction between the first and second alternatives listed above is unimportant for Phase I and Phase II operations. The large, permanently installed equipment items, such as the reactor pressure vessel, reactor coolant pumps, steam generators, and associated piping, would not be removed or refurbished until completion of these phases. If the future use of the site were known prior to decontamination of the reactor building this could affect the manner in which some smaller pieces of equipment are handled.* However, these operations constitute only a small part of the decontamination operations, and the difference in the

*For example, some items that were specifically designed for the TMI-2 plant, such as instrument panels, would have no salvage value if TMI-2 were decommissioned. These items might be removed immediately and discarded if the plant were decommissioned, whereas they might be decontaminated in place if the plant were refurbished.

impacts that might result from different alternatives for equipment decontamination/removal operations during Phases I and II are smaller than the uncertainty with which the impacts for all decontamination operations can be estimated. Much of the equipment will be needed during cleanup operations regardless of the decision on future use of the site (equipment for the building ventilation system is an example), and some equipment that was not too severely contaminated might be decontaminated for reuse elsewhere even if it were not used again at TMI-2 (electric motors are examples).

The third, fourth and fifth alternatives are very dependent on the options for future use of the site. None of these three options is compatible with refurbishment, and the last two are also incompatible with the decommissioning alternatives that are allowed by current federal guidelines and regulations.[1,2,3] However, for reasons given below and in Section 2.1, the staff concludes that none of these three alternatives is acceptable for TMI-2.

The alternatives that would involve permanent waste disposal onsite appear not to be technically feasible considering the location of TMI-2 (see Sec. 2.2). Such alternatives are also not compatible with current national policies and regulatory guidelines for radioactive waste disposal. For example, in his message to the Congress on February 12, 1980, President Carter noted that the present strategy for disposal of high-level waste and unreprocessed commercial spent fuel is focused on use of deep geologic repositories. A similar strategy applies to transuranic waste. The location of such repositories is to be based on detailed site-specific evaluations, and the location of TMI-2, on an island in the middle of a river, also raises the question about relying on long-term institutional controls beyond periods of 100 years to ensure the integrity of such a disposal site.

Three Mile Island has not been selected by the U.S. Government as a candidate site for evaluation of its suitability for location of a high-level waste repository. Therefore, unless major policy changes are made by appropriate governmental agencies, Three Mile Island would not qualify as a suitable site for permanent disposal of spent commercial fuel or other high-level waste. In any case, it is unlikely that the site could be qualified as a candidate high-level waste repository site because of such factors as nearby population densities and hydrology.

Similar considerations apply to permanent disposal of low-level radioactive waste. As in the case of high-level waste disposal, Three Mile Island has not been evaluated by federal and state authorities as a candidate regional site for permanent disposal of low-level waste. It is unlikely that the site could be qualified for this purpose because of the same factors mentioned previously for the case of high-level waste disposal.

These considerations do not preclude, however, interim storage of radioactive waste at the TMI-2 site. Since there currently are no waste repositories open for the disposal of high-level activity and transuranic wastes, it will be necessary to place these wastes in temporary storage until a permanent waste repository becomes available. Both onsite and offsite storage are considered for this temporary measure.

The alternatives available for full cleanup with usable equipment salvage, environmental impacts of each in terms of offsite releases radiation exposure to decontamination workers, and social and economic effects, are considered in detail in Sections 5 through 9. The decontamination and core disposal alternatives for partial cleanup are discussed below in Section 2.1. A preliminary analysis of the decommissioning alternatives is included in Section 2.2 in order to provide a clearer perspective of the relation between these alternatives and the alternatives for Phase I and Phase II cleanup operations.* A description of the classification schemes and disposal methods for radioactive wastes as they apply to the wastes generated by different cleanup options is given in Section 2.3.

2.1 DECONTAMINATION OF THE FACILITY

Cleanup of the facility includes decontamination of building and support facilities, removal of the core, and decontamination of the nuclear steam supply system. The operations involved may be described as follows:

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*It was not possible to provide even a preliminary analysis of the refurbishment alternatives because there is insufficient information available on the condition of the equipment. This information will have to be obtained during the Phase I and Phase II cleanup operations.

- Processing the contaminated water generated by the accident (and by decontamination operations) to remove radioactive materials
- Decontamination of surfaces inside the auxiliary and fuel handling buildings and the reactor building, and decontamination or disposal of equipment.
- Removal and disposal of internal parts and damaged fuel from the reactor vessel, and
- Cleanup of the reactor coolant system (including components such as steam generators and pressurizer).

For full cleanup, all cleanup operations would be carried through to the point that the facilities were ready to initiate decommissioning or refurbishment operations. For partial cleanup,one or more cleanup operations might be replaced by partial action or no action. Partial action in the core removal operation includes immobilization and storage of the reactor fuel within the reactor vessel (core fixation). Consideration of the partial cleanup options requires an examination of the impacts on other operations or modification or omission of an operation as well as impacts on the environment.

The primary criteria used by the NRC staff for judging the acceptability of partial cleanup alternatives are that the radioactivity left onsite and the resultant reactor core configuration should not pose a hazard of continuing radiation exposure to the public or to the workers after cessation of cleanup operations, and that all requirements for public health and safety should be met. Except as noted in Section 2.1.1, below, alternatives that imply use of the site for long-term storage of radioactive materials (beyond the normal operating lifetime of a power reactor, which is approximately 40 years) are not ruled out as unacceptable; however, they are given only limited treatment on the grounds that the scope of this document is restricted to alternatives that provide for eventual removal of all radioactive materials from the site, even if a decision to do so should be delayed beyond the cleanup period.

It is the staff's position that Three Mile Island should not become a permanent waste repository site. Hence, alternatives involving temporary onsite storage of waste that would greatly increase the effort required for subsequent offsite storage, processing, or disposal are regarded by the staff as unacceptable.

2.1.1 No Action--Maintain Safe Shutdown

For the purposes of this environmental statement, the terminology "no action" is to be understood as representing an alternative for which the reactor would be maintained in a safe shutdown condition, but no attempt would be made to decontaminate the TMI-2 facility. Under present regulations, licensing of the facility would be required and all equipment inside the reactor building would be operated as necessary but, because of the expected high radiation levels in the building, servicing of equipment inside would be strictly limited. The no-action alternative would, in effect, turn the reactor building into a long-term, limited-maintenance repository for nuclear waste--a function for which the facility was not designed. In time, some equipment and components within the reactor building could be expected to malfunction or fail as a result of limited maintenance. Several implications of implementing such a no-action alternative are discussed in Sections 2.1.1.1 and 2.1.1.2.

The NRC staff considers the no-action alternative unacceptable because the public health and safety cannot be adequately assured until the radioactive materials decay to innocuous levels. The times required for the predominant radionuclides to decay range from a minimum of about 300 years for Cs-137 and Sr-90 to thousands of years for the transuranic radioisotopes in the damaged fuel. In the staff's judgment, adequate institutional controls cannot be assured for these time periods, and thus the no-action alternative could pose a serious threat to future generations.

2.1.1.1 The Reactor

As of December 13, 1980, the decay heat level of the TMI-2 reactor was reported by the licensee to be about 70 kW. The primary coolant was being maintained at about 120°F and 94 psi, and natural recirculation was being used to remove decay heat from the reactor core. About half of the total decay heat was transmitted by convection to the reactor building, while the remaining decay heat was transferred to the cooling towers through steam generator "A". On January 5, 1981, the licensee stopped steaming the "A" steam generator by shutting a turbine bybass valve.

This put the RCS in a "loss-to-ambient" mode of cooling in which reactor decay heat is transferred from both RCS loops to the reactor building ambient environment.[4]

When the rate at which neutrons are generated from nuclear fission is just sufficient to sustain a nuclear chain reaction at a constant rate, a reactor is said to be "critical." A certain amount of neutron-absorbing material within movable control rod assemblies inside the core acts as a "brake" on the ability of the neutrons to sustain the chain reaction. When a rod is moved further into the core, more neutrons are absorbed, decreasing the rate of fission and resulting in a "subcritical" reactor state. Currently, several neutron-absorbing materials are used to control criticality, the most common being silver-indium-cadmium control rods and borated water (water containing boron, a neutron absorber, in the form of boric acid). Maintaining a sufficient amount of neutron-absorbing materials inside the TMI-2 core will keep it in its present subcritical condition.

It has been estimated by Met-Ed and independently analyzed by NRC[5,6] that the shutdown margin of the TMI-2 reactor is quite substantial (there are about 15 percent too few neutrons to sustain nuclear chain reaction at a constant rate), with the primary coolant containing about 3850 ppm of boron. Preliminary evidence indicates that some control rod material may have been relocated to regions of the reactor coolant system out of the core. This reduction in in-core control material can be compensated for by adjustments of the boron concentration in the primary coolant. Excess boron available in the primary coolant must be adequate to ensure subcriticality of the core at all times.

Of the two low-range neutron flux instruments, one is no longer operational. (The higher-range instruments are not sensitive enough to provide a reading at the current low neutron flux levels.) The criticality condition of the core is closely monitored by the remaining low-range instrument and by weekly analysis of the boron concentration in the primary coolant water. Since, under the no-action alternative, decontamination of radioactivity inside the reactor building would not take place, maintenance required by failure of neutron or boron monitors or other essential equipment would be carried out in relatively high contamination conditions (compared to post-decontamination) and radiation fields. Additionally, a limited preventive maintenance program necessitated by these higher radiation fields would increase the probability of equipment failure and the length of resulting equipment downtime.

2.1.1.2 Water

As a result of leakage from the primary system valves and pipe seals, the water level in the reactor building sump has been slowly, but continually, rising. Some instruments and electric cables have been affected by the water and are inoperative. As of January 7, 1981, the water depth was 8.05 ft above the basement floor (282.5-ft elevation) and increasing at the rate of about 0.04 ft/month due to leakage of about 3000 gallons/month from the primary system. Unless the water is transferred from inside the building to a different location, the plant status may deteriorate from the existing conditions as the water rises. The rate of leakage could increase with time as the system deteriorates.

2.1.2 Decontamination of Buildings and the Associated Support Facilities

2.1.2.1 Processing Contaminated Water

The contaminated water consists of water released during and after the accident in the reactor building and water released during the accident and generated by cleanup operations in the tanks and sump of the auxiliary and fuel handling buildings (AFHB).

On October 16, 1979, the Commission ordered the cleanup of the AFHB contaminated water to begin, using EPICOR II. The processing of contaminated water in the AFHB was essentially complete as of June 27, 1980. The processed water was being used at that time for washing and flushing contaminated components and piping in the AFHB. It also could provide a source of water for washing surfaces in the reactor building.

There were about 700,000 gallons of contaminated water in the bottom of the reactor building as of February 1981. This contaminated water (referred to as the sump water) overflowed the sump at the time of the accident and covered the entire floor of the reactor building, including the D-rings and reactor vessel cavity. Reasons for processing this water include:

- Transfer radioactivity from a mobile form in the water to a less mobile form with a reduced volume to facilitate disposal.

- Eliminate the potential of leakage of radioactivity from the reactor building sump to the environment.

- Reduce the inventory of sump water to levels so that equipment needed to maintain safe shutdown will not be adversely affected by rising water levels. Provide surge capacity for collection and temporary storage of water generated by leakage from the primary system or from decontamination activities involving building and equipment surfaces.

- Reduce radiation exposure to workers on cleanup of building and equipment surfaces and defueling.

- Alleviate potential psychological stress to the local population associated with lack of cleanup progress.

A criterion used by the NRC staff for establishing the acceptability of decontamination alternatives is that no unprocessed accident water should remain after cleanup activities have been terminated or suspended for an indefinite period of time.

Alternatives for managing the reactor building sump water that could be used for the full-cleanup option all involve processing the water to remove dissolved radionuclide ions, and are discussed in Section 7. Alternatives that could be used only for the partial-cleanup options allow other approaches that include: (1) leaving the sump water in place in liquid form, and (2) immobilizing the sump water in solid form by chemical combination with cement or absorption in solids. These alternatives are briefly discussed below.

The alternative of leaving the sump water in place in liquid form is not acceptable to the NRC staff because it leaves unprocessed accident generated water which may eventually leak from the reactor building. This water also results in high ambient radiation levels that would make cleanup difficult in the vicinity of the sump. Details of the contamination levels (concentrations of dissolved radionuclides) and exposure levels may be found in Section 7.

Immobilization processes using cement are complicated by the geometry and equipment on the sump floor. It would be almost impossible to introduce a portland cement slurry into some of the inaccessible spaces and to obtain adequate mixing with the water. Even if attempted, it would be impossible to assure that complete solidification had been achieved. Therefore, some water would probably remain and would continue to pose a leakage hazard to the public. If complete solidification could be achieved and assured, the resultant product would require extended (e.g., greater than 100 years) isolation from the environment, which could not be assured by existing institutional controls. The use of cement to solidify the sump water has the further disadvantage that it eliminates the sump as a useful means of removing decontamination liquids from the reactor building.

Because of the difficulty in removing the solidified material and the subsequent high occupational exposures, the implementation of this option would probably result in leaving the solidified material in place. Leaving the solidified material in place then implies that the TMI site will become a waste disposal facility.

On the basis of the reasons given above, the staff considers the use of cement to solidify the sump water in place to be an unacceptable alternative.

There are a number of solids commercially available for immobilizing the sump water in the form of a wet solid; however, for all the absorptive media known and considered by the staff, the volume of the wet solid would be larger than the volume of the sump water by a factor of at least three; for some the factor would be greater than 50. A cap of concrete would be needed in order to provide shielding protection. Subsequent removal would not be as difficult as for cement; however, it would still be much more difficult than pumping out the sump water, and the volume of the reactor building in the sump area within which the equipment was entombed would be about twice as large as for cement. The lower part of the steam generators and part of the steam generator piping would be covered, and defueling and cleanup of the reactor cooling system would be severely complicated. The problems of achieving complete filling due to complicated geometry, noted above for the cement alternative, occur also for the absorptive media alternative, and make likely the occurrence of pockets of contaminated water.

In view of the potential problems associated with absorptive media immobilization of the sump water and the lack of any significant advantages, the staff regards inplace immobilization of unprocessed sump water to be an unacceptable alternative.

2.1.2.2 Building Decontamination

The auxiliary and fuel handling buildings (AFHB) and the reactor building were contaminated during the accident. The AFHB contain tanks and equipment for preparing, storing and transferring water used as the primary system coolant and facilities and equipment for refueling the reactor. The reactor building contains the reactor, the reactor cooling system and the nuclear steam supply system.

The tanks and equipment in the auxiliary building are needed for maintaining the reactor in safe shutdown and for processing the contaminated water. The equipment in the fuel-handling building would be needed for defueling the reactor. The licensee, with the concurrence of the NRC, started work on decontamination of the AFHB in April 1979 in order to remove hazards from possible releases of radioactive contamination and to maintain the reactor in safe shutdown condition. As of January 1981, the work on decontaminating the AFHB was about two-thirds complete. The methods that have been considered and used for the decontamination are discussed in Section 5.

The considerations that led to the decision to begin decontamination of the AFHB are still valid. In the NRC staff's judgment, any alternative provisions for ensuring protection of the public from radioactivity dispersed in the AFHB would involve at least as much risk to workers and public as would continuation of current decontamination of the AFHB until it is completed as planned. Hence, alternatives for AFHB cleanup, other than the full cleanup alternatives examined in Section 5, will not be given further consideration.

The partial cleanup alternatives for decontamination of the exposed interior surfaces in the reactor building may be divided into three categories: (1) partial decontamination (decontamination of all surfaces, but leaving a higher level of contamination on completion than with full cleanup), (2) selective decontamination (decontaminating selected work areas, isolating the remaining areas), and (3) destructive decontamination (using methods without regard to the damage they may do to surfaces or equipment). Alternatives for these categories are considered individually below.

Partial Decontamination

The advantage of stopping decontamination efforts short of full decontamination is that the exposure to radiation of workers doing the decontamination and defueling would be decreased. The primary disadvantage is that the exposure of workers who subsequently would be engaged in disassembling and defueling the reactor and cleaning up the primary system would remain high (see Sec. 6).

A point of diminishing returns is eventually reached at which further reduction in the radiation fields (during defueling and primary system decontamination) is not sufficient to justify the additional decontamination effort. There are not sufficient data on TMI-2 to enable an accurate determination of this crossover point. However, on the basis of the estimates of the work efforts and worker exposures given in Section 5, the staff estimates that this point will be very close to the stage of full decontamination. By this stage the radiation fields would be relatively low so that the net increase or decrease in overall exposure from any additional decontamination work effort would be relatively small.

On the basis of the expected overall reduction in worker exposure, and the fact that partial decontamination would merely defer the cleanup problem at the expense of a continuing hazard of releases to the environment, the staff considers the partial overall decontamination alternatives to be unacceptable if the reactor is to be dissassembled and defueled and the primary system is to be cleaned up.

If the alternative of core fixation rather than defueling were chosen, and if it were decided not to clean up the primary system (steam generators, pressurizer and associated piping and equipment), then partial decontamination might be a more attractive alternative. This alternative has not been explored because examination of the core fixation alternatives by the staff has shown them to be unacceptable (see Sec. 2.1.3.1).

Selective Decontamination

One alternative for this category would involve the decontamination of a corridor to the reactor vessel or the construction of a shielded corridor to the reactor vessel, without upper-level decontamination. For this alternative, it is presumed that access to a local area around the reactor would suffice for reactor defueling. However, the confined space of the restricted areas provided by localized decontamination or a shielded corridor does not allow for the use of the overhead crane that is required for the reactor head lift nor does it take into account the areas needed for equipment laydown, the shield slab laydown or movement of the fuel-handling machines. The corridor concept is not practical since a much greater volume of the containment building is needed for defueling and therefore will not be considered further.

Another alternative in the selective decontamination alternative would be to cover contaminated surfaces with a high-density fixitive. For this alternative it is presumed that covering a surface with a high-density fixitive would reduce the dose rate by an appreciable amount. However, based on estimates of existing dose rates from gamma radiation, a one- or two-inch-thick coating with Gunnite, a typical fixitive, would reduce dose rates very little. The staff estimates that the worker-dose cost of applying such a coating to all of the interior surfaces would be comparable to the worker-dose cost of reducing the surface decontamination to ALARA levels; hence, the staff does not consider the use of high-density fixitives to be an acceptable alternative.

The preceding considerations indicate that the only acceptable variants of selective decontamination, if there be any, would be limited to isolating and shielding selected contaminated areas that were not needed for reactor defueling and primary system decontamination operations. The staff considers it to be unlikely that the gain from bypassing a limited number of selected locations would offset the impact of isolating and shielding them, and the extra effort in keeping track of and avoiding contaminated pockets. Hence, no further consideration will be given to any selective decontamination alternatives.

Destructive Decontamination

Alternatives in this category are of two kinds: (1) use of corrosive chemicals for surface decontamination without regard to damage to the surface, and (2) removal of equipment for burial offsite without decontamination (scrapping). These alternatives are not exclusive; both may be used to varying extents.

An advantage of using corrosive chemicals for cleaning is that they remove part of the surface so that radionuclides that are strongly adsorbed or chemically bonded to the surface also will be removed. The disadvantages are that there is an increased health and safety hazard for workers; the spent liquid must be collected by means of special facilities that isolate it from the sump, or special measures must be taken to ensure that the processing equipment for removing dissolved radionuclides from the sump water is not adversely affected; special measures must be taken to protect equipment that may be needed during the defueling operations; and difficulties may be encountered in waste treatment processes. Experience gained in decontaminating the auxiliary and fuel handling buildings (Sec. 5.1.1) and information gained on the condition of the reactor building interior surfaces and environment (Sec. 5.2.1) have shown that the use of corrosive chemicals will probably be unnecessary. The staff considers the balance between the advantages and disadvantages to be on the disadvantageous side and will not, therefore, give further consideration to the use of very corrosive chemicals. Less harsh chemical methods are considered in connection with the full cleanup alternatives in Sections 5 and 6.

The removal of equipment without decontamination (scrapping) would, under the assumption that defueling of the reactor would take place, be limited to those pieces of equipment that were not needed during decontamination or defueling. Most of the building services, such as the electrical system, ventilating and cooling systems, and polar crane, would be needed during the defueling operations. Equipment that was specific to the operation of the power plant, such as steam generators, pressurizer, and associated pumps and piping are not in this category; however, these items would not be removed prior to defueling because defueling operations require continued cooling and criticality control provided by the primary system. Thus, defueling must be completed before the primary system can be safely dismantled.

The first equipment for which the "scrap or refurbish" issue would be likely to arise would be salvageable equipment removed in the process of removing the reactor pressure vessel head and upper internals. By that time, progress should be close to a point where there would probably be

enough information on the state of the reactor to consider the issue of whether to decommission or rebuild, which is beyond the scope of this document. There does not, therefore, appear to be any need to give further consideration to the alternative of scrapping all unneeded equipment.

2.1.3 Reactor Core Removal and Decontamination of the Nuclear Steam Supply System

Cleanup of the reactor and the nuclear steam supply system (NSSS) comprises the major stages for which the prior stages of building and water decontamination would merely be preparatory. The full-cleanup operations for these major stages involve opening and inspecting the reactor pressure vessel (RPV), which contains the fuel and control rod mechanisms; removing the mechanical parts and fuel from the RPV; draining and cleaning the piping and equipment on the primary side of the NSSS (i.e., the piping and equipment through which water from the RPV circulates, referred to as the reactor coolant system (RCS)); and processing the water that is now in the RPV and RCS in order to remove dissolved and suspended solid radionuclides. The full-cleanup alternatives for these operations are considered in Section 6.

Taking no action on any of the preceding operations is considered unacceptable by the staff for reasons discussed in Section 2.1.1 and will not, therefore, be given any further consideration.

The partial-cleanup alternatives considered in this section for the reactor and NSSS are of two kinds: alternatives that might be considered if an early decision were made to discard all of the equipment, and alternatives that might be considered if the decision to rebuild or decommission were deferred indefinitely and it was decided to place the entire plant in safe storage for an indefinite period with a minimum immediate expenditure of effort.

The primary public health and safety criteria used by the staff for assessing the acceptability of partial-cleanup alternatives are: (1) there should be essentially no risk of occurrence of criticality, (2) there should be essentially no risk of release of radioactivity in excess of that which is allowed from an operating reactor, and (3) the core should remain in a physically stable configuration with adequate means for removing heat generated by decay products. The staff dismissed from consideration at this time, for reasons noted in the introduction to Section 2, those core-fixation methods that would imply use of the site as a waste repository by greatly increasing the effort that would be required for subsequent completion of cleanup operations.

The partial-cleanup alternatives for the different operations are not independent. Since there would be little justification for undertaking full action for one operation and partial action for another, the staff will consider only certain combinations of partial cleanup alternatives that can be examined independently for each operation.

2.1.3.1 Core Fixation

It is unlikely that any of the fuel or internal parts in the RPV are salvageable for use in their present condition. Hence, the only alternative to defueling that would be opened up by an early decision not to rebuild would be that of immobilizing the core and removing the entire RPV, with its contents, and transporting it to some other site for ultimate disposal. In view of the size (32 feet high and 14 feet in diameter) and weight of the RPV and its contents (greater than 1500 tons), the high radiation levels, the weight of shielding required for transportation and the danger of spillage, the staff considers this option to be unfeasible and unacceptable.

The partial-cleanup alternatives to defueling that have been considered by the staff also involve core fixation. For these alternatives, the fuel would be immobilized by means intended to prevent occurrence of criticality and then sealed inside the RPV by means intended to prevent leakage of radioactivity to the environment.

The problems that must be dealt with in core fixation are: (1) ensuring that the neutron-absorbing material is uniformly distributed and the fuel is immobilized in a permanently stable configuration so that criticality cannot occur, and (2) ensuring that the vessel remains completely sealed.

The alternatives considered by the staff, together with brief assessments of their feasibility, are as follows.

Fill the Reactor Pressure Vessel with Concentrated Boric Acid Solution

In order to prevent loss of the boric acid, which is essential for preventing criticality, it would be necessary to seal off the openings into the RPV. This would be difficult to do without draining the RPV first. The radiation levels and worker exposure would then be very high and it is unlikely that the seals could be made secure enough to ensure that no leaks would occur. Lack of knowledge of the reactor vessel integrity increases this uncertainty. In addition, the elimination of an external recirculating heat removal system would increase the average core temperature (i.e., heat removal would be effected by conduction only). Therefore, since the long-term physical stability of the core could not be ensured, the staff does not regard this alternative as an acceptable one.

Fill the Reactor Vessel with Solid Neutron Absorber Additives in Pellet or Particulate Form

If this alternative were implemented, there would be no way to ensure that the pellets or grains were uniformly distributed throughout the core, especially in view of the lack of knowledge of the damaged core configuration. There may be regions of the core which have been fused or compacted together in such a manner that the pellets or grains could not be properly distributed throughout the core, and there would be no reliable way of being sure that there were no voids. Under these conditions, it would not be possible to ensure that the core would not go critical. To avoid the necessity of providing watertight seals and to avoid the risk of pressure buildup or leakage of radioactivity, it would be necessary to drain the reactor vessel after the pellets or grains had been distributed. The dry core would have poor heat conductivity, and high local temperatures could occur that would make the core physically unstable.* The water provides shielding from the radiation emitted by the core; hence radiation fields outside the pressure vessel would be much higher. For these reasons, the staff does not consider this alternative to be sufficiently safe and reliable to be acceptable.

Fill the Reactor Vessel with Cement Containing a Neutron-Absorbing Material

Distributing a cement slurry throughout the core, without voids, to ensure the physical stability needed to prevent criticality would be extremely difficult. High local temperatures that might cause the cement to pulverize could occur. Removing the damaged core when it was immersed only in water would be difficult enough. Removing the damaged core at some future time after it had been encased in cement would be far more difficult. Removing the fuel from a solid block of cement, 14 ft in diameter and 20 ft high, filled with fuel elements and fuel rods would require the use of saws and air hammers that generate a great deal of debris, and would likely generate much higher occupational radiation doses than removing the fuel from water. The staff does not, therefore, consider immobilization in cement to be an acceptable alternative.

Partial Core Removal and Dry Layup

If a sufficiently large fraction of the fuel were removed from the core, enough to ensure that the amount remaining was below the threshold for which criticality could occur, then if the reactor vessel were drained the remaining fuel could be left in place without danger of coolant leakage to the environment. However, integrity of all seals to the reactor vessel could not be assured for long periods of time, and escape of some radionuclides would be likely. Another disadvantage of this alternative is that with the water removed, the radiation levels outside the pressure vessel would become very high. However, the primary reasons that the staff does not regard this as an acceptable alternative is that after the work effort and worker exposure for removing most of the fuel had been incurred, there would be no advantage gained by not completing the task and removing all of the fuel and thus removing a potential hazard for the future.

On the basis of the preceding considerations, the staff finds core fixation to be an unacceptable option that does not merit further consideration.

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*The TMI-2 core would not go critical, regardless of the configuration, if it were perfectly dry and no other neutron moderator were introduced. Precautions would be needed to ensure that there is sufficient boron in the water to prevent recriticality before it was fully drained, and to ensure that the core could not go critical from inleakage of unborated water.

2.1.3.2 Reactor Coolant System Treatment

Full decontamination of the reactor coolant system (RCS) would involve draining the contaminated water; removing the fuel-failure debris from the system by flushing, chemical treatment, or other means; and flushing with appropriate solutions in order to remove any remaining contamination adhering to inside surfaces. Decontamination of the RCS is treated in Section 6.

An early decision to decommission the plant would open up the alternative of dismantling and scrapping all of the RCS equipment. Even if the decommissioning alternative were selected, it would be necessary to drain the system and remove any fuel debris that might spill during dismantling and removal. In addition, an analysis would be needed to determine whether additional decontamination, beyond that needed to avoid spillage, would decrease the overall occupational exposure. The exposure to decontamination workers could be less than the exposure that workers would receive while dismantling the equipment and packaging it for transport without further decontamination. There is not sufficient information on the condition of the RCS to carry out such an analysis at this time, and it extends into a post-decision period that is beyond the scope of this document; hence, the analysis of this alternative will not be carried through in this statement. By the time the cleanup work has reached the point where decontamination of the RCS can begin, the information needed to make a decision between rebuilding or decommissioning should be available.

The no-action option for cleanup of the RCS would be reasonable only in the context of a complete no-action plan. This option is unacceptable for reasons presented in Section 2.1.1; hence, it will not be given further consideration.

Of the partial-cleanup alternatives that might be considered, any that would leave contaminated water in the system can be ruled out as unacceptable for the same reasons that were given in discussing treatment of the sump water (Sec. 2.1.2.1) and core fixation (Sec. 2.1.3.1). The acceptable alternatives for the partial-cleanup options would, therefore, involve draining the contaminated water from the steam generator and piping and removing the water from dead legs (i.e., sections of pipe that are connected to the system in such a manner that they do not drain when the system is drained). This leaves a range of alternatives that differ only in the amount of contamination removed. Debris in the form of sludge would dry and cake with the water removed and become more difficult to remove later.

Insofar as the staff has been able to determine, there are no advantages to be gained by partial cleanup of the RCS if the fuel and internal mechanisms must be removed from the RPV. In view of this circumstance and the above-noted hazards associated with partial RCS cleanup, the staff considers the partial-cleanup alternatives for the RCS to be unacceptable.

2.1.3.3 RCS Water Decontamination

The only alternatives that have been found to be acceptable for defueling the reactor and decontaminating the reactor coolant system require draining of the system. The only alternatives open for managing the contaminated water are to store it without processing, or to process it to remove the solid radionuclides. The first alternative is considered unacceptable by the staff for the reasons given in Section 2.1.2.1. The second alternative is examined in detail in Section 7.1.

2.2 DECOMMISSIONING

The purpose of this section is to: (1) review briefly the steps in the cleanup process that precede the alternatives of restarting or decommissioning in order to determine whether an early decision to decommission has any impact on the nature or sequence of those steps, and (2) evaluate the impacts on the environment that would result from the decommissioning of TMI-2 sometime in the near future. Under normal circumstances, decommissioning follows the orderly shutdown of a facility at the end of its planned operating life. The situation at TMI-2 is significantly different from normal, with the reactor building and the AFHB severely contaminated, and much of the fuel core damaged. In addition, a major cleanup effort is currently underway.

Figure 2.1 is a simplified decision point diagram which illustrates the decision flow and activities which would be involved in arriving at either a refurbished facility, which could be restarted, or a decommissioned facility, which would eventually result in a site without any significant residual radioactivity from the previous existence of TMI-2. Several decommissioning alternatives, evaluated in this respect, are also indicated.

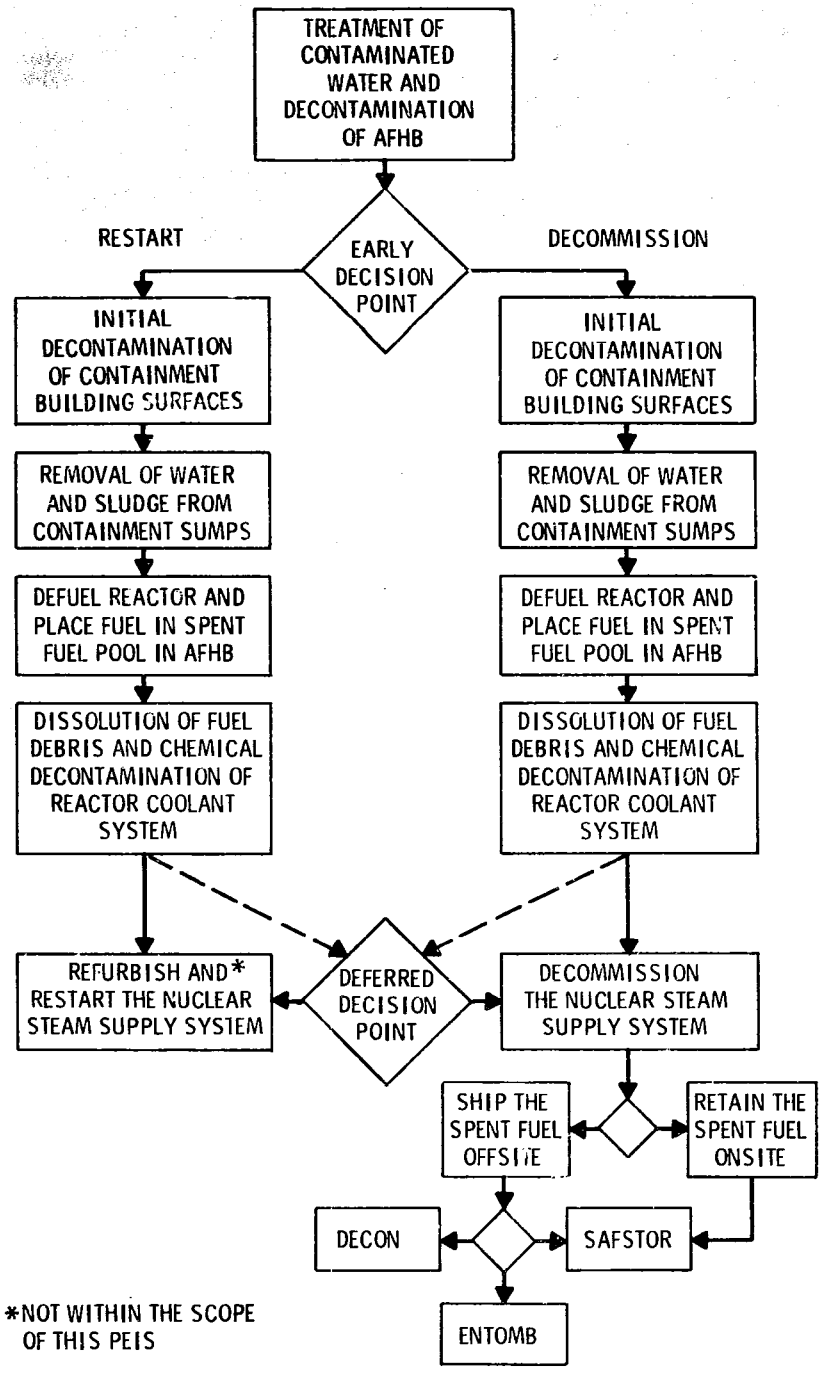


Figure 2.1. Simplified Decision-Point Diagram.

It should be emphasized at this point that neither the decision to refurbish or decommission TMI-2, nor the evaluation of the full environmental impacts of these activities, per se, are within the scope of the PEIS. However, because of the question of whether the consequences of an early decision to decommission affect alternatives in the cleanup of TMI-2, the impacts of such an early decision to decommission have been evaluated and sufficient material on decommissioning in general has been provided in this section and in Appendix U to make the discussion understandable. The cleanup processes and the rationales for performing the various steps are presented in Section 2.2.1. The alternative approaches to decommissioning are discussed in Section 2.2.2, with the impacts of these alternative approaches given in successive sections: waste volumes generated (Sec. 2.2.3), effluents and releases to the environment (Sec. 2.2.4), environmental impacts (Sec. 2.2.5), and decommissioning costs (Sec. 2.2.6). Details are given in Appendix U.

2.2.1 Rationale of Initial Cleanup Activities

As discussed in Sections 1.1 and 2.1, the initial cleanup campaign at TMI-2 has two principal goals: (1) to safely defuel the reactor, placing the fuel in a storage configuration that is safe from nuclear criticality and/or fuel meltdown, thus precluding the possibility of an inadvertent uncontrolled nuclear reaction occurring in the fuel, and (2) to collect and package for disposal the large quantities of water-soluble and otherwise readily dispersible radioactivity present in the plant and to ultimately remove the radioactivity from the site, thus precluding the possibility of its inadvertent release to the environment. For this reason it is postulated that prior to initiating any decommissioning activities, the reactor building sump water has been processed and the sludge has been removed, the wash down of the reactor building and the installation of temporary shielding has resulted in general-area radiation exposure rates on the operating floor (347-ft level) in the 5-10 mR/hr range, and in the basement of the reactor building in the 30 mR/hr range, and building surfaces have smearable contamination levels in the 3000-4000 dpm/100 cm^2 range, exclusive of hot spots.

The irradiated fuel elements and debris have been removed and are assumed to be stored in the spent fuel pool in the AFHB. Shipment of the irradiated fuel to an away-from-reactor fuel storage facility (AFR), reprocessing plant, or some other disposal facility is assumed to begin when decommissioning begins and to continue until all irradiated fuel has been removed from TMI-2. However, lack of a suitable facility to receive the fuel may result in its retention in the spent fuel pool in the AFHB for an extended period of time, in effect, converting that portion of TMI-2 into an AFR.

In terms of the schedule shown in Figure 1.4, active decommissioning efforts would begin at the conclusion of the fuel debris dissolution and the chemical decontamination of the reactor coolant system and associated systems. As a practical matter, the earlier cleanup efforts contribute to the total decommissioning effort, but for convenience in this analysis, decommissioning is necessarily treated separately from the initial cleanup. To obtain estimates of the total impact, the impacts from the initial cleanup (to allow safe defueling) should be added to the impacts from decommissioning.

Summaries of waste volumes generated during decommissioning operations, effluents and releases, doses, and costs are provided for each decommissioning alternative in this section. These values are separate from the values provided in other sections of the PEIS for the cleanup and decontamination steps.

2.2.1.1 Building Decontamination

To accomplish the first goal, the removal of the irradiated fuel from the reactor vessel, it is necessary to reduce the amount of penetrating radiation in those work areas pertinent to the defueling operations to levels that will permit reasonable occupancy times for the workers without undue radiation exposure. In keeping with ALARA considerations, decontamination and shielding work in the containment building prior to the defueling operations will be limited to those activities that produce the greatest reductions in area radiation dose rates per unit radiation dose absorbed by the workers doing the decontamination and shielding (first phase decontamination, as discussed in Section 5.2). It is anticipated that these activities will include washdown of the containment building surfaces using water jet equipment, draining and flushing of the containment building sumps, and installation of temporary shielding around localized hot spots to reduce general area radiation dose rates. Hands-on decontamination work using mops, wipes and assorted cleansers will be done only where significant reductions in local area radiation dose

rates can be achieved. Surface decontamination efforts beyond those outlined above will generally be performed as part of the second phase of operations discussed in Section 5.2.

2.2.1.2 Fuel Fragment Dissolution

It is presumed that many of the fuel rods in the reactor core were damaged during the March 28 accident, with resulting loss of cladding integrity and potential for dispersion of fuel particulates into the reactor coolant system (RCS). While most of the larger fuel fragments probably remain within the reactor pressure vessel, it is anticipated that many small fuel fragments were carried out into those portions of the RCS that are external to the reactor pressure vessel. Thus, there may be collections of small fuel fragments at various locations within the RCS, which may create strong local radiation fields due to the intense radioactivity present in the irradiated fuel. These local hot spots create two problems regardless of which choice is made for the future disposition of TMI-2: (1) the strong local radiation fields will cause increased radiation dose to the personnel working in the vicinity of these spots unless carefully shielded, an action which in itself will result in additional radiation dose to the workers, and (2) the possible dispersion of these very small and very radioactive fuel fragments outside of the RCS in the event of equipment disassembly or replacement represents an additional hazard to the workers. Past experience has shown simple flushing to be relatively ineffective for removing very small fuel particles from complex fluid handling systems. Therefore, it is postulated that an effort is made, following defueling of the reactor pressure vessel, to chemically dissolve and remove any fuel fragments not previously removed. The fuel dissolution step is followed by a chemical decontamination of the RCS, to remove most of the remaining fission products and activated corrosion products plated out on the interior surfaces of the system. Nuclear criticality considerations would be addressed in assessing fuel dissolution options.

2.2.1.3 RCS Chemical Decontamination

The chemical decontamination step is required to further reduce the radiation dose to workers within the containment building, regardless of which alternative for facility disposition is eventually implemented. The rationale for doing the chemical decontamination at this time is that doing so preserves all of the options for future disposition of the facility, in a way that is quite effective in terms of radiation exposure reduction. Failing to perform the chemical decontamination before placing the facility in safe storage might preclude ever taking such action, since the pumps, valves, and associated equipment required to handle the decontamination solutions would most likely be unusable after an extended storage period, and the impacts of the equipment refurbishment that would be needed to accomplish a chemical decontamination after an extended period could be significant.

2.2.2 Decommissioning Alternatives

Decommissioning means to remove the facility from service and to safely dispose of the radioactive residues.[7] Once the initial surface decontaminations are completed, the reactor defueled, the fuel debris removed, and the fission product and activation product depositions removed from the coolant systems, decommissioning of the nuclear steam-supply system can begin. Several alternative approaches to decommissioning are possible. These alternatives range from minimal further cleanup with subsequent physical security under appropriate nuclear licensing restrictions, to complete cleanup and removal of all radioactivity from the facility and its site with release of the plant from all nuclear licensing restrictions.

Three alternatives can be used for decommissioning reactor facilities: Immediate removal of radioactive materials (DECON), safe storage with deferred removal of radioactive materials (SAFSTOR), and entombment of radioactive materials (ENTOMB). DECON permits termination of the facility operating license, while SAFSTOR and ENTOMB require the continuance of an amended version of the license for extended periods of time, since licensable quantities of radioactive material remain onsite (i.e., the facility becomes an interim waste storage site). The amended nuclear license allows the licensee to possess but not operate the facility. Each of these alternatives, as applied to TMI-2, is defined and discussed in the following subsections, with supporting analyses and details given in Appendix U.

2.2.2.1 Immediate Dismantlement (DECON)

DECON is a pseudo-acronym defined by the NRC as the immediate removal of all radioactive material to permit unrestricted release of the property.[7] To achieve this condition, the residual radioactivity levels in the facility must be sufficiently small such that members of the public who

may occupy the decommissioned facility would receive a negligible amount of radiation. This condition is much more stringent than would be required for refurbishment and restart of the facility.

DECON meets the requirements for termination of the facility operating license and renders the facility and site available for unrestricted release within a finite period of time. In exchange for prompt availability of the facility and site for other purposes, large commitments of personnel radiation exposure, disposal-site space, and money are required. Another factor favoring DECON is the availability of the facility operations staff to form a decommissioning work force that is highly knowledgeable about the facility. Elimination of continuing security, maintenance, and surveillance requirements (i.e., continuing care for SAFSTOR or ENTOMB) is of small benefit on a site such as TMI which has another nuclear power unit on the property.

2.2.2.2 Safe Storage with Deferred Decontamination (SAFSTOR)

SAFSTOR means to fix and maintain the property so that risk to safety is acceptable for a period of storage followed by deferred decontamination and/or decay to an unrestricted level.[7] SAFSTOR consists of: (1) a period of facility and site preparation, which includes removal of fuel and concentration and immobilization of dispersible radioactive materials, followed by (2) an interim period of continuing care (i.e., safe storage) that encompasses security, surveillance, and maintenance, and concludes with (3) the deferred removal of any remaining radioactivity. The nuclear license remains in force throughout the safe storage period since materials having radioactivity levels above unrestricted release levels are still onsite. SAFSTOR could be modified to include storage of the irradiated fuel on the site for a finite period of time.

Deferred decontamination includes whatever actions are required at the end of the period of continuing care to terminate the nuclear license and to release the property for unrestricted use. Some disassembly and disposal of activated components is still required, but the personnel radiation exposure and the disposal-site space requirements are potentially greatly diminished. Deferred decontamination cannot, however, rely on the availability of facility operations staff for personnel familiar with the facility.

2.2.2.3 Entombment (ENTOMB)

ENTOMB means to encase and maintain property in a strong and structurally long-lived material (e.g., concrete) to assure retention and isolation from the environment until the contained radioactivity decays to an unrestricted level.[7] Depending on the nature of the entombed radioactivity, the necessary entombment period can range from about 100 years for short-lived contaminants such as Co-60 to many thousands of years for transuranic nuclides in spent fuel.

ENTOMB is similar in nature to SAFSTOR in that it also consists of a period of facility and site preparation, which includes removal of fuel and concentration and immobilization of dispersible radioactive materials, followed by a period of continuing care that includes security, surveillance, and maintenance activities. ENTOMB also requires a nuclear license to remain in force as long as the entombed radioactivity exceeds unrestricted release levels. The facility and site preparations include comprehensive cleanup and decontamination outside of the entombment structure and confinement of nonreleasable materials within the monolithic structure. Continuing care activities are minimal, unless the irradiated fuel is stored onsite.

In the ENTOMB strategy considered for TMI-2, the fuel is removed, but the reactor vessel and internals are left in place. As much as possible of the radioactive equipment from outside the entombed structure is consolidated and entombed within. Under existing regulations, the nuclear license must remain in force for an indefinite period of continuing care, until either the entombed radioactivity has decayed to unrestricted release levels or the entombment structure is dismantled and the entombed radioactivity removed.

When it becomes desirable to terminate the nuclear license for ENTOMB, dismantling of the entombment structure will probably be required. This represents a task that is much more difficult than dismantling the unentombed facility, since the entombment structure is built to endure for a long period of time. Therefore, while dismantlement of the entombment structure is not technically impossible, ENTOMB would be an almost irreversible creation of a radioactive waste repository on the site and commitment to long-term maintenance of the nuclear license.

2.2.3 Waste Volumes Generated during Decommissioning

The quantity of radioactive waste requiring disposal varies significantly with the decommissioning alternative selected, with DECON producing the largest volume, followed by ENTOMB, and then SAFSTOR. Estimates of the radioactive waste volumes requiring disposal are developed in Appendix U and are summarized in Table 2.1.

Table 2.1. Estimated Volumes of Radioactive Waste from Decommissioning Alternatives

	Burial Volume (ft^3)[a]			
Alternative	Activated	Contaminated	Radwaste	Total
DECON	40,600	609,000	22,000	671,000
SAFSTOR[b]	--	--	22,000	22,000
ENTOMB[b]	--	372,000	22,000	394,000

[a]Values are rounded to three significant figures.

[b]Values include only those wastes from initial decommissioning activities. Cumulative volumes including those from deferred decontamination should not exceed the DECON values.

The NRC staff considers ENTOMB to be an unacceptable alternative because its implementation would (1) create a long-term radioactive waste repository on the TMI site and (2) require maintenance and administrative controls for a length of time considered nonfeasible (100 years or longer).

In the case of DECON, all of the radioactive wastes must be removed from the facility and its site to satisfy the conditions for termination of the nuclear license and unrestricted use of the property. In the case of SAFSTOR, all or part of the packaged wastes could be placed in storage within the facility for the duration of the storage period, with transfer to a low-level waste burial ground when deferred decontamination begins. In the case of ENTOMB, all of the residual radioactivity must be confined within the entombment barriers. Storage of packaged wastes within the facility but outside of the barriers would not be in compliance with the conditions defined for entombment. Thus, all of the packaged wastes outside the barriers would have to be shipped to a disposal site during the ENTOMB operations.

2.2.4 Effluents and Releases to the Environment

Decommissioning operations are designed such that the containment/confinement capability of the facility is maintained while the operations are in progress. As a result, the airborne releases of radioactive materials from decommissioning operations are quite small, and the associated radiation doses to the public are also quite small. As discussed in elsewhere in Section 2.2, accident-related radioactive liquids are already processed and the radioactivity concentrated and solidified. Therefore, no significant releases of radioactivity in liquids are postulated to occur during any of the decommissioning operations.

The estimated releases of airborne radioactivity resulting from decommissioning operations are summarized in Table 2.2. Not all types of operations take place in all decommissioning alternatives. The releases listed represent upper-bound estimates for each type of operation.

2.2.5 Environmental Impacts

The principal environmental impacts resulting from decommissioning are the radiation doses received by the decommissioning workers and the volume of space required at a low-level waste disposal site for burial of the radioactive wastes. Some small radiation doses to the populace

in the vicinity of the TMI site could result from the releases of airborne radioactivity postulated in Section 2.2.4. Likewise, some small radiation doses to the populace along the transportation routes between the site and the low-level waste burial ground could result from the radiation emanated by the packaged wastes while in transit. These impacts are summarized in Table 2.3, with brief discussions of the development of these dose estimates presented in the following subsections.

Table 2.2. Postulated Releases of Airborne Radioactivity to to the Environment during Decommissioning Operations

Decommissioning Operation	Reference Radionuclide Inventory[a]	Airborne Release (Ci)		
		DECON	SAFSTOR	ENTOMB
Segmenting Contaminated Equipment	4	6×10^{-6}	--	5×10^{-6}
Activated Concrete Removal	3	0.2×10^{-6}	--	--
Contaminated Concrete Removal	5	1.6×10^{-9}	--	1.4×10^{-9}
Water-Jet Cleaning	5	9.8×10^{-6}	1.14×10^{-5}	9.8×10^{-6}

[a]See Tables U.8 through U.11 in Appendix U for characterization of these inventories.

Table 2.3. Estimated Cumulative Radiation Doses Resulting from Decommissioning Operations

Radiation Dose Recipient	Cumulative Radiation Dose (person-rem)[a]		
	DECON	SAFSTOR[b]	ENTOMB[b]
Decommissioning Worker	1800	220	1300
Transport Worker			
Truck	160	13	85
Rail	6	6	6
Public			
Normal Decommissioning Activities	6×10^{-5}	7×10^{-5}	6×10^{-5}
Radwaste Transport	74	6	40
Spent Fuel Transport	1	1	1

[a]Results rounded to two significant figures.

[b]Values include only doses from initial decommissioning activities. Doses from deferred decontamination will be no more than the doses from DECON, decreasing with increased deferral time.

2.2.5.1 Occupational Radiation Doses

The radiation doses received by decommissioning workers are estimated by multiplying the number of exposure hours experienced by the direct decommissioning crews while performing a given task times the average local radiation dose rate in the area where the task is performed, and summing over all tasks. A detailed estimate for each of the decommissioning alternatives is developed in Appendix U, with the summaries for each major structure presented in Table 2.4. The estimated exposure hours and doses do not include any contributions from supervision and support staff not directly engaged in the decommissioning activities in radiation zones. Radiation doses received by these types of personnel are postulated to be small, not exceeding a few percent of the amount received by the direct decommissioning crews, and well within the accuracy of the estimates made for the work crews.

The radiation doses received by transportation workers resulting from the shipment of radioactive materials from the site to a low-level waste burial ground and from the shipment of the irradiated reactor fuel from the site to a storage or disposal facility are estimated in detail in Appendix U. The number and type of shipments, together with the associated radiation doses, are presented in Table 2.5 for each of the decommissioning alternatives.

Table 2.4. Estimated Cumulative Radiation Doses Received by Direct Decommissioning Workers

	DECON		SAFSTOR[a]		ENTOMB[a]	
Structure	Exp. Hrs.	Person-Rem	Exp. Hrs.	Person-Rem	Exp. Hrs.	Person-Rem
Containment Bldg.	110,300	1,500	15,400	82	110,000	1,020
AFH Bldg.	139,000	278	53,900	121	107,000	213
Other Bldgs.	18,000	35	5,000	15	18,000	35
Totals[b]	267,000	1,810	74,300	218	235,000	1,270

[a]Values include only doses from initial decommissioning activities. Doses from deferred decontamination will be no more than the doses from DECON, decreasing with increased deferral time.

[b]Totals rounded to three significant figures.

Table 2.5. Estimated Cumulative Radiation Doses Received by Transportation Workers

Decommissioning Alternative	Truck Transport		Rail Transport	
	No. Shipments	Person-Rem	No. Shipments	Person-Rem
DECON	1,400	157.0	26	6.2
SAFSTOR[a]	114	12.8	26	6.2
ENTOMB[a]	755	84.6	26	6.2

[a]Values include only doses from initial decommissioning activities. Doses from deferred decontamination will be no more than the doses from DECON, decreasing with increased deferral time.

2.2.5.2 Offsite Radiation Doses

The dose estimates presented in Section 2.2.5 and listed in Table 2.3 for normal decommissioning activities are developed in Appendix U. Also in Appendix U are dose estimates to the maximum-exposed individual. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3.

Radiation doses to the populace residing along the transportation routes between TMI and disposal sites are estimated assuming each shipment emits gamma radiation at the maximum level permitted by DOT regulations. The resulting estimated radiation doses, which are upper-limit estimates, are also shown in Table 2.3. Actual experience indicates that these doses will be much less. The significance of these doses is discussed in Section 10.3.

2.2.5.3 Other Environmental Impacts

Other environmental impacts resulting from the decommissioning of TMI-2 are very similar to those for the cleanup effort discussed in Section 10.6, but of a lesser magnitude since fewer workers will be involved (~500,000 person-hours) and since the gross decontamination and cleanup efforts will have been completed prior to the start of decommissioning.

Removing all of the radioactive materials from TMI-2, as would occur for DECON, may tend to reduce the anxiety level and psychological stress of the local residents. However, transporting material from TMI to a disposal site may heighten the anxiety level and psychological stress of those people residing along the transport routes.

Placing TMI-2 in safe storage will require even fewer workers than DECON or ENTOMB (~125,000 person-hours), thus resulting in a smaller local payroll. The retention of the bulk of the radioactive materials onsite during the safe storage period might tend to continue the existing levels of anxiety in the local community, even though the readily dispersible radioactive materials have been solidified and packaged. Since transport of the radioactive materials from TMI to a disposal site produces the largest estimated radiation dose to the public of all of the decommissioning operations, <u>not</u> shipping that material would tend to reduce the anxiety level among the populace along the transport routes.

Entombing the lower levels of the containment building would have an impact on the local payroll very similar to that of DECON. In terms of public anxiety and psychological stress, ENTOMB combines the worst features of DECON and SAFSTOR, since a large amount of radioactive material is transported offsite but the entombed reactor containment building remains as essentially a permanent waste repository.

2.2.6 <u>Decommissioning Costs</u>

The cost estimates for decommissioning TMI-2 following the initial cleanup effort, removal of the irradiated fuel from the reactor, and dissolution of fuel debris and chemical decontamination of the RCS and associated fluid-handling systems are developed in Appendix U (Sec. U.3.5 for DECON, Sec. U.4.7 for SAFSTOR, and Sec. U.5.5 for ENTOMB), and are summarized in Table 2.6. These estimates are based in large part on information developed in earlier studies of decommissioning a large pressurized water reactor power station,[8,9] with labor and materials escalated by 17 percent to bring their costs to mid-1980 levels. Energy costs estimated earlier[8] are escalated by 32 percent. Costs for transporting radioactive materials by truck are based on current rates for exclusive-use vehicles. Handling and disposal charges at a low-level waste burial ground are based on a price list (dated November 17, 1980) issued by the Nuclear Engineering Company for their Richland, Washington, site.

2.3 DISPOSAL OF RADIOACTIVE WASTE

The accident at TMI-2 led to the generation of materials that contain or are contaminated with radionuclides. These radioactive materials, in the form of solids, liquids, semisolids, and gases for which no further use is foreseen, are referred to as radioactive wastes. These radioactive wastes must be managed from their initial generation through their ultimate disposition in a manner consistent with personnel and public safety, minimum spread of contamination, and compliance with regulatory requirements.

This section presents a discussion of the major alternatives for disposal of radioactive wastes that could be generated by the decontamination and defueling of TMI-2. The forms of waste are

characterized, regulatory requirements that define the constraints for various steps within the waste management cycle are discussed, and the alternatives considered for waste treatment, conditioning, packaging, shipment, and ultimate disposition are discussed.

Table 2.6. Estimated Costs for Decommissioning TMI-2 Via DECON, SAFSTOR, and ENTOMB

Cost Category	Costs (millions of mid-1980 dollars)[a] DECON	SAFSTOR	ENTOMB
Decommissioning Labor			
Direct	7.200	1.890	7.200
Support	8.900	3.048	8.920
Radwaste Disposal	14.600	1.210	8.500
Spent Fuel Shipment	2.500	2.500	2.500
Energy	4.620	2.640	4.620
Other Costs			
Supplies	1.820	1.040	1.820
Equipment	0.960	0.090	0.960
Contractors	0.640	0.123	0.510
Nuclear Insurance	0.940	0.344	0.940
Licensing Fees	0.060	0.045	0.050
Subtotal	42.200	12.900	36.000
25% Contingency	10.600	3.230	9.000
Totals	52.8	16.2	45.0
Annual Continuing Care Cost	--	0.060	0.040
Deferred Decontamination	--	43.8	~44.0

[a]Values rounded to three significant figures.

2.3.1 Waste Characterization

The radioactive wastes at TMI-2 arise from the accident and from decontamination operations. Each source generates primary waste, which is the form of the radioactive material at the time it is generated, and secondary waste, which is the form of the material that arises from treatment of the primary waste. The types of waste that could be generated at TMI-2 are shown by source in Figure 2.2.

A major factor governing the alternatives to management of radioactive wastes is its physical form. The waste types shown in Figure 2.2 can be divided into four categories which reflect their physical characteristics: (1) solid materials (dry materials in bulk form), (2) process solids (wet solids in the form of sludges, high-solids-content slurries, or granular materials and ion-exchange media and spent filter cartridges), (3) liquids (liquids contaminated with radioactive materials), and (4) irradiated fuel assemblies, (5) gases (the Kr-85 in the reactor building). The characteristics of each waste type within these categories are described below.

2.3.1.1 Solid Materials

Irradiated Hardware

These wastes consist of the structural hardware and other components within the reactor vessel at the time of the accident and the debris generated as a result of fuel assembly and component

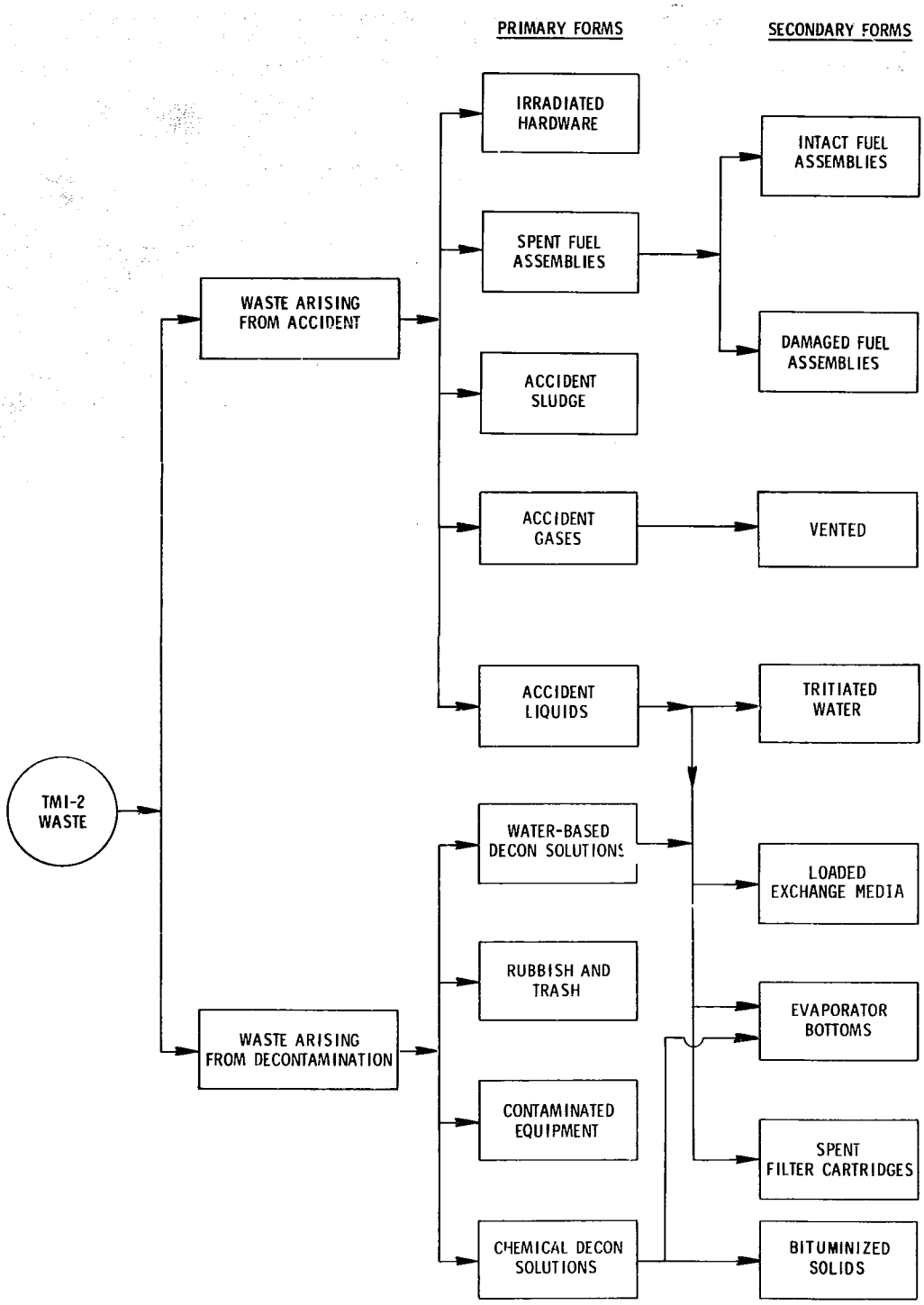


Figure 2.2. Characterization of TMI-2 Radioactive Waste Forms.

damage during the accident. The wastes generally are solid metals and represent a low- to relatively high-activity source of solid waste material. Core internals and components have been removed from other LWRs and disposed of as waste. However, the possible extent of damage to these components and the nature of the debris that could be present in the reactor vessel make these wastes unique to TMI-2.

Trash

Radioactive trash generated during the decontamination operations will consist of compactible and noncompactible solid material, some of which also is combustible. The compactible and combustible solids consist of disposable clothing, rags, plastic covers, laydown pads, and miscellaneous trash. The amount of material generated is proportional to the number of personnel engaged in decontamination and the time required to complete decontamination operations. The noncompactible solids consist of tools, hoses, safety goggles, miscellaneous construction materials, and other small items of equipment used by decontamination personnel. The amount of noncompactible solids generated also is dependent on the number of personnel involved and the time required for decontamination. The form and specific activity of the solid waste generated by the decontamination crew is comparable to the solid waste generated from decontamination operations at LWRs and other nuclear facilities.

Contaminated Equipment

It may not be possible to decontaminate some of the equipment contaminated during the accident. This equipment, in the form of motors, pumps, valves, instrumentation, and other components, will be handled as radioactive waste. The form and specific activity of this type of solid radioactive waste is comparable to that generated from decontamination operations at other nuclear facilities.

2.3.1.2 Liquids

Accident Water

Accident water, which will be handled as radioactive waste, is present in two locations: the reactor building (RB) and the reactor coolant system (RCS). The accident water which collected in the AFHB after the accident has been processed and the processed water is presently stored in tanks onsite. The water in the reactor building sump is somewhat complex in both chemical and physical makeup. The reactor building water includes sludge, high total suspended and dissolved solids, and colloids. It also contains tritium, radioactive fission products, sodium, and boron. The specific activity of the reactor building water is higher than that of the water in the RCS. The water in the RCS contains tritium, radioactive fission products, sodium, and boron. The water is contaminated with radioactive fission products and for some radionuclides has specific activity levels higher than the reactor building sump water and may contain fuel particles and transuranics. Processing of accident water using ion exchange, evaporation, or filtration techniques could lead to the generation of secondary waste forms consisting of tritiated water, process solids in the form of loaded ion-exchange materials and evaporator bottoms, and solid waste in the form of spent filter cartridges.

Tritiated Water

There are no practical large-scale techniques available to decontaminate accident water to remove the tritium originally present and all techniques will leave trace amounts of fission products. Therefore the processing will result in wastewater containing tritium at concentrations essentially the same as before treatment. This tritiated water will represent a fairly high volume of very low-specific-activity liquid waste.

Water-Based Decontamination Solutions

Some of the decontamination procedures that will be used involve the use of water to wash down internal building surfaces and equipment. This water will become contaminated with the radioactive materials it washes off the surfaces. The chemical and physical characteristics of this contaminated water will depend on the contamination levels of the surfaces washed, the procedure used for application of the water, and the extent to which detergents are used. Generally, such water-based decontamination solutions contain suspended solids, fission products, and small amounts of chemical contaminants and detergents. The solutions represent a relatively large volume of low-specific-activity liquid waste comparable to the solutions generated from

decontamination of other types of nuclear facilities. Processing of these solutions using ion exchange or evaporation could lead to the generation of secondary waste forms consisting of process solids in the form of loaded ion-exchange materials and evaporator bottoms.

Chemical Decontamination Solutions

Decontamination operations also can be performed with chemical solutions (e.g., strong detergent solutions or foam-type decontamination agents). The physical and chemical characteristics of these decontamination solutions are more complex than those arising from the use of water. Their effectiveness could be greater than that of water-based solutions so that the resultant liquid may have higher specific activities than would water-based solutions. The volume of chemical solutions generated could be less than the volumes of water-based liquids generated. While relatively large volumes could be generated, the nature of these chemical solution liquid wastes and their specific activity would be comparable to that generated from decontamination operations at other types of nuclear facilities. Processing of these chemical solutions using evaporation could lead to the generation of secondary waste in the form of evaporator bottoms. Bituminization could lead to secondary waste in the form of bituminized solids.

2.3.1.3 Process Solids

The process solid wastes that arose from the accident and could arise from the treatment of accident water and decontamination liquids are discussed in this section.

Accident Sludge

This form of process solids arose from the accident and is present in the sump of the reactor building and in tanks and sumps in the auxiliary and fuel handling buildings. The chemical and physical characteristics of the sludge will be relatively complex and variable. This sludge is contaminated with radioactive fission products and will represent a relatively low-volume, high-specific-activity form of radioactive waste. The nature of this radioactive waste is not directly comparable in form or specific activity to sludges generated by other LWRs.

Loaded Ion-Exchange Materials

The use of ion-exchange media in the form of organic resins or zeolites to remove fission product contaminants from the liquids could lead to the generation of process solids in the form of loaded ion-exchange materials. The specific activities and radionuclide contents of the loaded ion-exchange media will vary with the liquids processed through the media, the contaminants selectively removed from the liquids, and the capacity of the media to retain contaminants. Some of the liquids that could be treated using ion-exchange techniques will generate wastes with specific activities well above those normally generated in LWRs, while processing of other liquids will lead to wastes with specific activities well within the range normally generated by a PWR. Thus, loaded ion-exchange materials could represent a relatively high volume of process solid waste with specific activities in the low to very high range relative to the expended ion-exchange material generated by other PWRs. The relatively high specific activity and nature of the fission product contaminants on some wastes will make them unique to TMI-2.

Evaporator Bottoms or Sludges

The use of evaporation techniques to reduce liquid waste volumes could lead to the generation of process solids in the form of evaporator bottoms or sludges. The physical characteristics of these process solids would depend on the solids content of the liquids evaporated and the equipment used for evaporation. These characteristics could range from slurries containing 10-20 weight percent solids to sludges with solids contents in excess of 50 weight percent.

The specific activities of these process solid wastes also could vary over a wide range, and could be above those normally generated by LWRs. Evaporation of decontamination solutions could lead to higher-than-normal concentrations of chelating agents in these process solids.

Filter Cartridge Assemblies

This form of solid waste will arise from the treatment of liquids. Filter cartridge assemblies are typically right-circular cylinders which are used to remove particulates from liquid waste; the contaminated particulates are deposited on the filter. They represent a low- to very-high-specific-activity form of solid waste, with their specific activity dependent on the contaminants

in the waste stream processed. Most of these cartridges are similar to those used for liquid treatment in other LWRs, and their specific activity could be comparable to the higher-activity filter assemblies generated at other pressurized water reactors (PWRs). The nature of some of the contaminants in certain of the TMI-2 liquid wastes may lead to the generation of spent filter cartridges with unique contaminants and specific activities higher than normal.

Bituminized Solids

The use of bituminization could lead to the generation of process materials in the form of bituminized solids. These wastes will be in the form of a monolithic solid containing about 50 weight percent evaporated salts and 50 weight percent bitumen.

2.3.1.4 Irradiated Fuel Assemblies

The fuel assemblies that make up the reactor core were damaged in varying degrees during the accident. These irradiated fuel assemblies contain uranium, plutonium, and assorted fission products and are high level wastes. The possible extent of damage to these fuel assemblies makes them unique relative to the irradiated fuel assemblies at other light water reactors (LWRs).

2.3.1.5 Gases

Since essentially all of the Kr-85 gas in the reactor building atmosphere that arose from the accident was vented* into the outside atmosphere, no secondary waste forms will arise.

2.3.2 Classification

Regulatory classification of radioactive waste is a factor in evaluating alternatives for ultimate disposition of these waste materials. Classification will determine the standards applicable to packaging and transportation and the types of facilities that can be used for their ultimate disposition. Federal regulations concerning radioactive materials are discussed in Section 1.6. Historically, three classes of waste commonly have been referred to:

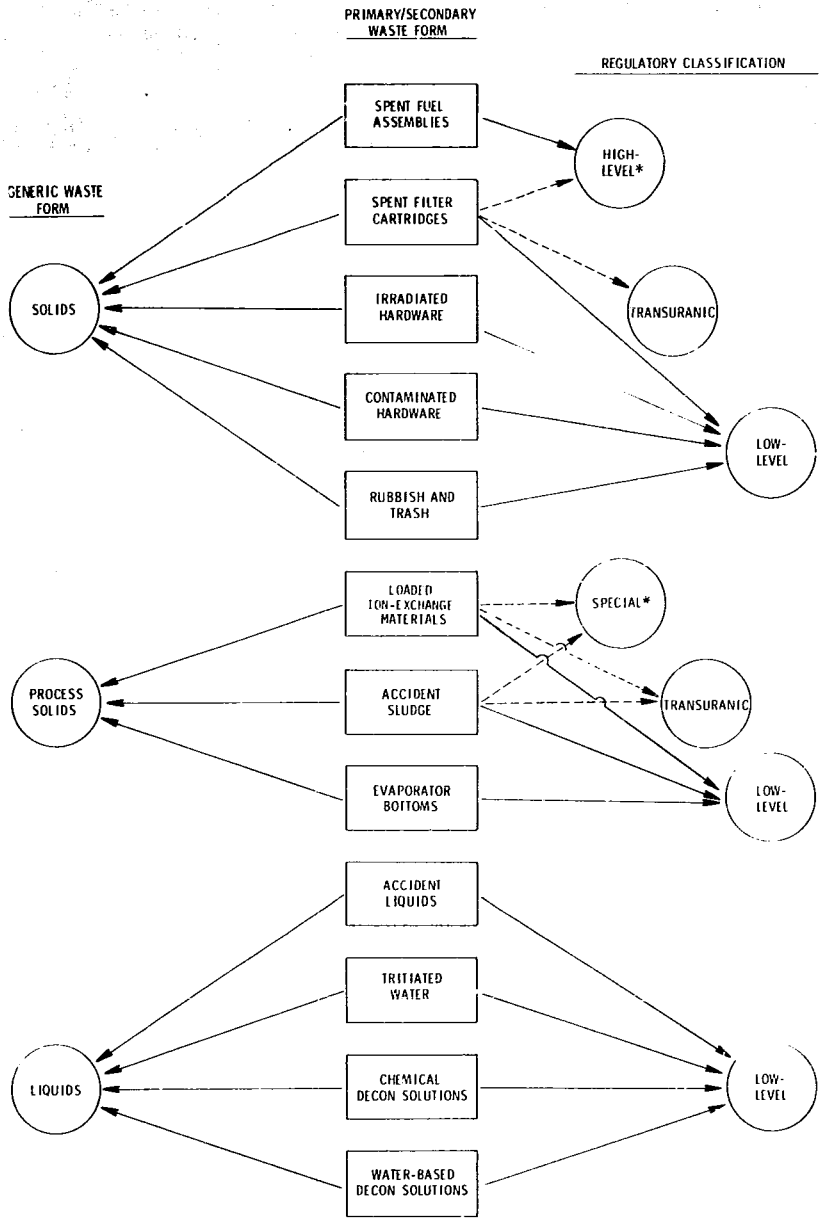
- High-level waste (HLW). These wastes are spent fuel or the wastes resulting from irradiated nuclear fuel during reprocessing.
- Transuranic (TRU) waste. These wastes are made up of materials which contain or are contaminated by transuranic elements, with atomic numbers higher than uranium. When waste material contains concentrations of transuranic elements in excess of 10 nanocuries per gram, they are considered TRU waste.
- Low-level waste (LLW). These wastes are made up of all radioactive waste materials that are not HLW or TRU waste.

It should be noted that these three broad waste classifications have evolved over the years based upon operations of the nuclear industry. In this regard, no regulatory framework was developed to specifically address the types of unique waste forms that may be generated at TMI-2 as a result of the accident. Accordingly, the wastes resulting from TMI-2 cleanup will have to be carefully reviewed on a case-by-case basis with regard to important characteristics such as specific activity, radionuclide content, total radioactivity inventory, and waste forms and stability.

Ultimate disposition of such wastes will depend on the unique characteristics they possess and on suitable facilities for their handling and disposal. For example, because of the high concentrations of certain long-lived nuclides (approximately 30-year half-lives), particularly Cs-137 and Sr-90, certain forms of TMI-2 wastes may not be suitable for handling by routine shallow land burial techniques. The waste materials affected could include very high specific activity loaded ion-exchange materials, accident sludges, and spent filter cartridges.

The primary and secondary waste forms and their potential dispositions considering the current regulatory framework are shown in Figure 2.3. Many of the wastes can likely be classified under

*Residual amounts of Kr-85 from offgassing in the reactor building continue to be vented from the reactor building (i.e., prior to personnel entries). In addition, the fuel still contains approximately 45,000 additional curies of Kr-85, some of which could be released during defueling.



*THESE WASTE FORMS MAY REQUIRE MORE RESTRICTIVE HANDLING AND DISPOSAL THAN APPLIED IN ROUTINE SHALLOW LAND BURIAL.

——————— KNOWN CLASSIFICATION FOR WASTE FORM

- - - - - - - - POTENTIAL CLASSIFICATION FOR A PORTION OF THE WASTE FORM

Figure 2.2 TMI-2 Waste Forms and Classification

present interpretations as low-level waste or transuranic waste. However, because various characteristics of some of the wastes that will be generated are unique to TMI-2, higher standards of care than those required for routine standard treatment of low-level waste might have to be applied. Thus, certain batches of loaded ion-exchange materials, accident sludges, and individual spent filter cartridges are likely to require special consideration and treatment. The standards applied to these unique wastes will be considered on a case-by-case basis and, where warranted, these wastes will be handled, packaged, and disposed of in accordance with special requirements.

2.3.3 Alternatives for Disposition of Radioactive Waste

The alternatives considered for ultimate disposition of TMI-2 radioactive waste depend on the physical and chemical characteristics of the radioactive materials, their radionuclide content and specific activity, and the regulations governing packaging, transportation and disposal. These factors establish the boundary conditions for consideration of the alternatives or options available within the waste management cycle. The steps within the radioactive waste management cycle at TMI-2 are shown in Figure 2.4 and are discussed below.

Collection and Segregation. In the initial step, waste materials are accumulated and segregated according to chemical and physical form and radionuclide content.

Treatment. Treatment is a step which refers to the processing operations leading to the concentration of radionuclides into a smaller volume. For liquids, the radioactivity of the treated liquid is substantially reduced, enabling it to be disposed of by various means or reused.

Conditioning. Conditioning is a step which refers to those operations that transform the concentrates produced during treatment or untreated materials into forms suitable for transportation or disposal. Conditioning includes immobilization, which converts radioactive waste materials in the form of liquids and process solids into a stable immobile form with the radioactive materials homogeneously dispersed within it. Conversion to this form minimizes radionuclide release to the environment during storage, transportation, and disposal. A detailed description of the techniques used to immobilize radioactive waste and the characteristics of the resultant forms using a variety of binder materials is presented in Appendix H.

Packaging. Packaging refers to placement of the radioactive material into a disposable container. The container is a barrier to radionuclide release and a means of handling the waste during interim storage, transportation, and disposal. The container also may serve as the vessel in which immobilization operations are performed.

Package Handling. Package handling refers to those operations that move containers within the facility. The methods used for handling depend on the radiation level of the container and its size and weight. These methods include hands-on, semiremote, and remote handling, as well as placement of the container within a transfer shield to reduce radiation levels.

Storage. Storage is the containment of radioactive wastes in a manner that provides for their subsequent retrieval. Storage onsite could be required for varying time periods depending on the availability of suitable offsite disposal, processing, and storage facilities.

Shipment. Shipment refers to placement of radioactive waste on transport vehicles and transfer to another facility. Unshielded shipments contain radioactive materials in containers with radiation levels that comply with transportation regulations without the addition of shielded overpacks. Shielded shipments are those where the containers are placed within shipping casks that provide shielding and additional mechanical integrity during transport.

Disposal. Disposal refers to the emplacement of the radioactive waste containers in facilities without intending to retrieve them. Discharge, which is a form of disposal, refers to the controlled release of liquids or gases to the environs.

As shown in Figure 2.4, some liquids may be suitable for discharge following treatment. Additionally, shipment to another facility would not necessarily entail disposal. Some of the wastes will have characteristics of routine low-level waste and may be disposed of by routine shallow land burial. Some of the wastes might contain transuranic contaminants with concentrations in excess of 10 nCi/g, which would require their shipment to a transuranic waste storage facility. Other wastes expected to be generated could have specific characteristics which preclude their disposal at a commercial low-level disposal facility by routine methods and special measures may

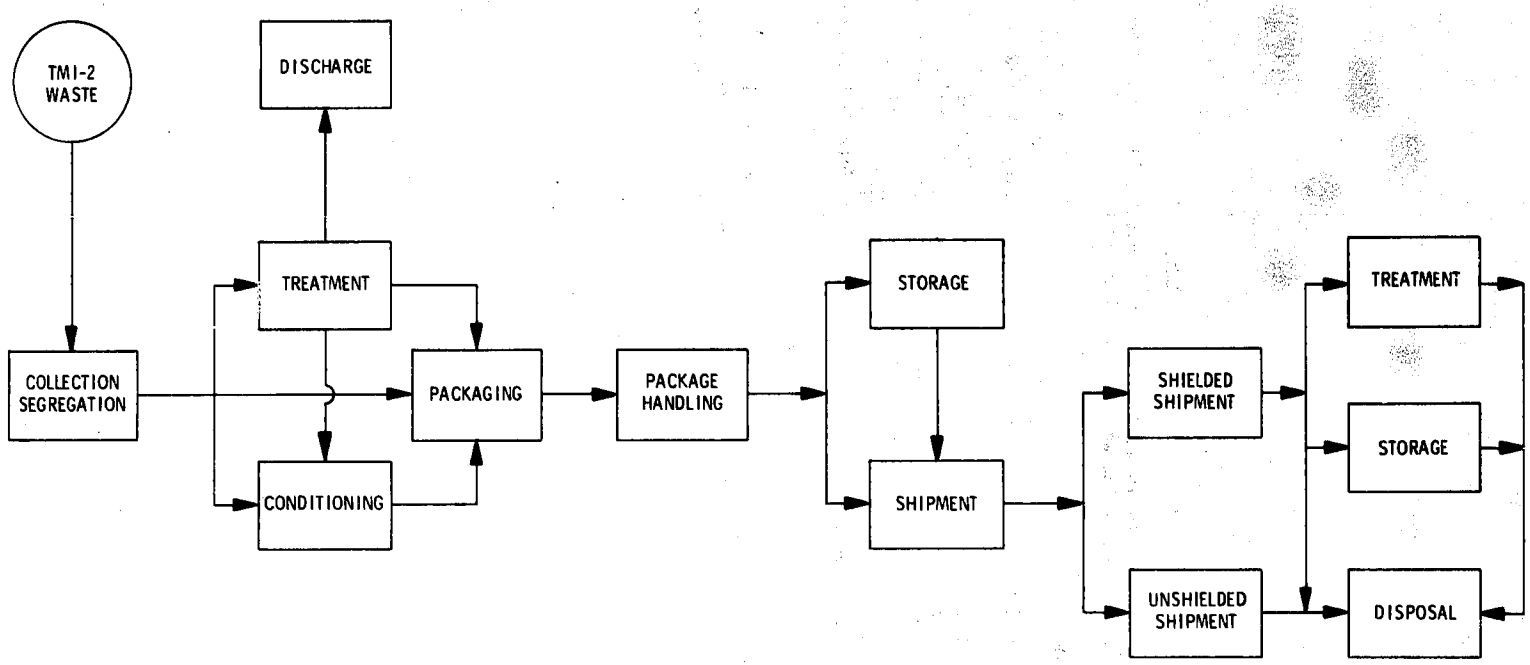


Figure 2.4. Radioactive Waste Management Cycle.

have to be taken. Fuel assemblies also could be shipped to treatment or storage facilities which handle spent fuel.

Within the steps shown on Figure 2.4, the alternatives considered for each waste type during each step are variable but can be summarized for each of the five waste form categories described in Section 2.3.1. A summary of these waste management alternatives, which are described in detail in subsequent sections of this document, is presented below.

2.3.3.1 Alternatives for Management of Solid Materials

The alternatives considered for solid materials are dependent on their physical form, specific characteristics, and classification, where relevant. The major steps within the waste management cycle that were considered as a function of waste type are shown in Table 2.7. Only those waste management steps applicable to solid materials are shown. Accordingly, discharge to the environs and conditioning are not included.

Table 2.7. Alternatives for Management of Solid Materials

	Waste Types[a]			
Waste Management Alternatives	Trash	Contaminated Equipment	Irradiated Hardware	Fuel Assemblies
Treatment				
Incineration	X			
Compaction	X			
Disassembly/sectioning		X	X	
Packaging				
55-gallon drums	X			
LSA boxes	X	X	X	
Steel liners		X	X	
Special containers				O
Package Handling				
Hands-on	X	X	X	
Remote		X	X	X
Transfer shield			X	
Onsite storage				
Interim	X	X	X	X
Long-term				O
Shipment				
Unshielded	X	X	X	
Shielded		X	X	X
Disposal				
Commercial low-level	X	X	X	
Transuranic				
High-level treatment/storage				O

[a]X denotes alternative considered for waste type.
O denotes alternative applicable to special waste only.

The treatment alternatives considered depend entirely on the physical form of the material. Combustible trash can be incinerated to reduce volume by factors of 80 to 100. Incineration leads to the generation of ash which is handled as a process solid. Noncombustible compactible trash can be compacted to reduce volumes by factors of about five. Contaminated equipment and irradiated hardware consists of metallic solids, and disassembly and mechanical sectioning are the only techniques considered for volume reduction.

The disposable containers considered for packaging include 55-gallon drums, boxes for low-specific-activity material, and rectangular or cylindrical steel containers with capacities as large as 200 ft^3. Special containers are required for fuel assemblies and could be required for spent filter cartridges if they are contaminated with transuranic contaminants with concentrations above 10 nCi/g or if they have high specific activities.

The package handling techniques considered depend on the radiation level of the disposable container, its size and weight. Low-specific-activity materials in drums or wooden boxes are handled manually and transferred within the facility using forklifts, motorized pallets, or other package-handling equipment. For higher-activity disposable containers, semiremote and remote handling systems are used to minimize radiation exposure to personnel. Individual containers also can be placed within transfer shields and thus permit the use of manual handling techniques similar to those used for low-specific-activity containers. Fuel assemblies are handled underwater using remote techniques.

Generally, all waste is transferred to an onsite storage facility to await shipment. Except for trash boxes, these storage facilities are shielded enclosures, and facilities are divided by container type and surface radiation level. Fuel assemblies would be stored in the spent fuel pool, and interim, as well as long-term, storage of these assemblies was considered. Long-term storage may also be required for spent filter cartridge assemblies that might have to be handled as transuranic or high-level waste.

The waste types shown will be transported offsite in both unshielded and shielded containers. Low-activity drums and LSA boxes can be shipped unshielded, while higher-activity drums and the other container types shown will be transported in licensed, shielded shipping casks.

The destination of each shipment depends on the characteristics of the materials being transported. Some waste will be transported to a commercial low-level waste disposal facility. Certain high-specific-activity wastes might require transport to special waste storage facilities. Special destinations were considered for fuel assemblies, including high-level waste storage.

A detailed discussion of the above alternatives and their impacts is presented in subsequent sections of this document.

2.3.3.2 Alternatives for Management of Liquid Waste

The alternatives considered for liquid waste are dependent primarily on the chemical characteristics, radionuclide content, and specific activity of those wastes. Table 2.8 contains a summary of the alternatives for liquid waste considered for each step within the waste management cycle shown in Figure 2.4.

The treatment alternatives considered for liquids include filtration, ion exchange, evaporation, and bitumenization. All four techniques were considered for accident water. The use of these techniques substantially reduces the specific activity of the water and leads to the generation of secondary wastes in the form of spent filter cartridge assemblies, loaded ion-exchange materials, and evaporator bottoms or sludges and bitumenized solids. Use of these techniques will not reduce the tritium concentration of the treated water; hence, discharge of tritiated water to surface waters or through evaporation, packaging for disposal as low-level radioactive waste, and deep-well injection were considered for this tritiated water. These treatment techniques were also considered for the waterbased decontamination solutions. However, the highly complex nature of some of the chemical decontamination solutions eliminates the use of ion-exchange techniques as feasible alternatives.

A conditioning step involving immobilization was considered for liquids that were not treated because with only few exceptions liquids cannot be transported or disposed of at commercial low-level waste disposal facilities. A detailed discussion of four different immobilization agents and the techniques used to implement them is presented in Appendix H. Of the four agents

described in that appendix, only three were considered for TMI-2 wastes. The use of urea formaldehyde was eliminated because of the problems associated with obtaining a liquid-free product. The remaining three--cement, vinyl ester styrene, and bitumen--were considered for most of the TMI-2 waste forms requiring immobilization. In cases of immobilization of liquids, disposable containers (drums or cylindrical steel liners) were considered for packaging. The package-handling techniques considered for these drums and steel liners are similar to those described in Section 2.3.3.1. Packaged, immobilized liquids will be stored onsite to await shipment with other packaged waste by container type and radiation level.

Table 2.8. Alternatives for Management of Liquids

Waste Management Alternatives	Waste Type[a]		
	Accident Water	Water-Based Decon Solutions	Chemical Decon Solutions
Treatment			
Filtration	X	X	
Ion exchange	X	X	
Evaporation	X	X	X
Bituminization	X		X
Discharge	O		
Conditioning			
Immobilize/cement	X	X	X
Immobilize/vinyl ester styrene	X	X	X
Packaging			
55-gallon drums	X	X	X
Steel liners	X	X	X
Package Handling			
Hands-on	O	X	X
Remote	X	X	X
Transfer shield	X	X	X
Onsite storage			
Interim	X	X	X
Long-term	O		
Shipment			
Unshielded	O	X	X
Shielded	X	X	X
Disposal			
Deep-well injection	O		
Commercial low-level	X	X	X

[a]X denotes alternative considered for waste type.
O denotes alternative for tritiated water only.

Shipment would be made either shielded or unshielded, as dictated by container radiation level. If tritiated water is immobilized, the radiation levels of this waste may be compatible with unshielded shipment. Some immobilized decontamination solutions also might be shipped unshielded. The destination of all shipments will be a commercial low-level waste disposal facility.

Detailed discussion of these alternatives and their impacts is presented in subsequent sections of this document.

2.3.3.3 Alternatives for Management of Process Solids

The alternatives considered for process solids are dependent primarily on their physical form, chemical characteristics, radionuclide content, and specific activity. Table 2.9 contains a summary of the alternatives considered for each step within the waste management cycle for each type of process solid. Only those steps applicable to this waste form are shown, and treatment and discharge were not considered.

The conditioning alternatives considered include dewatering and use of the three immobilization techniques considered for liquids. To permit disposal at a commercial low-level waste disposal facility, evaporator bottoms and incinerator ash must be immobilized. All three immobilization techniques were considered for evaporator bottoms and incinerator ash. Ion-exchange materials and sludges can be shipped to some commercial low-level waste disposal facilities in the dewatered condition, while other disposal facilities require these materials be in an immobilized form. EPICOR II ion-exchange materials used to process AFHB liquids are currently required to be solidified in accordance with Commission Order. The use of bitumen to immobilize accident sludges was not considered because of the expected high specific activity of the sludges and the low threshold for radiation damage of bitumen.

The disposable containers considered for packaging include 55-gallon drums and large cylindrical steel liners. Where ion exchange treatment techniques incorporated disposable demineralizer vessels, such vessels ranging in size from 10 ft^3 to 195 ft^3 also were considered as disposable containers. These disposable demineralizer vessels also were considered as the vessel within which immobilization of expended ion-exchange materials could be performed. Special containers also were considered for special ion-exchange materials and accident sludge. These waste materials could contain transuranic contaminants with concentrations above 10 nCi/g, requiring their handling as transuranic waste, or they also could have characteristics which might require special packaging, handling, storage, and disposal methods. Drums were the only disposable containers considered for incinerator ash because of the relatively low volume of this waste expected to be generated.

The surface radiation levels of all these disposable containers will require their handling by semiremote and remote techniques to minimize personnel exposure. Some of the handling techniques considered include underwater handling of ion-exchange material containers, as well as placement of individual containers within transfer shields for movement within the facility.

All these waste types will be transferred to an onsite storage facility. Special facilities to store ion-exchange material containers have been considered, including surface and subsurface cells and storage under water within the spent fuel pool. Where ion-exchange materials, evaporator bottoms, or accident sludges warrant special handling, long-term onsite storage also was considered.

All containers will likely be shipped in a shielded configuration. Some of these waste containers will be shipped to a commercial low-level waste disposal facility. Those containers with ion-exchange materials, evaporator bottoms or accident sludges which require special handling could be stored onsite for periods of 20 to 30 years or shipped to special facilities for either storage or treatment.

A detailed discussion of the above alternatives and their impacts is presented in subsequent sections of this document.

Table 2.9. Alternatives for Management of Process Solids

Waste Management Alternatives	Waste Type[a]				
	Loaded Ion-Exchange Materials	Evaporator Bottoms	Accident Sludge	Incinerator Ash	Spent Filter Cartridge Assemblies
Conditioning					
Dewatering	X		X		
Immobilize/cement	X	X	X	X	
Immobilize/vinyl ester styrene	X	X	X		
Packaging					
55-gallon drums	X	X	X	X	X
Steel liners	X	X	X		X
Special containers	0		0		0
Package handling					
Hands-on					
Remote	X	X	X	X	X
Transfer shield	X	X	X		X
Onsite storage					
Interim	X	X	X	X	X
Long-term	0	0	0		0
Shipment					
Unshielded					
Shielded	X	X	X	X	X
Disposal					
Commercial low-level	X	X	X	X	X
Transuranic	0		0		0
High-level treatment/storage	0		0		0

[a]X denotes alternative considered for waste type.
0 denotes alternative applicable to special waste only.

References--Section 2

1. Regulatory Guide 1.86, U.S. Atomic Energy Commission, June 1974.

2. "Plan for Reevaluation of NRC Policy on Decommissioning of Nuclear Facilities," U.S. Nuclear Regulatory Commission, NUREG-0436, Revision 1, December 1978 and Supplement 1, August 1980.

3. "Thoughts on Regulation Changes for Decommissioning," Draft Report, U.S. Nuclear Regulatory Commission, NUREG 0590, Rev. 2, 1980.

4. U.S. Nuclear Regulatory Commission, TMI Program Office Weekly Status Report, for period January 4-10, 1981, dated January 12, 1981.

5. Memorandum from C. Marotta, Office of Nuclear Materials Safety and Safeguards, to K. Kniel, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, "Recriticality Potential of TMI-2," May 14, 1979.

6. R. DiSalvo et al., "A Further Evaluation of the Risk of Recriticality at TMI-2," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, April 14, 1980.

7. G. D. Calkins, "Plan for Reevaluation of NRC Policy on Decommissioning of Nuclear Facilities," NUREG-0436, Rev. 1, Suppl. 1, August 1980.

8. R. I. Smith, G. J. Konzek, W. E. Kennedy, Jr., "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Station," NUREG/CR-0130, June 1978.

9. R. I. Smith and L. M. Polentz, "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," NUREG/CR-0130 Addendum, August 1979.

3. THE ENVIRONMENT AND POPULATION WHICH MAY BE AFFECTED

3.1 LOCATION OF THE SITE AND ACCESS FACILITIES

The TMI nuclear power station is located on the northern end of Three Mile Island, one of a group of islands on York Haven Pond of the Susquehanna River. Harrisburg, the state capitol of Pennsylvania, is about 12 miles northwest, and Baltimore, Maryland, is about 65 miles south of the station (Fig. 3.1).

TMI is in Londonderry Township of Dauphin County (Fig. 3.2), at coordinates 40°9'10" north latitude and 76°43'25" west longitude. Middletown Borough is three miles north of the island and Harrisburg International Airport is a short distance to the west of Middletown; both are on the east bank of the Susquehanna. Goldsboro Borough is about one mile west of the station, along the west riverbank in York County. The surrounding lands are predominantly rural, supporting dairy, poultry, and agricultural uses, as well as forestry. The topography undulates slightly, with maximum relief of about 200 ft.[1,2]

About 200 acres of TMI's 472 acres are occupied by the station, with the facilities for Unit 2 immediately south of Unit 1 (Fig. 3.3). The island is about 11,000 ft long and 1700 ft wide and is aligned north-south, paralleling the east riverbank 900 ft away. The west bank is 6500 ft away. In between are Beech Island, Shelley Island, and several smaller islands.

Three Mile Island is linked near its center to the east riverbank by Red Hill Dam (Fig. 3.3), and York Haven Dam curves southward from the end of the island to the west bank. The island also is connected to the east bank of the mainland dam by two bridges--a permanent bridge at the northern end which is used primarily for Unit 1, and a temporary bridge at the southern end for Unit 2. Access to both bridges is provided by State Highway 441 which parallels the river in this area. A one-track railroad spur crosses the permanent bridge from the rail line, which also runs close to the river on the eastern bank.

3.2 GEOLOGY

The TMI site is within the Gettysburg Basin section of the Piedmont Physiographic Province, which is bounded to the north and west by the Northern Valley and Ridge Province and to the south and east by the Atlantic Coastal Plain. The Gettysburg Basin is one of a series of long, narrow basins of Triassic (225 to 190 million years before present) deposits which extend in broken patches from Connecticut to North Carolina. The portion of the Gettysburg Basin in Pennsylvania is referred to as the Triassic Lowland. North and west of the Triassic Lowland are the folded and thrust-faulted Paleozoic rocks of the Appalachian Mountains. The Piedmont southeast of the Triassic Lowland consists of igneous and metamorphic rocks of the Precambrian to Early Paleozoic Age (more than 600 to 400 million years B.P.).[3]

The topography of the area immediately surrounding the site is slightly undulating, with a maximum relief of about 200 ft and highest elevation seldom above 500 ft. From the east, drainage is primarily by Swatara Creek, which has its mouth near Middletown and flows southwest, and by the more westerly flowing Conewago Creek, which empties into the Susquehanna River at the southern end of the island. Fishing Creek flows into the Susquehanna west of the site, and the northwesterly flowing Conewago Creek terminates at York Haven. Three Mile Island has very little relief, with elevations above sea level ranging from about 280 ft (+280 ft MSL) at the water's edge to slightly more than +300 ft MSL in the north-central portion of the island.

The island is composed of stratified sand and gravel containing varying amounts of silt, clay, and clean sand. Density values range from loose to very dense. Boulders are present at depth and are mainly confined to the lower portions of the soil zone on the northern end of the island. Soil thicknesses vary from about 6 ft at the southern end of the island to a maximum of 30 ft

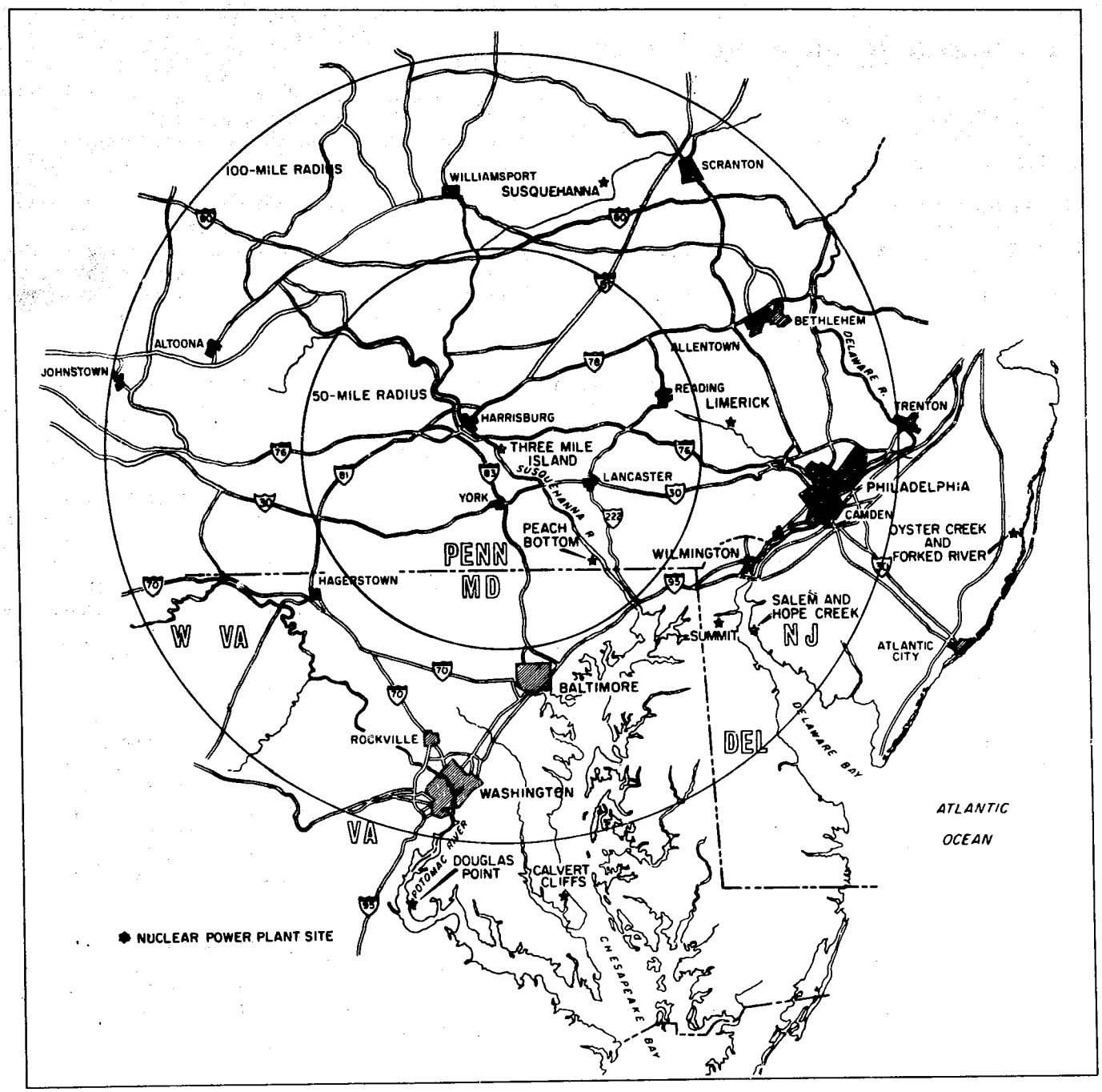


Figure 3.1. Map of the Area within 100 Miles of the TMI Site.

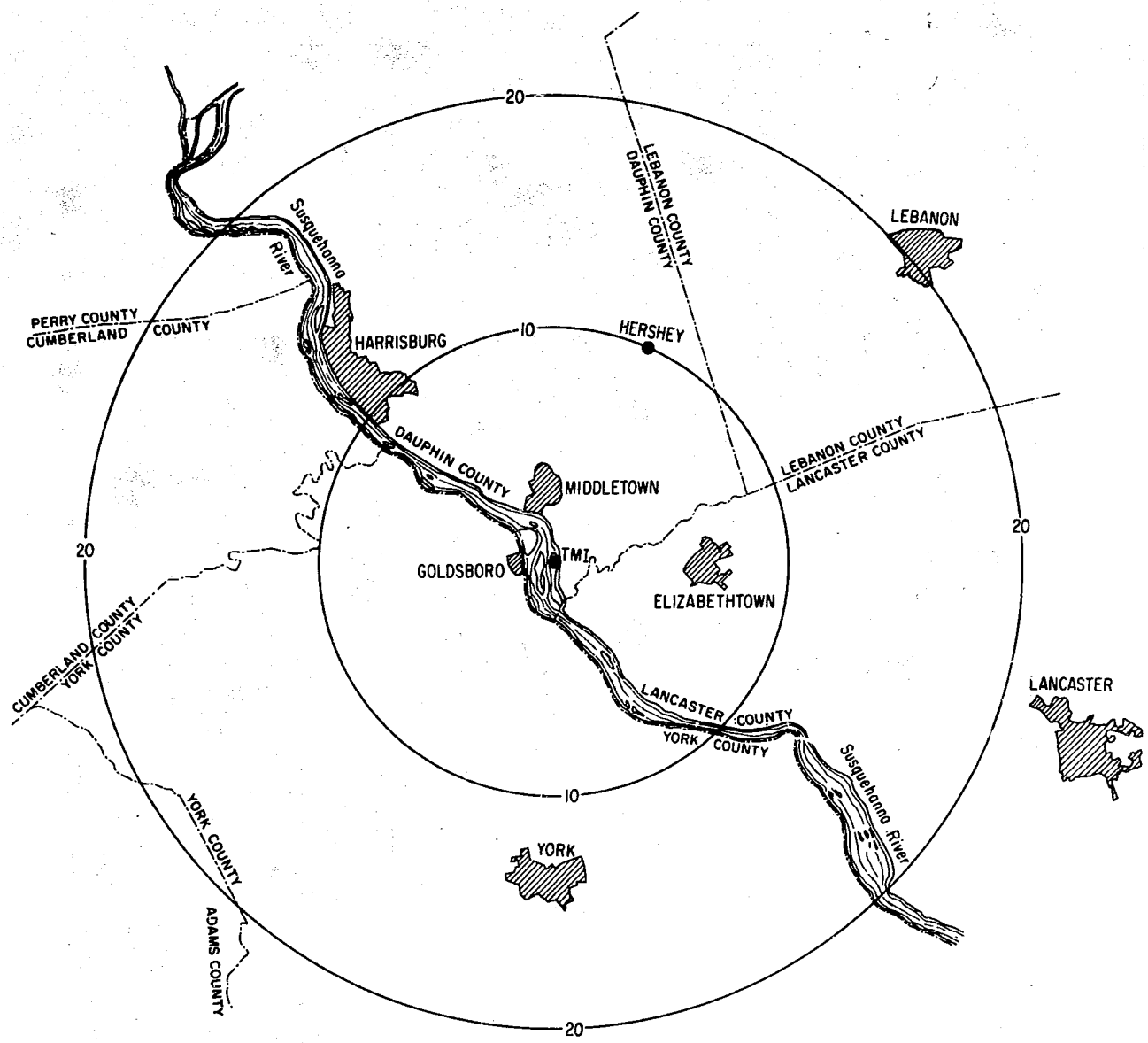


Figure 3.2. Map of the Area within 20 Miles of the TMI Site.

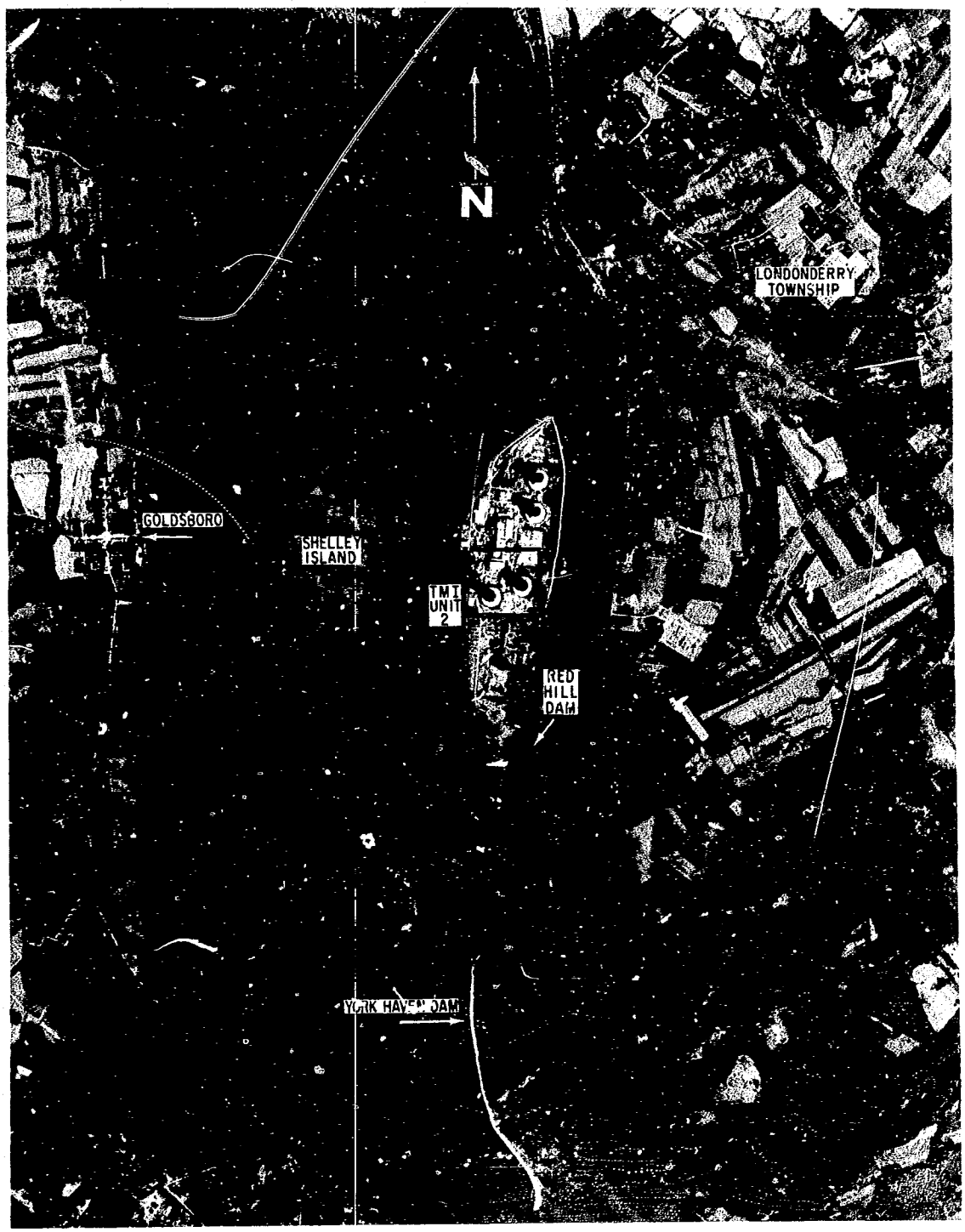


Figure 3.3. Aerial Photo of TMI Site.

near the center of the island. Depth of soil is relatively constant at about 20 ft in the vicinity of the plant site. From one-half to one foot of topsoil, composed of sandy silt with organic material, covers the island. Beneath the soil overburden the site is underlain by the Gettysburg shale. Bedrock surface beneath the site is essentially flat, with an approximate average elevation of +277 ft MSL. The rock consists of red to brown, interbedded, fine- to medium-grained sandstone, shaly siltstone, and shaly claystone. Bedrock ranges from medium-hard to hard, with seismic compressional wave velocities ranging from 8,500 to 11,500 ft/s. The upper 1 to 3 ft of bedrock is weathered. Based on mapping of surface outcrops in the site area, bedrock strata strike from 65°N to 80°E and dip 30° to 70° NW. However, fairly consistent northerly dips of 37.5° to 45° were observed in core borings at the site.[4]

Jointing in bedrock is near vertical and strikes N10°E. Many of the joints are healed, and others which are not have been altered by oxidation. The jointing becomes tighter with depth.[4] Drilling water loss was noted in two of the core borings (DH-1 and DH-4), suggesting the possibility of open fractures, although there are other causes of water loss. The water level of the Susquehanna River controls the groundwater level on TMI and most groundwater moves to the river under the influence of the hydraulic gradient. Contamination of groundwater in the Gettysburg Formation would be highly unlikely since it is confined and under artesian pressure. The Gettysburg Formation is a major source of base flow to the Susquehanna River. Any contamination of this aquifer would flow to the Susquehanna River and would not affect groundwater users.[5]

A healed normal fault was encountered in one boring at a depth of about 95 ft. This minor fault exhibited three inches of normal displacement and a dip of 80° in a northerly direction, the same direction as the bedding. Its age has been interpreted as being Late Triassic (190 million years B.P.).[4] The staff concluded during the Operating License (OL) review that under Appendix A, 10 CFR Part 100, this fault is not capable. Fractures are present in the rock parallel to the fault. These fractures have been completely healed with calcite.

All Category I structures and tanks are founded in bedrock, except the Category I storage tanks, which are founded in compacted backfill. Results of subsurface investigations show that the foundation conditions in sound bedrock underlying the site were adequate for the proposed facility. There were no extraordinary geologic engineering problems associated with construction of TMI-2.[6]

The maximum seismic intensity that the site has experienced in historic times is intensity V on the Modified Mercalli (MM) scale based on the March 8, 1899, earthquake at York, Pennsylvania. There are no capable faults or other geologic structures that could be expected to localize earthquakes in the immediate vicinity of the site. The maximum credible earthquake for this site is the largest earthquake that has occurred in the Piedmont Province, intensity VII (MM). Using the Trifunac-Brady[7] empirical correlation between horizontal ground acceleration and intensity, intensity VII (MM) corresponds to an acceleration of 0.12 g. This value should be used as the high-frequency anchor point for the design response spectra. Thus, the Safe Shutdown Earthquake (SSE) was selected to have a horizontal ground acceleration of 0.12 g. The Operating Basis Earthquake was selected to be one-half of the SSE, or 0.06 g. The staff's seismology consultant, the USGS, concluded that the ground accelerations considered for the site are adequate based on the seismic history of the site and the surrounding area.[8]

3.3 METEOROLOGY

Meteorological conditions presently existing in the vicinity of TMI-2 do not differ significantly from those described in the Final Environmental Report and the Safety Evaluation Reports relative to operation of these units.[1,9] Portions of the previously published descriptions are reprinted below.

3.3.1 Regional Climatology

The climate of southeastern Pennsylvania is primarily continental in character. Although the proximity of Chesapeake and Delaware Bays, and to a lesser extent the Atlantic Ocean, tends to exert a moderating influence on air temperatures over much of the region, these effects are weak as far inland as Harrisburg. Continental polar air originating in Canada is the predominant type of air mass over the region in winter. However, these air masses are usually modified and warmed somewhat as the air descends the eastern slopes of the Appalachians before reaching the southeastern section of Pennsylvania. Maritime tropical air masses originating over the Gulf of Mexico or the Carribbean Sea predominate over this region in summer. Winters are relatively mild for the latitude; summers are warm and humid.[10]

Temperatures of 90°F or higher may be reached on 20 to 25 days annually over the region. Temperatures of 0°F or lower may be expected on only one or two days annually; temperatures of 32°F or lower may be expected on about 108 days per year. Precipitation is generally well distributed throughout the year, but the greatest monthly amounts occur in the summer in association with thundershowers. On an annual basis, relative humidity averages around 70 percent.[10]

3.3.2 Local Meteorology

Long-term weather records from Harrisburg, Pennsylvania, record the extreme maximum and minimum temperatures as 107°F in July 1966 and -14°F in January 1912. Maximum 24-hour precipitation totaled 12.55 inches in June 1972 and maximum 24-hour snowfall totaled 21.0 inches in January 1945.[10] The normal annual snowfall for the Harrisburg area is 37 inches, while freezing precipitation occurs on an average of two to three days per year. Heavy fog (visibility one-quarter of a mile or less) occurs an average of 21 days annually.[10]

Onsite wind data at the 100-ft level (10 m above nearby obstructions) indicate that the predominant wind flow is from the northwest with a frequency of 12.7 percent (Fig. 3.4).

Monitoring of wind speed and direction onsite, as well as atmospheric stability, has been conducted since 1967 and is continuing. The data will allow real-time evaluation of atmospheric relative concentrations (χ/Q) for locations surrounding the site. The highest offsite χ/Q's should lie within a range from 2.3×10^{-6} sec/m^3 (the annual average continuous release χ/Q) to 8.3×10^{-4} sec/m^3 (the 0- to 2-hour χ/Q that would be exceeded 5 percent or less of the time at the exclusion area boundary).

It is possible to place meteorological controls on the cleanup operations in order to optimize atmospheric diffusion of any gaseous effluents released. Diffusion optimization can be accomplished by limiting releases to periods having moderate to strong winds, 10 mph or greater, and a Pasquill-Gifford stability classification of D through A. Although such stability conditions are preferable, whenever winds are 10 mph or greater, the relative concentration offsite would be substantially less than the 0- to 2-hour value given above. Another factor that would aid in optimizing (reducing) possible χ/Q at places where public exposure is likely would be to release the gases when winds are blowing toward the southern end of the island, where no actual nearby receptors are located.

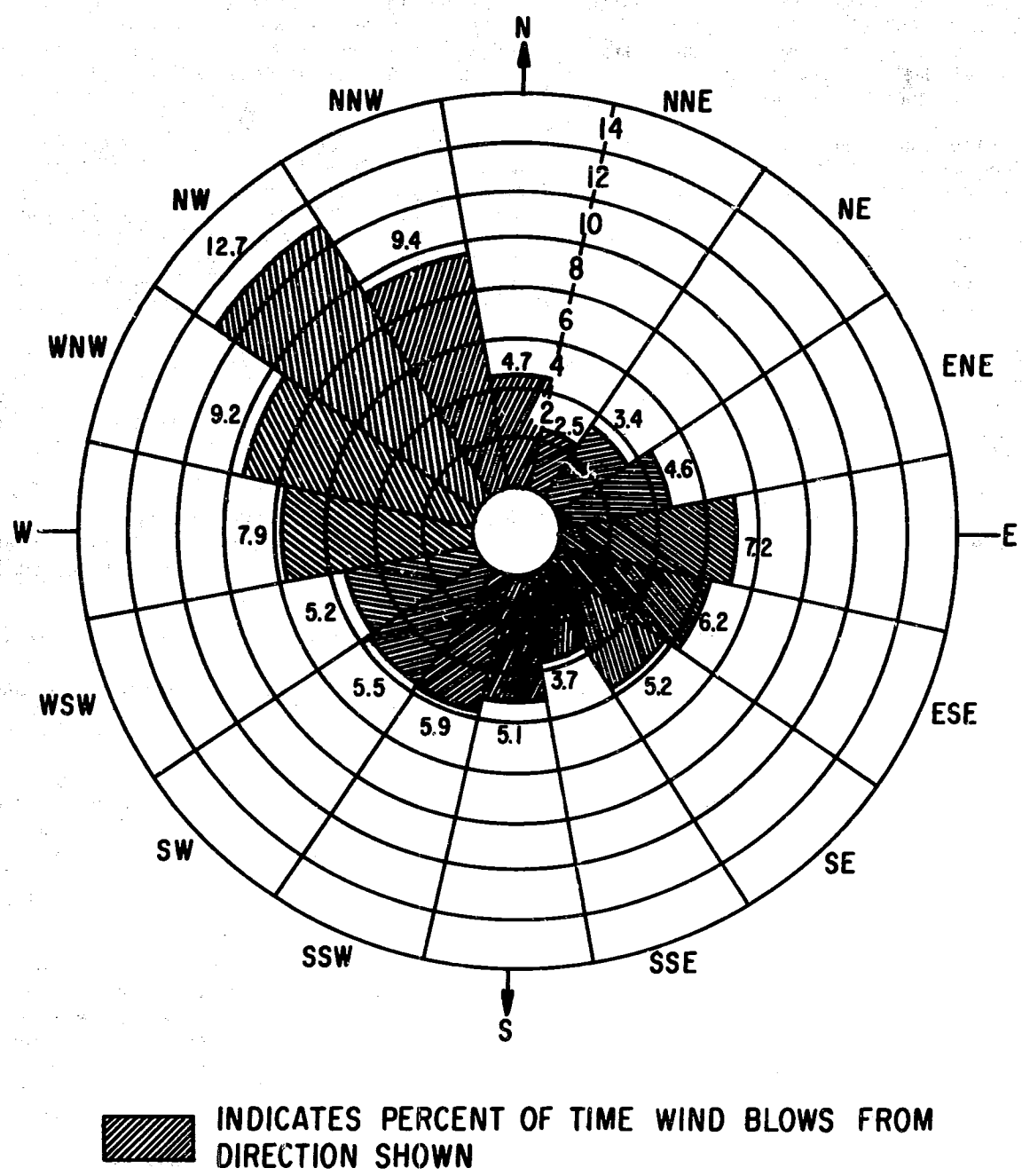
Evaporational methods for the disposal of water from the site were also reviewed by staff meteorologists. Forced evaporation, through addition of heat to holding ponds, may result in fog over the ponds and the river channels adjacent to the island, depending on air temperature, humidity, and wind speed. Natural evaporation of water from the onsite ponds would have a lesser environmental impact than forced evaporation. However, the rate of natural evaporation would be controlled by air motion over the pond and by increasing the water temperature by solar radiation to increase the evaporation rate. In this area of Pennsylvania, the average annual evaporation is within the range of 33 inches (lake evaporation) to 45 inches (evaporation-pan measure),[11] depending on the volume of the body of water. Seventy percent of the evaporation occurs from May through October. Because annual total precipitation in this area is expected to exceed 40 inches, significant net water loss to the atmosphere is not expected from closed ponds.

3.4 HYDROLOGY AND WATER USE

3.4.1 Surface Water Hydrology

The major stream affected by liquid discharges from TMI is the Susquehanna River, which has a total drainage area of 27,400 square miles where it enters the Chesapeake Bay, and of about 25,000 square miles near the TMI site. The TMI plant is about 11 river miles downstream from the Harrisburg gaging station of the USGS. This station has had a continuous period of record since 1890. Data for the Harrisburg gage are assumed to be applicable at the plant site and show the river's highly variable characteristics:[12]

Characteristic	Value, cfs
Minimum daily flow (9-18-64)	1,700
Average annual discharge	34,500
Mean annual flood	260,000
Maximum flood of record (1972)	1,020,000



INDICATES PERCENT OF TIME WIND BLOWS FROM DIRECTION SHOWN

Figure 3.4. TMI Annual Average Wind Rose at 100 ft, 1972-1975 Data.

No large dams or reservoirs exist immediately upstream from the site. The Corps of Engineers has constructed a new dam for flood control on the Raystown Branch of the Juniata River, which enters the Susquehanna about 25 miles upstream of TMI. The project also is used for low-flow augmentation and recreation. For low-flow control, the project increases dependable flow at the Raystown dam site to 480 cfs. This benefit will also be felt at TMI. Whenever possible, a flow of 480 cfs is released from the reservoir, provided that a significant drop in the upstream reservoir level does not occur. However, the Corps plans to release at least 200 cfs at all times to maintain adequate flow downstream.

At TMI the Susquehanna is about 1.5 miles wide. At the southern end of TMI, York Haven Pond on the Susquehanna is impounded by York Haven Dam and Red Hill Dam.[1,13] The river channel is braided and flows around several islands, forming several channels (Fig. 3.5). The western channel is shallow and conducts about two-thirds of the normal flow;[14] the eastern channel, approximately 5.5 ft deep, conducts a minor portion of the flow, which overtops Red Hill Dam part of the year.[14] The central channel is about 10 ft deep under normal flow conditions and joins the western channel to spill over the central portion of York Haven Dam. Some flow is diverted along the dam to the York Haven Generating Station, a hydroelectric facility.

Downstream from York Haven Dam, the Susquehanna is impounded by three dams at Safe Harbor, Holtwood, and Conowingo before entering Chesapeake Bay (Fig. 3.6). Just downstream from the Holtwood Dam is Muddy Run, a pumped storage facility capable of altering river discharge by removing water for storage or releasing stored water to rejoin the river discharge. The Muddy Run removal rate is 24,500 cfs and the release rate is 32,000 cfs.[15,16]

At the head of Chesapeake Bay, water from the Susquehanna River mixes with the discharge of other tributaries in a wide, shallow area. From the head to the mouth, Chesapeake Bay is about 195 miles long and 3 to 35 miles wide, with a mean depth of about 30 ft, although a deep channel (120 ft) is present. The complex water circulation and salinity patterns are influenced by the tides, tributary river discharges, and weather.[17]

Low flow studies of the Susquehanna River have been conducted using data from the Harrisburg (upstream of TMI) and Marietta (downstream) gaging stations. Table 3.1 contains flow-duration data for the Susquehanna River at Harrisburg, based upon daily flows. These data indicate the frequency of flows for given durations. For example, a flow of 5000 cfs or less, lasting 183 days, could be expected about once in 20 years.

The minimum flow of 1600 cfs occurred on November 29, 1930, as the result of an ice blockage of the river. However, the minimum daily discharge since the construction of a dam downstream from the Harrisburg water filtration plant was 1700 cfs on September 18, 1964. The percentage of the time that the flow is equal to or less than specified discharge is summarized in Table 3.2.

The mean monthly flows for the period 1891-1979 are shown in Table 3.3. These values illustrate the monthly variation in average flows.

The amount of river dilution received by liquid waste from the TMI plant depends primarily upon the river flow at the time of discharge. The maximum capacity of the turbines at York Haven Dam is about 16,000 cfs. During periods of low flow when the river discharge is less than 16,000 cfs, essentially all of the river flow will pass through the generating units. Therefore, any discharged wastewater downstream of the dam will be fully mixed with the river after it passes through the turbines. For higher river flows, some flow will also pass over the dam. This case is much less critical for the purposes of computing concentrations of contaminants. However, mixing will occur downstream aided by the turbulent flow over the dam. The flow of the Susquehanna is split into several segments at the head of TMI and Shelley Island. The amount of water flowing past the discharge point (in the middle channel between TMI and Shelley Island) is about 25 percent of the total river flow. Below York Haven Dam, additional mixing occurs and the full flow of the river may be used in determining dilution factors.

For steady flow in the river, dilution of a contaminant released from the plant is approximately inversely proportional to the flow rates. For the purposes of estimating an average concentration in the Susquehanna River for a given release, a "plug flow" model with no longitudinal mixing is used. The correct flowrate for this model is not the "arithmetic mean flow," $\bar{Q}$, but the "reciprocal mean" flow, $(\overline{1/Q})^{-1}$. The arithmetic mean flow is heavily weighted by flood events, while the reciprocal mean flow is weighted in favor of low flows, for which concentrations would be higher. The reciprocal mean flow has been calculated for flow measurements at Harrisburg to be about 12,000 cfs. The reciprocal mean is conservative because deviations from

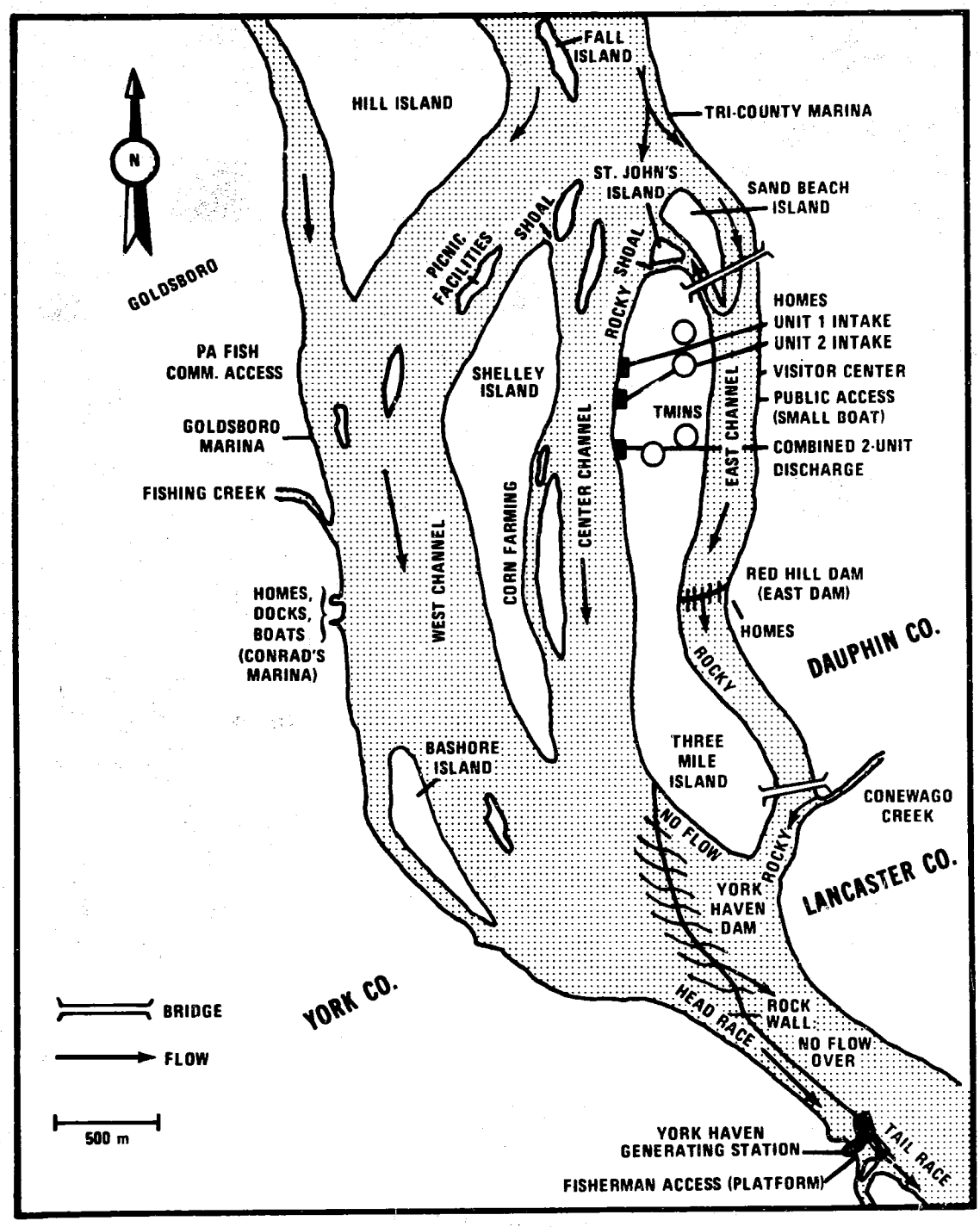


Figure 3.5. York Haven Pond of the Susquehanna River, Showing the Flow Patterns in the Channels and over the Dam at Three Mile Island.

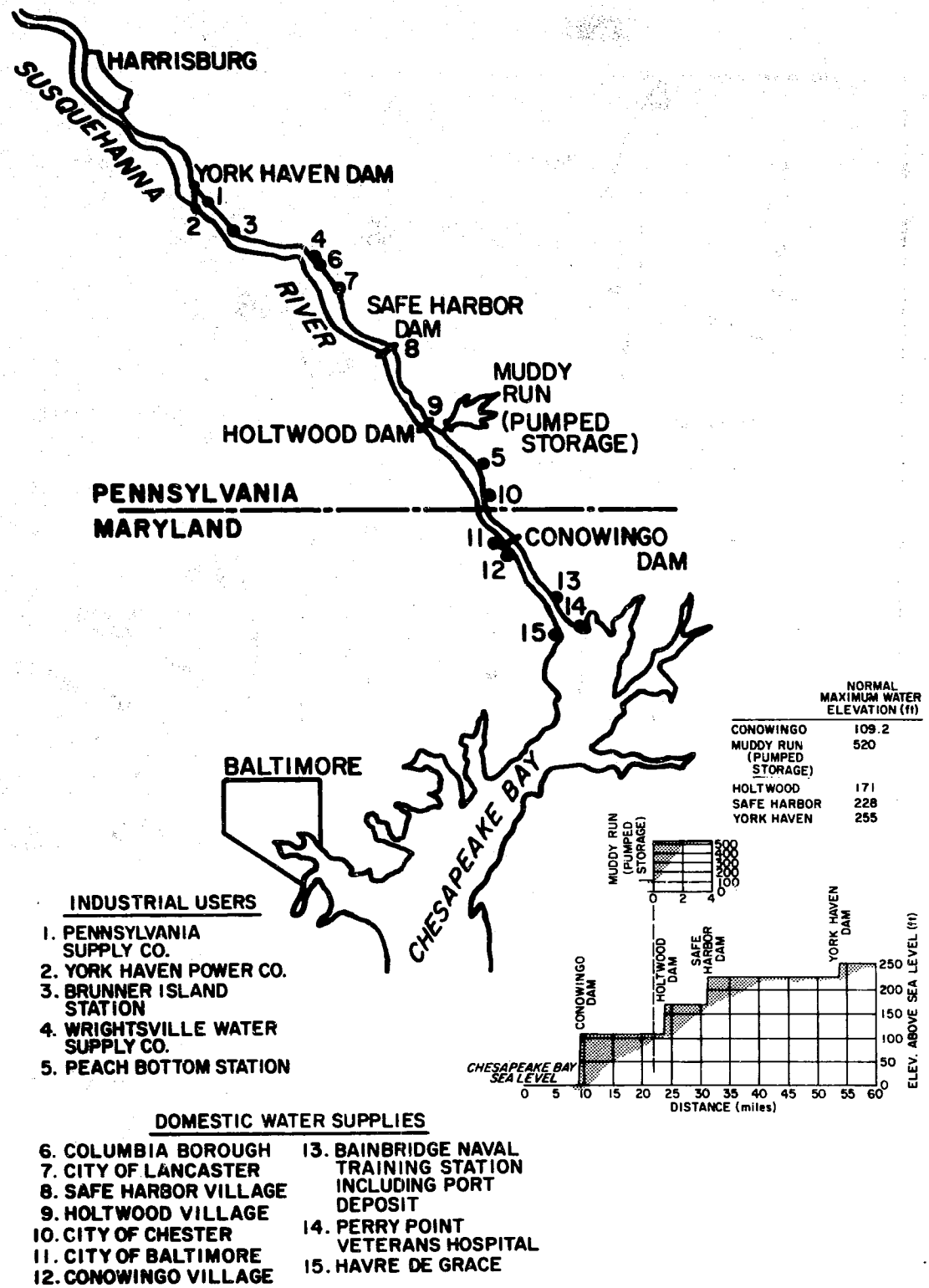


Figure 3.6. Major Inputs and Outputs of the Susquehanna Showing Principal Water Users below the TMI Plant. (Modified from "Final Environmental Impact Statement Related to Operation of the Fulton Generating Station, Units 1 and 2," U.S. NRC Docket Nos. 50-317 and 50-318, 1973.)

Table 3.1. Magnitude and Frequency of Low Flow, 1892-1972[a]

Period of Consecutive Days	Discharge (ft^3/s) for Indicated Recurrence Interval in Years					
	2	5	10	20	30	50
7	3,900	2,800	2,400	2,200	2,100	1,900
14	4,200	3,000	2,600	2,300	2,100	2,000
30	4,900	3,400	2,900	2,600	2,400	2,200
60	6,300	4,000	3,300	2,800	2,600	2,400
120	9,000	5,400	4,400	3,600	3,200	2,900
183	13,000	8,100	6,300	5,000	4,400	3,800

[a]From L.V. Page and L.C. Shaw, "Low Flow Characteristics of Pennsylvania Streams," Commonwealth of Pennsylvania, Department of Environmental Resources, 1977.

Table 3.2. Percentage of the Time that the Susquehanna River Flow at Harrisburg Equals or Is Less than Specified Values[a]

Flow, cfs	Percent of Time
2,000	Negligible
3,200	2
4,000	5
5,200	10
7,600	20
11,000	30
15,000	40
19,000	50
35,000	70
50,000	80
79,000	90
110,000	95

[a]From L.V. Page and L.C. Shaw, "Low Flow Characteristics of Pennsylvania Streams," Commonwealth of Pennsylvania, Department of Environmental Resources, 1977.

Table 3.3. Mean Monthly Flows of the Susquehanna River at Harrisburg for the Period 1891-1979

Month	Mean Flow (cfs)
January	37,700
February	40,100
March	77,700
April	82,600
May	45,700
June	26,300
July	15,200
August	11,700
September	11,700
October	17,300
November	25,900
December	33,300

the simple plug flow model, such as reservoir storage and longitudinal mixing, tend to desensitize the average concentration to very low flows, which essentially increases the effective mean flow.

3.4.2 Surface Water Uses

The surface water of the Susquehanna River downstream from Harrisburg is acceptable for all general uses, e.g., supporting aquatic life, recreation, and primary contact. It is not an attractive source of public water supply because of occasional high sulfate levels and high amounts of wastewater-derived coliform bacteria.[18] Below Harrisburg, late summer algal blooms occur that are indicative of high nutrient levels, primarily phosphates and nitrates.[19] This is attributable both to wastewater treatment and runoff from agricultural areas.

The river and the streams in the vicinity of TMI presently are used for water supplies, both public and industrial; power generation; boating; sport fishing; and recreation. Sport fishing is done in all streams in the general area of the site; however, there is no commercial fishing. Brunner Island steam-electric generating station, five miles downstream, is the nearest user of some of the river as potable water. The locations of downstream surface water users are indicated in Figure 3.6.

The available information on water supplies downstream from TMI is summarized below; the consumers include public water supplies, industries, and utilities:

- The Pennsylvania Supply Company takes 2200 gpm per nine-hour work day from the mouth of Conewago Creek (near the southern tip of TMI) for sand and gravel processing. The water is released into the Susquehanna River after clarification.

- York Haven Power Company (a wholly owned subsidiary of Met-Ed) owns a hydroelectric generating station at York Haven with a total installed capacity of 20,000 kWe. The station is about two miles downstream of the TMI discharge structure. A dam across the Susquehanna River impounds 8000 acre-ft of water for power generation, air conditioning, cooling, and fire protection. Potable water is obtained from two wells.

- PP&L owns and operates Brunner Island, a 1415-MWe-capacity steam-electric generating station on the west shore of the Susquehanna River five miles downstream from TMI. The station utilizes 1155 cfs of river water for the circulating water system and ash removal; the water is returned to the river without reconcentration. River water for potable and cycle makeup demineralizer requirements is treated prior to use.

- The Wrightsville Water Supply Company has a public water supply intake on the Susquehanna River 16.2 miles downstream from the nuclear station. The intake is utilized as a summer reserve supply, with treatment facilities provided in the system.

- The borough of Columbia takes an average supply of two million gallons per day from the Susquehanna River by an intake located 16.7 miles downstream from TMI. The water is treated prior to distribution.

- The city of Lancaster withdraws an average of 8 million gallons per day from the Susquehanna by an intake located about 17 miles downstream from TMI. The water is treated prior to distribution. Additional water is taken from the Conestoga River.

- The Safe Harbor Water Power Corporation owns and operates a hydroelectric generating station on the Susquehanna River about 27.2 miles downstream from TMI. The dam across the river impounds 92,000 acre-ft for power generation. Water for a public water supply system for the village of Safe Harbor is withdrawn from the reservoir at a rate of 25,000 gallons per day and receives complete treatment prior to distribution.

- A public water supply system serves the village of Holtwood on the eastern side of the Susquehanna River 34.7 miles downstream from TMI at the Holtwood Hydroelectric Station. About 22,000 gallons per day is withdrawn from the 19,300 acre-ft Holtwood Reservoir and treated prior to distribution.

- The Muddy Run pumped storage generating station is 38 miles downstream from the TMI nuclear station. It is a remotely controlled 800-MWe-capacity station, operating between an upper reservoir and Conowingo Reservoir. River water is used for fire protection. Potable water is supplied by deep wells.

- Peach Bottom Nuclear Generating Station is on the west bank of the Susquehanna River about 41 miles downstream from TMI. Unit 1, rated at 40 MWe, was part of the Atomic Energy Commission's Power Reactor Demonstration Program and has been decommissioned. Two additional nuclear units, each rated at 1000 MWe, are now operating and withdraw a total of 3450 cfs from the river for cooling purposes.

- The city of Baltimore has an intake on the river which draws water from Conowingo Reservoir about 49 miles downstream from TMI. Baltimore is permitted to withdraw up to 250 million gallons per day when the river flow exceeds 5000 cfs, but is limited to 65 million gallons per day at lower river flow. Two water-treatment plants provide treatment prior to distribution. The last significant withdrawals occurred in 1968; future use of potable water from the Susquehanna is expected to be limited to periods of severe drought conditions which necessitate the use of alternative supplies such as the Susquehanna.

- Philadelphia Electric Company owns and operates a hydroelectric generating station at Conowingo Dam, which impounds 321,500 acre-ft in Conowingo Reservoir. The generating station has a capacity of 513,000 kWe; it is located 50 miles downstream from TMI. Potable water for the station and for Conowingo Village is obtained from the reservoir. Treatment is provided to an average supply of 12,000 gallons per day.

- Arundel Sand and Gravel Co. has been issued a permit to withdraw 900,000 gallons per day for industrial use from intakes located about four miles upstream from Havre de Grace in the vicinity of Susquehanna Park.

- The city of Havre de Grace is permitted to withdraw about 12.5 million gallons per day from intakes located on the west side of the river near its mouth. About 1.8 million gallons per day are currently used for potable supplies and are treated prior to distribution. Future plans are to expand water supplies to other areas of Harford County.

- Bainbridge Naval Training Station and the town of Port Deposit have been withdrawing about 225,000 gallons per day for potable supplies from intakes located on the east side of the river at Port Deposit. The water receives treatment prior to distribution.

- Perry Point Veterans Hospital and the town of Perryville withdraw about 400,000 gallons per day from intakes located on the east side of the river. The water is treated before use.

3.4.3 Surface Water Quality

3.4.3.1 Chemical and Bacteriological

Water quality data have been given in monitoring records provided by the licensee over the period 1967-1976 for CP and OL licensing actions,[19] and in USGS Water Resource Data for water year 1977 (October 1976 through September 1977).[13] Water quality data for the Susquehanna River near TMI and at three other locations are presented in Table 3.4. The important characteristics are a moderately high total hardness, averaging about 130 mg/L as $CaCO_3$ (typically due to about 35 mg/L of calcium and 10 mg/L of magnesium), and a high and variable sulfate concentration, averaging about 80 mg/L, but occasionally exceeding 200 mg/L. The sulfate limit set by the revised state water quality criteria is 250 mg/L.[20] The sulfate contamination is largely attributable to acid drainage from old coal mines within the watershed, which also contribute a high and variable total iron concentration, ranging from 0.4 to 5 mg/L, often in excess of the state limit of 1.5 mg/L.

The high sulfate concentration results in a relatively low alkalinity, averaging about 60 mg/L of $CaCO_3$. As a result, about half of the hardness is noncarbonate or permanent (not deposited as scale on heating), but the high total hardness would severely impair the detergent action of conventional soaps.

Oxygen concentrations, water temperature, and sediment loads of the river have been altered by the construction of impoundments. The impounded pools allow suspended sediment to settle out and increase the surface area exposed to solar radiation. Increased retention time within the impoundments allows increased solar exposure and subsequent heat gain. Temperatures recorded at Harrisburg during water year 1977 ranged from 37.4°F to 85.1°F. Discharge over spillways and through hydroelectric facilities increases turbulence, providing greater oxygenation of the water.

The fecal coliform count, attributed to domestic and agricultural waste, often is in excess of the state limit of 2000 colonies per 100 mL (mean of five consecutive days).[19]

Water quality throughout the Chesapeake Bay is generally adequate for supporting aquatic life and recreation. In some areas, poor tributary water quality causes local degradation and limited use. Chesapeake Bay is an estuary, a dynamic physicochemical system in which freshwater from tributary rivers mixes with seawater. The water quality at any location in the Bay is a function of the relative contributions from and mixing of the different water masses (Table 3.4). Generally, salinity depends on the amount of freshwater discharged, precipitation, and evaporation. In Chesapeake Bay salinity is lower in spring and summer and higher in autumn and winter.[21] Tributary discharge and tides affect the intrusion of seawater toward the head of the Bay. Because of the tides and river discharges, the Bay is constantly receiving nutrients and salts, the concentrations of which vary over time and area.

3.4.3.2 Radioactivity

The USGS records[13] include measurements of radioactivity, both dissolved and suspended, at Harrisburg. The tritium concentration was measured during the 1977 water year and found to be fairly constant, 178 pCi/L. Gross beta activity was measured on November 8, 1976, and reported as follows:

Dissolved gross beta:	2.4 pCi/L as Cs-137
	1.9 pCi/L as Sr-90/Y-90
Suspended gross beta:	0.4 pCi/L as Cs-137
	<0.4 pCi/L as Sr-90/Y-90

The method of reporting suggests that these are direct beta counting results using cesium and strontium standards, and not specific radiochemical measurements of these radionuclides.

Table 3.4. Water Quality of the Susquehanna River at Several Locations[a]

Parameter	Susquehanna River			Chesapeake Bay[d]
	Near TMI[b]	Holtwood Dam[c]	Conowingo Pond[c]	
General Characteristics				
pH (units)	6.5 - 8.2[e]	6 - 8	6.6 - 7.9	7.3 - 8.4
BOD	1.0 - 5.4	0 - 4.4		1.7 - 7.4
DO	5.1 - 15.2		3.8 - 9.2	5.4 - 9.9
TDS	78 - 397		167	
Alkalinity (as $CaCO_3$)	23 - 172	-	-	-
Hardness (as $CaCO_3$)	46 - 242	-	-	-
Specific conductance, μmhos	77 - 356			
Coliforms (cols/100 mL)	15 - 21,000[e]			
Cations				
Sodium	2.3 - 52.9[e]	8 - 17	2.6 - 10.8	3200 - 4700
Potassium	1.3 - 2.4	2.1 - 3.3	1.3 - 6.1	115 - 150
Magnesium	3.5 - 11	7.2 - 18	2.4 - 22	375 - 550
Calcium	12 - 37	21 - 49	13.5 - 44	130 - 190
Iron	0.01 - 0.17	0.2 - 0.6	0.0 - 0.2	0.27 - 1.9
Anions				
Chloride	5.7 - 20[e]	10 - 22	4.1 - 20	5800 - 7800
Sulfate	14 - 204[e]	70 - 205	30 - 130	770 - 1150
Phosphate	-	0.06 - 1.1	0.0 - 0.87	0.02 - 0.14
Radiological[f]				
Dissolved gross α as U_{nat}, μg/L	<1.6	-	-	-
Suspended gross α as U_{nat}, μg/L	0.7	-	-	-
Dissolved gross β as Cs-137, pCi/L	2.4	-	-	-
Suspended gross β as Cs-137, pCi/L	0.4	-	-	-
Dissolved uranium, μg/L	0.06	-	-	-

[a]The two values presented are the maximum and minimum in mg/L unless otherwise stated.

[b]From "Water Resources Data for Pennsylvania, Water Year 1977," Vol. 2, "Susquehanna and Potomac River Basins," U.S. Geological Survey, Water Data Report PA-77-2, Harrisburg, Pa., 1978.

[c]From "Final Environmental Statement Related to Operation of the Fulton Generating Station, Units 1 and 2," U.S. Nuclear Regulatory Commission, NUREG-75/033, 1975.

[d]From "Final Environmental Statement Related to Operation of the Calvert Cliffs Nuclear Power Plant, Units 1 and 2," U.S. Atomic Energy Commission, Docket Nos. 50-317 and 50-318, 1973.

[e]From "Final Supplement to the Final Environmental Statement Related to Operation of the Three Mile Island Nuclear Station, Unit 2," U.S. Nuclear Regulatory Commission, NUREG-0112, Docket No. 50-320, December 1976.

[f]November 8, 1976, from Footnote b above.

Radium-226 was measured on the same date by the radon method as 0.08 pCi (alpha) per liter. Gross alpha activity on the same date is reported as:

Gross dissolved alpha: <1.6 μg/L as natural U (<1.08 pCi/L)

Gross suspended alpha: 0.7 μg/L as natural U (0.5 pCi/L)

The conversions to pCi(α)/L assume the specific alpha activity of chemically pure natural uranium (U-234, U-235, and U-238 in their natural abundances) to be 0.6735 pCi/μg (approximately 1.5 alpha disintegrations per minute per microgram). A measurement of uranium concentration, presumably by the chemical (fluorimetric) method, also made on November 8, 1976, gave a value of 0.06 μg/L. This is presumably included in the dissolved gross alpha, but it would contribute only about 0.04 pCi/L, assuming natural composition.

The following limits for radioactivity in the Susquehanna River are proposed by the revised State of Pennsylvania criteria:[19]

Total alpha: Not to exceed 3 pCi/L over natural background

Total beta: Not to exceed 1000 pCi/L over natural background

Tritium: Not specified, but may be included in total beta

It is difficult to define a strictly natural background for fission products and tritium, because of small but significant contributions from nuclear weapons testing, which depend on latitude. The contribution from the commercial nuclear fuel cycle is negligible. The radioactivity observed in the Susquehanna River at Harrisburg during 1977 is below the level regarded as normal for this latitude zone.*

3.4.4 Groundwater Hydrology

The site has a water table elevation of about 280 ft MSL, depending upon the Susquehanna River stage, which is normally at 277 ft MSL. Site borings and observation wells indicate that water table elevations vary about 5 ft from a high at the island's center to the shores. The water table gradient is about 0.006 toward the river. The nearest potable water supplies are three wells located on the east bank of the Susquehanna River directly across from TMI. All of these wells have groundwater elevations above the river and above the groundwater level at TMI. Since they are upgradient, these wells are not affected by site activities.

As stated in Section 3.2 the site is underlain by sandy silts, sands, gravels, weathered bedrock, and hard siltstone (Gettysburg Formation). The Gettysburg Formation has basic artesian characteristics in the site area. Groundwater flow is highly anisotropic along the strike direction, with specific capacities ranging from 0.33 to 15.0 gpm per foot of drawdown. The leakage of groundwater from the Gettysburg Formation would be expected to be upward but would vary considerably with the degree of jointing and relationship to strike direction. Therefore, effluents released accidentally from the plant should not migrate into the Gettysburg Formation.[5]

Measures have been taken by the licensee to monitor the quality of groundwater at TMI. At the request of the NRC staff, the licensee installed a series of eight monitoring wells and, subsequently, nine observation wells at the TMI site. These wells were sited so as to detect leakage of contaminated water from the Unit 2 reactor and auxiliary buildings and outside storage tanks. The monitoring wells have been sampled at about one-week intervals since installation in late January 1980. The observation wells, installed to clarify anomalous data from some of the monitoring wells, also have been sampled at about one-week intervals since completion in late April 1980. Weekly sampling of all wells is continuing. The samples have been tested for radioactivity and chemical contamination. To check for errors, the analyses were done by several different laboratories.

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*In Ref. 22, the average radioactivity levels in surface water in the Chicago area are given as: alpha, 0.1-3 pCi/L; beta, 5-10 pCi/L. In Ref. 23, an average tritium level of 287 pCi/L in surface water is cited for latitudes 30°-50°N.

Initial analyses of water samples from some monitoring wells showed concentrations of tritium above normally occurring background concentrations in the vicinity of major plant structures. However, these tritium levels are all less than the Maximum Permissible Concentration (MPC) of 3×10^{-3} μCi/mL, as specified in 10 CFR Part 20, Appendix B, for unrestricted locations.

Concentrations of other radionuclides also were found to be below MPC for unrestricted areas.

The highest tritium levels occurred at wells that were near the Unit 2 borated water storage tank (BWST). The fittings and valving appurtenant to the BWST have leaked onto the immediately surrounding ground surface. At the request of the NRC staff, additional observation wells were drilled to confirm that the BWST was the source of contamination. Soil samples were taken near the BWST during the drilling of the observation wells. The concentrations of various radionuclides were measured at various depths in the soil column, from ground level to a depth of over 25 ft. The tests on these soil samples show higher concentrations of tritium at locations above the water table and close to the BWST. Infiltration of precipitation has carried the contamination downward to the water table, through which it has been transported to other locations. Based on the well data, soil samples, and the fact that the BWST has leaked, the analysis of the test data supports the hypothesis of the BWST leakage through the fittings as the source of contamination.

If the monitoring program indicated that significant concentrations of nuclides were present in the groundwater, there would be several methods to prevent additional contamination of the site groundwater and the Susquehanna River. Examples include:

- The monitor wells could be pumped so as to remove the contaminated groundwater. The wells are located to permit a large volume of groundwater to be pumped.

- A slurry wall of bentonite or other relatively impermeable material could be installed to surround the contaminated area. The wall would inhibit the passage of groundwater both to and from the site. A minimum of about a year's time would be available to install the wall, because it would take the contaminated groundwater at least this long to migrate to the Susquehanna River. (See Appendix V for concentration estimates in the event of an accidental release.)

3.5 ECOLOGY

3.5.1 Aquatic*

The biota of the Susquehanna River include organisms usually associated with flowing waters and (because of the impoundments) with standing waters.

Aquatic macrophytes are rare because of fluctuating flows and water levels and the type of river bottom substrates, which in most of the free-flowing areas are sand or rock. Most macrophytes are found in the fluctuating littoral zone of the impoundments.[16] A dominant source of primary production are algae. The annual cycle of algal succession follows a pattern indicative of a mesotrophic to eutrophic aquatic system: a spring bloom of diatoms, a summer abundance of green algae, and a late summer/early fall increase in blue-green algae and flagellates.[16,24] This pattern is also representative of algal succession in a lake and indicates the importance of the impoundments in the trophic structure of the river.

Zooplankton composition and abundance are variable. The dominant groups are rotifers (Brachionus sp.), cladocerans (Bosmina sp.), and copepods (Cyclops sp.).[19] The periodic high abundance of rotifers also suggests excessive domestic waste loadings to the river.

The composition and abundance of the benthic invertebrate communities are influenced strongly by the quantity and quality of available substrate. Fluctuating flows and impoundments that act as settling basins control the deposition of sand, silt, and finer-sized particles in the various sections of the river. The finer particles tend to settle in the impoundments; the larger, heavier particles settle in the flowing sections. The available sediments are primarily sand, with varying amounts of silt, clay, and organic matter. The most abundant benthic invertebrates are tubificid worms and insect larvae.[16,24]

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*For selected details on York Haven Pond, Conowingo Pond, and upper Chesapeake Bay relative to this environmental impact statement, see Appendix E.

The fish community can be characterized as a warm water assemblage, and is dominated by members of the minnow, perch, and sunfish families.[1,15-17,19] The lower portion of the river (below the Conowingo Dam) receives spawning migrations of some anadromous species, primarily members of the herring family and striped bass. There is an active sport fishery throughout the river. Major species sought are crappie, bass, and sunfish.[14,16,19,24]

In the shallow water of upper Chesapeake Bay, aquatic macrophytes are abundant, and plants such as cord grass (Spartina sp.) and wild celery (Vallisneria sp.) are quite productive as well as an attractive food source for waterfowl. Planktonic algae, primarily diatoms, dominate throughout the Bay, although green, blue-green, red, and brown algae also are present.[17]

The invertebrate fauna is diverse and includes a gradation from freshwater to marine types, depending on the salinity and bottom substrate. Important commercial fisheries are oysters, clams, and bluecrabs.[17,25] In 1976, the harvest of important shellfish exceeded 11,000 tons.[25]

The fish fauna of the Bay also is diverse, and dominant species change with the season and migratory patterns. The commercial fishery of the Bay is important to the Bay states; major species harvested from Maryland waters include menhaden, striped bass, and bluefish.[15]

3.5.2 Terrestrial

Land use and major vegetation types within a one-mile radius of the plant are shown in Figure 3.7. The northern two-thirds of Three Mile Island is occupied by the plant and the periphery of most of the northern portion is forested. Only the eastern half of the lower, third and the western periphery of this lower third are forested; the remainder of this portion of the island is mostly covered by grasses and low shrubs. The plant community is less than 80 years old and of a type common to the area. A 400-ft-wide bulldozed strip parallels the construction road that runs from the plant to the temporary access bridge at the southern end of the island. The nearby islands are similarly forested, except for Shelley Island, which is partly used for agriculture.[1,19,26]

Downriver,[16] the riparian, bottomland and wetland vegetation are similar to vegetation in comparable habitats in the vicinity of Three Mile Island.[1,19,24,26]

The composition of the forest is indicative of a stage in the succession of a floodplain sere between the cottonwood-willow and oak-hickory stages. The majority of the forest land on Three Mile Island can be classified as bottomland hardwood forest, stream terrace hardwood forest, or black locust forest.

The distribution of terrestrial habitats is shown in Figure 3.7. The bottomland hardwood forests, about 230 acres, are in low-lying areas where flooding from the Susquehanna River has been frequent. Silver maple, ash, and river birch are the most common overstory species; they are found in association with American elm, sycamore, catalpa, cottonwood, and tulip poplar.

The stream hardwood forests are found on the higher bottoms and stream terraces. About 90 acres of this forest type were reported. Red oak is the most abundant species, although one stand had chestnut oak and beech as dominant overstory species. Species occurring less frequently include black cherry, black locust, black walnut, mockernut hickory, silver maple, tulip poplar, and ash.

The black locust forest is found primarily in a woodlot of about 50 acres on the southeastern portion of Three Mile Island. This is a relatively young second-growth forest. In some areas black cherry and black walnut occur with black locust as the dominant species; in others, black locust alone is dominant. Less frequently occurring species include sassafras, cottonwood, ash, staghorn sumac, and box elder.

The miscellaneous areas include a very old 3-acre stand of beech forest on Beech Island, hedgerows and windbreaks planted between agricultural fields, and a double row of red pine bordering a cart road on Shelley Island.

Of the 212 species of terrestrial vertebrates found in the TMI vicinity, there were 179 birds, 19 mammals, 8 reptiles, and 6 amphibians (Ref. 11, Tables 2.7-5 through 2.7-8). Small-game animals were eastern cottontail rabbit and gray squirrel. Mammalian predators were longtail weasel and red fox. The largest mammal on the site was the white-tail deer. Four species of upland game bird were found onsite: ring-necked pheasant, American woodcock, mourning dove, and rock dove. Whistling swan, Canada goose, nine species of dabbling duck, seven species of diving duck, and three species of mergansers also were reported. This sampling is typical of the fauna

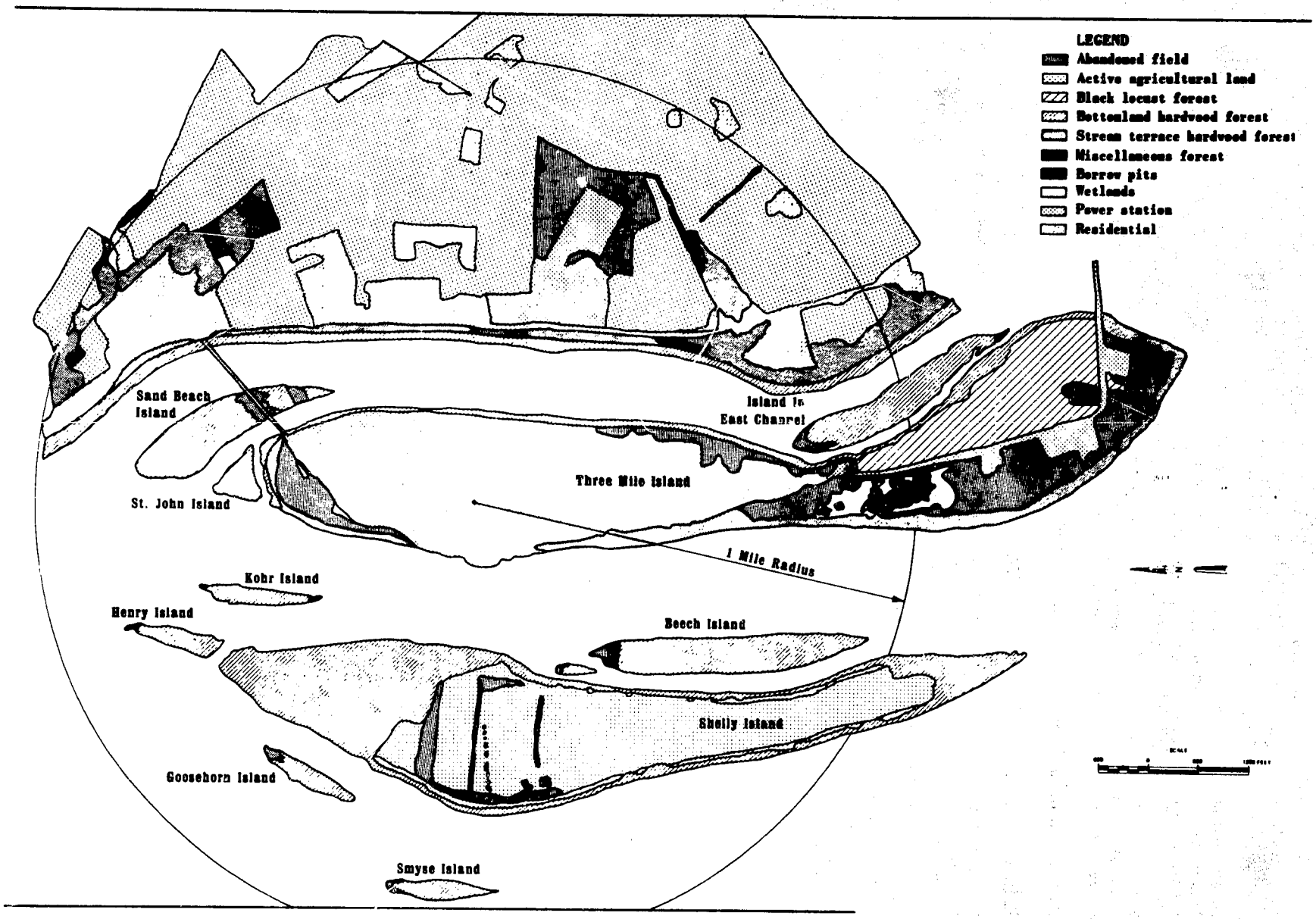

Figure 3.7. Major Types of Terrestrial Communities within One Mile of the Three Mile Island Station.

found downriver.[16] Because the Susquehanna River is a major flyway, large numbers and many species of migratory and resident waterfowl nest and feed on the ponds and reservoirs along the river.

No endangered species are known to occur on the plant site.[11,27] The site lies within the ranges of occurrence of three endangered species--southern bald eagle (Haliaetus leucocephalus), peregrine falcon (Falco peregrinus), and Indiana bat (Myotis sodalis)--and it is possible that individuals could visit the site, particularly during periods of migration. Several American osprey, a species designated as of undetermined status, have been seen onsite, but no nesting activity has been observed.[11] The bald eagle and osprey, which are likely to take food from the river, can be seen downriver, as well as at the site.

3.6 POPULATION, LAND USE, AND OTHER SOCIOECONOMIC CONSIDERATIONS

3.6.1 Population Groups and Geographic Areas of Interest

The staff has identified four population groups and geographic areas that have the potential for being affected by decontamination activities at TMI-2. The area within 15 miles of TMI, hereafter referred to as the "vicinity" of TMI, is expected to experience such impacts as are generated by in-moving technicians, traffic to and from the site, and various cleanup activities. The second subarea subject to potential impact is the Susquehanna River and those who use the river and adjacent lands for recreational purposes. The Chesapeake Bay is the third subarea identified by the staff. Potential impacts to the Bay would affect those who make their living from the Bay's resources, those who use the Bay for recreation, and those who consume Bay shellfish and finfish in marketing areas across the nation. Finally, the transportation of waste from TMI could impact people and areas adjacent to the selected transportation routes. For illustrative purposes, the analysis in Section 9 features two feasible routes having extremes in distance traveled; these routes are to the commercial burial site at Richland, Washington, a distance of about 2750 miles, and to the waste management facilities at West Valley, New York, 370 miles from TMI.

3.6.1.1 Population and Land Use in the Vicinity of TMI

An estimated 489,000 people live in the vicinity of TMI (see Table 3.5). The population density of 570 persons per square mile is substantially higher than the state as a whole. The relatively high population density in the 10- to 15-mile zone is attributable to Harrisburg, the state capital, which is located 12 miles to the northwest of TMI; in 1970 Harrisburg had a population of 68,000. Goldsboro, with 576 people, is the municipality closest to TMI (about 1 mile) and is located directly opposite the nuclear station on the west shore of the Susquehanna. Other communities within the vicinity of TMI are listed in Table 3.6.

Table 3.5. Population and Population Density Distribution in the Vicinity of TMI, 1978

Radius (miles)	Population	%	Density (pop./sq. mi.)
0-5	37,842	7.7	353.1
5-10	116,973	23.9	487.8
10-15	334,305	68.3	654.0
Total	489,120	100.0	570.0

Source: Commonwealth of Pennsylvania, "The Socio-Economic Impacts of the Three Mile Island Accident."

Table 3.6. Municipalities in the Vicinity of TMI

Municipality	County	1970 Population	Distance from TMI (miles)
Goldsboro	York	576	1
Royalton	Dauphin	1,040	2
Middletown	Dauphin	9,080	2½
Highspire	Dauphin	2,947	4
Yorkhaven	York	671	4
Elizabethtown	Lancaster	8,072	6
Manchester	York	2,391	6½
Steelton	Dauphin	8,555	7
New Cumberland	Cumberland	9,803	9
Hummelstown	Dauphin	4,723	9
Hershey	Dauphin	7,407	10
Harrisburg	Dauphin	68,061	12

Source: U.S. Nuclear Regulatory Commission, "Final Environmental Statement Related to Operation of Three Mile Island Nuclear Station Units 1 and 2," December 1972.

While urban development is concentrated around population centers and along major transportation corridors, much of the land in the vicinity of TMI is devoted to farming. The soils, which exhibit good to excellent characteristics, combined with favorable physiographic and climatological features produce higher than average crop yields in the state. Field crops such as corn and wheat, and dairy, poultry, and livestock operations are prominent.[28]

3.6.1.2 Transportation

The vicinity of TMI is broadly delineated by four transportation routes that encompass an irregularly shaped area. Interstate 83, which is oriented north-south, connects Baltimore, York, and Harrisburg. Interstate 76, the Pennsylvania Turnpike, connects Harrisburg with urban centers to the east and west. Interstate 95 forms the southern border of the area and connects Baltimore City with other East Coast cities. State Route 10, which is oriented north-south, connects I-76 and I-95 on the eastern boundary. With the exception of the state route, the roadway net provides high-speed and high capacity access to the area. U.S. Route 30, another high capacity road, connects York with Lancaster and divides the region of impact into two subareas.

3.6.2 Socioeconomic Status of the Region

The immediate and short-term socioeconomic impacts of the accident at TMI have been reported in a number of studies. In this section, the status and continuing social and economic impacts of the accident are discussed in terms of relocation, property values, the economy, government, and recreational fishing. Most of the discussion that follows is in terms of county-level data.

3.6.2.1 Relocation

Of 14,000 households within five miles of TMI prior to the accident, 1504 experienced at least one member relocating to a new permanent address between March 28, 1979, and March 1980. Followup studies of those who relocated have not yet been undertaken to determine the extent to which the accident was a causative factor in the relocation decision. The data indicate that the

mobility rate is less than the rates for both the nation and the region. In addition, these data indicate that 44 percent of those who relocated within the 5-mile zone; an additional 30 percent relocated within the greater Harrisburg area.[29] Although specific reasons for relocation have not been established, the data appear to indicate that relocation to escape believed dangers of TMI may have been initially overstated by the staff (see DPEIS for TMI Unit 2, p. 3-19).

3.6.2.2 Housing and Property Values

In the vicinity of TMI, the latest available census data indicate that 152,600 housing units were occupied. Of this total, two-thirds of the units were occupied by their owners and nearly the same proportion of the units were at least 20 years old in 1970.[30] Housing studies of Dauphin and Lancaster Counties indicate that construction is meeting or perhaps exceeding demand.[31,32] The housing stock in York County has increased by an average of 1.8 percent annually since 1970; this rate accommodates household growth but not vacancies and housing losses.[33]

Two independent studies evaluated the impact of the accident on property values, sales, the number of days required to sell a property, and perceptions of mortgage lenders, realtors, and contractors.[28,34] In general, both studies conclude that the accident did not have a significant and continuing negative effect within 20 miles of TMI. With respect to sales price, historical and cross-sectional analyses indicate no adverse impact even when distance from the nuclear station is used as a category with which to classify properties. For a period of at least one month after the accident, the number of sales within 5 miles of the station did decline dramatically and brought the real estate market in this area to a virtual standstill. However, during the last half of 1979 sales resumed a more normal pattern. By the end of 1979, sales had dropped by only 7.3 percent below the 1978 rate despite high interest rates and the limited availability of mortgage funds in the last two quarters of 1979. During 1979, an average of 84.3 days was required to sell a residential property, which compares with 79.3 days required in 1978. This 6.3 percent increase is not considered significant in view of the depressed mortgage market in late 1979. Interviews with realtors indicate that the accident may have had an adverse impact in second quarter sales figures but that economic conditions had primary importance in explaining overall trends for the year. Some realtors have indicated that visibility of the towers has had a negative effect on the sale of individual homes; however, such instances are not numerous or severe enough to affect the housing market within 5 miles of TMI.

3.6.2.3 The Economy

The economy of the region (Cumberland, Dauphin, Lancaster, Lebanon, and York Counties) is diversified, although manufacturing is the dominant activity. In 1978 manufacturing firms employed 31 percent of the total labor force in the five counties. The three leading manufacturers are in machinery (except electrical), fabricated metals, and food and kindred products. The retail trade industry is the second leading source of employment. Census data for 1970 indicate that the occupational pattern of the region closely resembles that of the nation in the high proportion of professional, managerial, craftsman, and operative categories. Dauphin and Cumberland Counties lead the other counties in the proportion of their labor forces employed in white-collar occupations, a reflection of the concentrated opportunities for government employment in the greater Harrisburg area.[27]

In the paragraphs that follow, the effects of the TMI accident on four areas of the economy are reviewed.

An estimated 65 percent of all manufacturing firms within 20 miles of TMI (1141 firms employing 101,700 people) experienced no adverse economic impact for the 12-month period ending March 1980. Of the 245 firms which experienced some adverse impact, 68 percent (167 firms) reported no continuing effect by July 1979; 17 firms reported economic losses for the full year after the accident. In addition to economic impacts, an estimated 80 percent of the firms (918) indicated that their products suffered no adverse "image" effects in the market place; almost 7 percent of the firms (79) indicated that their products suffered one year after the accident.

Of the 20,197 nonmanufacturing units within a six-county area surrounding TMI,* an estimated 66 percent (13,418 units) experienced no economic loss as a result of the accident; 28 percent (5727 units) experienced adverse economic impact. By July 1979, 70 percent of those previously

*The six counties are Dauphin, Lancaster, Cumberland, Lebanon, Perry, and York.

experiencing losses (4000 units) ceased to do so. An additional 26 percent experienced no losses after December 1979, and 4 percent experienced economic adversity for the full one-year period. The services and retail sales sectors accounted for 80 percent of all lost sales, which amounted to $74 million; wages lost in all sectors were estimated to be $5.5 million. The proportion of nonmanufacturing units that indicated TMI was having a negative image effect has remained stable at 10 percent during the year.

The accident has been cited as being the cause of an estimated $5 million loss to tourist-oriented businesses in the six-county area. This loss occurred during the four-week period following the accident. Tourist industry losses continued through the summer of 1979 when the trend peaked and continued into the fall at reduced levels. Other factors, such as the outbreak of polio in the Amish Community, the gasoline shortage, adverse weather conditions, and worsening national economic conditions, also were seen as having a depressing effect on tourism, particularly during the summer and fall seasons. Indicators for the 1980 tourist season--hotel/motel occupancy rates and attendance figures--are approaching preaccident levels. Even among those businesses indicating continuing adverse impact, the impact is evaluated as being slight.

Economic losses in the agricultural sector were estimated to be less than $500,000; these losses were suffered during the five-week period after the accident. Farmers who sell milk directly to customers--"milk juggers"--are a notable exception to the general experience. For juggers, milk sales continued to linger 10 percent below 1978 levels as late as October 1979. However, this experience does not extend to farmers directly selling fresh fruit and vegetables to customers.

3.6.2.4 State and Local Government

The jurisdictional structure of the area in the vicinity of TMI is notable for its complexity. Parts of five counties--Cumberland, Dauphin, Lancaster, Lebanon, and York--lie within 15 miles of the station, and these jurisdictions are further subdivided into 37 townships and 25 cities or boroughs. One of the cities, Harrisburg, is the capital of Pennsylvania. As part of their general administrative responsibilities, county-level governments are charged with planning and managing emergency situations.[35]

Between March 30, 1979, and mid-August 1980, the State of Pennsylvania incurred TMI-related personnel and operating costs of $761,482.[28] Increased local government expenditures for emergency services and employee overtime costs amounted to $148,355. As of April 30, 1980, Metropolitan Edison's insurance covered 95 percent of the claims filed and 57 percent of the expenses reported (see Table 3.7).

Table 3.7. Status of Local Government Insurance Claims Against Met Ed Related to TMI Accident (as of April 30, 1980)

Government Level	Reported Expenses	Insurance Claims Filed	Insurance Claims Paid
Municipal	$ 90,319	$49,981	$45,444
County	58,036	39,558	39,429
Total	$148,355	$89,539	$84,873

Source: Commonwealth of Pennsylvania.

3.6.2.5 Recreational Fishing

In the months following the accident, surveys of fishing activity on the York Haven Pond of the Susquehanna indicated below normal or low normal levels of fish caught, fish kept (harvested), number of anglers, and total hours spent fishing (pp. 56-61 of Ref. 14). By July 1979, fishing conditions had returned to historically normal levels. The post-accident depression of fishing indicators has been attributed to altered behavior by local anglers--the bulk of fisherman on the York Haven Pond--who either ceased fishing activity or fished elsewhere.

In late July 1979, about 4000 gallons of water were released to the Susquehanna River from TMI. The amount of radiation was insignificant and posed no threat to public health and safety. The release was widely publicized by the news media.[36] Harvest indices declined in August and returned to normal in September.

3.6.2.6 Summary and Conclusions

The socioeconomic effects of the accident were mainly precipitated by the evacuation of households and individuals and by the altered behavior of those who remained in their communities. In retrospect, the socioeconomic effects, as indicated by lost sales and wages, decline in services, and increased expenditures, were, with some notable exceptions, short-lived. For a small number of business firms, tourist-oriented businesses, state and local government agencies, and milk juggers, the cost imposed by the accident continued for months after the accident. However, even among these groups the adverse socioeconomic impact appears to have declined by the end of 1979.

3.6.3 Psychological Setting

While some level of stress exists in all communities, the unique situation of the TMI-2 accident has sensitized the TMI population to plant-related activities. This results in increased levels of stress in some TMI community members, with greater awareness of the psychological consequences of the accident to themselves and the unknown prospects of the decontamination operations. The Kemeny Commission concluded, "The major health effects of the accident appear to have been on the mental health of the people living in the region ..."[37]

Stress is a complex process that involves environmental, social and psychological events that can cause harm, disruption, or loss.[40,68,69] These events, called stressors, are subject to people's appraisal of them. An event will elicit a stress response only if it implies threat or danger. If it does, the stressor will evoke physiological and psychological arousal that attempts to cope with or reduce believed threat. The consequences of arousal and response may include performance deficits, somatic distress, depression or anxiety. Thus, the experience of stress is determined by appraisal of the stressor. If the event is believed benign, a stress reaction is unlikely. If the event implies threat, a stress response is likely. Distress is a disruptive reaction to a stressor. The length of the stress response is determined by a person's ability to reduce the threat and the duration of the stressor. Some stressors, however, have consequences that last beyond their duration.

Stress is a complex phenomena that also presents a number of obstacles in studying it, especially definitional and measurement problems. The experience of stress varies widely across individuals and events. Low levels of stress may facilitate some task performances and inhibit others. In examining stress, it is useful to have a baseline measure with which to compare subsequent stress responses. Retrospective questions about past experiences or reactions are susceptible to bias, such as memory or mental status.[51] In spite of these problems, a wealth of information about stress has accumulated. Various psychological studies of the TMI communities have attempted to discern the nature and the extent of distress associated with the accident and with continued concern about decontamination.

Various studies agree that immediately following the TMI accident there was an increase in psychological distress in the communities surrounding the plant and that symptoms diminished by mid-summer 1979.[37,38,42-45] Some researchers conclude that the drop indicates the threat linked to the accident is acute or event specific.[37,45,47] Others suggest that while several items drop, others related to anxiety continue to remain high after the accident because of continuing uncertainty.[44,45,47,48] The continuing tension is largely due to the continued presence of a crippled reactor and anticipation of decontamination activities as well as to the accident experience. Often-mentioned evidences of stress are the public's expressed distrust of those responsible for these activities and their fears of possible, unanticipated, or underestimated health

consequences, or of delays in cleanup. DuPont,[49] in his analysis of ten years of nuclear energy TV-news coverage, suggests that public concern is amplified by the national TV media's generally negative coverage. Baum[47] suggests that a history of negative images linked to radiation and the atomic bomb is an important determinant in the public's assessment of nuclear power as unsafe or very risky.

Other research indicates that a portion of the public fears nuclear power for many of the same reasons.[41] The Kemeny Commission[37] suggests that psychological distress in the TMI community is induced and exacerbated by a lack of confidence in those parties directly involved in the cleanup. They conclude, however, that distress associated with the TMI accident is acute, with no significant increase of long-term mental or physical health problems anticipated.

Recent studies,[50,51] however, suggest measurable long-term psychological impact. Bromet[50] finds distress, measured by high levels of anxiety and depression, among the sample of mothers living in the vicinity of TMI. Mothers who exhibited symptoms of distress were likely to have one or more of the following characteristics: to have prior psychiatric history before the accident, to live within 5 miles of the plant, to not have extensive social support networks, and to be pregnant during the accident. Symptoms reported for the mothers living in the plant vicinity, however, are only marginally greater than those reported for the control group.

Houts[51] finds long-term psychological differences compatible to Bromet's[50] findings when he compares a TMI sample with a sample of people who live over 40 miles from the station. He is, however, very cautious in his interpretation because of the use of retrospective data.

In conclusion, the available TMI psychological studies[37,38,42-48,50,51] suggest to the staff that some portion of the population in the vicinity of TMI is stressed. There is probably more stress measured at TMI than found in other communities. The accident and post-accident events, however, have sensitized the population to psychological consequences of activities associated with the TMI nuclear station. This sensitivity may amplify stressors or make stressors appear to be related to TMI. The current stress levels in downriver communities have not been specifically studied. Houts,[42,51] however, uses a sample of the population over 40 miles from TMI as his reference for normal stress levels in his analysis. His sample is assumed to reflect stress levels downriver.

3.6.4 Susquehanna River/Chesapeake Bay

The most notable physical features of the impact region are the Susquehanna River and the Chesapeake Bay. Draining an area of 27,510 square miles, the 450-mile-long Susquehanna is a major river in the eastern United States and supplies about 50 percent of the fresh water in the bay. The Chesapeake Bay is one of the largest estuaries in the world, having a surface of about 4400 square miles, a length of nearly 200 miles, and more than 7000 miles of shoreline.[52,53] The Susquehanna River-Chesapeake Bay system is the focus of intense commercial and recreational fishing, boating, and water consumption.

3.6.4.1 Commercial Fishing

In 1978, the latest year for which summary data are available, the commercial harvest of shellfish and finfish in the Chesapeake fisheries* was 214 million pounds, with a landed value of $89 million.[54,55]

Data on harvests for selected years are summarized in Table 3.8. About 11,150 individuals currently earn their living as "watermen" (see Table 3.9).

For both the Maryland and Virginia portions of the Bay, shellfish are more important economically than finfish, representing more than 90 percent of the landed value in 1977 and 1978. In

*Chesapeake fisheries summaries incorporate data on Chesapeake Bay and its tributaries and the Atlantic Ocean off the Maryland-Virginia coast. Since 1970, the contribution from the Atlantic Ocean has grown as a percentage of the total Chesapeake fisheries harvest. In 1970, the ocean harvest represented 14 percent of the weight and 20 percent of the value; by 1978, these percentages had increased to 26 percent and 45 percent, respectively.

Table 3.8. Chesapeake Fisheries: Summary of Catch for Selected Years[a]

	Maryland				Virginia					
	Finfish		Shellfish		Finfish		Shellfish		Grand Total	
Year	Millions of Pounds	Millions of Dollars	Millions of pounds	Millions of Dollars	Millions of Pounds	Millions of Dollars	Millions of Pounds	Millions of Dollars	Millions of Pounds	Millions of Dollars
1960	21	2	48	12	307	6	59	14	436	35
1965	35	2	52	11	435	9	70	18	592	40
1970	16	2	64	17	495	12	55	10	630	41
1975	15	2	49	21	360	13	85	20	509	56
1977	16	3	46	29	544	27	64	29	670	87
1978	14	3	47	32	84	10	68	44	214	89

[a]Data refer to Chesapeake Bay, including its tributaries, and the Atlantic Ocean. Numbers may not add to totals because of rounding.

Source: U.S. Department of the Interior, Fish and Wildlife Service, "Fishery Statistics of the United States" for 1960, 1965, 1970, and 1975; U.S. Department of Commerce, National Marine Fisheries Service, "Maryland Landings Annual Summary 1978," Current Fisheries Statistics No. 7814 and "Virginia Landings Annual Summary 1978," Current Fisheries Statistics No. 7815.

Table 3.9. Registered Full-Time Watermen in Maryland and Virginia, 1977-1979

Year	Maryland	Virginia	Total
1977	4,003	4,700	8,703
1978	5,235	5,060	10,295
1979	5,349	5,803	11,152

Source: National Marine Fisheries Service.

Maryland waters, commercial fishing relies principally on blue crabs, oysters, soft-shelled crabs, surf clams, menhaden, and striped bass. Oysters represent the single most important cash crop i[illegible] and, exceeding $14 million and $15 million in 1977 and 1978, respectively. The Virginia commercial fishery relies principally on harvests of hard blue crabs, oysters, surf clams, sea scallops, croaker, menhaden, and flounder. Until 1978 the top Virginia fishery in terms of pounds caught and dollar value was the menhaden. Landings of menhaden peaked in 1977 with a harvest exceeding 501 million pounds that had a landed value in excess of $19 million. Harvests of blue crabs exceeded 36 million pounds in both 1977 and 1978, with a dollar value of $6.7 million for each of those years.[54,55]

That portion of the finfish and shellfish harvest not sold directly to restaurants, markets, and individual consumers is sold to processors and wholesalers. In 1978, the value of processed fish products in Virginia and Maryland was $221 million.[56,57] (An undetermined portion of fish entering the processing/wholesaling system originates in the Atlantic Ocean off Virginia and Maryland and in water bodies elsewhere.) The processing and wholesaling system employs an average of 6900 people throughout the year in 350 establishments. Each year during the peak months the number of employees added to establishment payrolls rises. For instance, in 1978, the average employment for the peak months was 8900, an increase of 29 percent above the yearly average. These data are summarized for selected years in Table 3.10.

The economic benefits generated by harvesting Chesapeake seafood include stimulation of economic activity in other segments of the fishery industry and in other economic sectors. These benefits accrue to areas throughout the nation in addition to the Maryland-Virginia area. A recent study concluded that the multipliers for income and for employment were 3.47 and 3.13, respectively.[58] That is, for every dollar of seafood harvested from the Chesapeake fishery, an additional $2.47 is generated in other sectors, and each full-time waterman supports an additional 2.13 jobs. As indicated in Table 3.11, the 1978 Chesapeake harvest had a landed value of $89 million and produced an additional impact of more than $219 million; the 11,150 full-time, licensed watermen of the bay helped to create employment for an additional 23,700 people.

The Chesapeake Bay commercial fishing industry can be characterized in the following ways:*

- Harvesting, processing, and wholesaling is organized around widely scattered individuals or small groups using simple--often homemade--gear. The waterman operates on a low-capital, low-overhead basis that reflects the need to function in an environment of great uncertainty and flux. In many instances harvesting and processing operations are supported by family members, are labor intensive, and, with respect to processing and packaging facilities, are marginally profitable.

*Information in this section is based on References 59-64.

Table 3.10. Chesapeake Fisheries Processing and Wholesale Activity for Selected Years

	Maryland				Virginia				Total			
Year	Value of Processed Product[a]	Seasonal Employment[b]	Yearly Employment[b]	Number of Plants	Value of Processed Product[a]	Seasonal Employment[b]	Yearly Employment[b]	Number of Plants	Value of Processed Product[a]	Seasonal Employment[b]	Yearly Employment[b]	Number of Plants
1960	31	6,687	4,271	253	24	7,123	2,726	378	55	13,810	6,997	631
1965	37	4,965	4,112	285	30	4,714	2,914	336	66	9,679	7,026	621
1970	58	4,812	3,790	205	42	5,834	4,021	260	100	10,646	7,811	465
1975	98	4,264	3,395	175	63	6,048	4,513	186	161	10,312	7,908	361
1977	126	3,831	2,761	169	99	5,226	4,041	180	225	9,057	6,802	349
1978	118	3,632	2,775	169	103	5,281	4,101	185	221	8,913	6,876	354

[a]Data are in millions of dollars.

[b]Employment averages for both the season and the year.

Source: From U.S. Department of the Interior, Fish and Wildlife Service, "Fishery Statistics of the United States," for 1960, 1965, 1970, and 1975; U.S. Department of Commerce, National Marine Fisheries Service, "Processed Fishery Products, Annual Summary 1977," Current Fisheries Statistics No. 7503 and Annual Summary 1978, Current Fisheries Statistics No. 7803.

Table 3.11. Economic Impact of Chesapeake Bay Commercial Fisheries

	Dollars	Employment
Base[a]	89,000,000	11,150
Multiplier	3.47	3.13
Gross impact	308,830,000	34,900
Net impact	219,830,000	23,750

[a]Landed value and number of employees.

- The industry is subject to seasonal cycles and biological and environmental fluxes. Watermen work closely within these natural constraints.

- Recent productivity data indicate a trend toward a general, long-term decline as well as a cyclical low point in the Bay's resources. A general decline occurring simultaneously with increased pollution represents the single most important threat to continued fishery stability.

- The industry is faced with increasing competition as products from other states supplant Maryland and Virginia products. Crabs from the entire Gulf of Mexico and the Southeast Atlantic Coast and oyster imports are particularly notable in this regard. In addition, the market for fresh Maryland seafood is being subjected to pressures from frozen seafood products.

- Despite the rise in capital costs and operating expenses, gross incomes appear to be reasonable. Income has not experienced greater downward pressure because of the existence of an extensive non-cash economy and because of self-imposed limitations on the modernization of equipment. Among processors and wholesalers, profit margins are low and variable and are subject to pressure from rising transportation and labor costs.

- The markets for Chesapeake fishery products are widely scattered and are composed of many individual outlets: restaurants, retail stores, chain stores, and institutions. Distributors have individual marketing systems based on personal knowledge and geographic routes. Regular markets are found as far north as Montreal, as far south as Texas, and west to Chicago and Los Angeles.

- The Chesapeake fishing industry is supported by small, scattered, and self-sufficient community structures that are characterized by extended family settlement, kinship and informal relationships, and non-cash exchanges for materials and services.

3.6.4.2 Sport Fishing

Sport fishing on the Chesapeake Bay is an activity that rivals commercial fishing in both intensity and value. Estimates based on the 1976 sport fishing survey conducted by the State of Maryland indicate that almost 2 million finfishing and crabbing trips were made to the Maryland tidewater (comparable data for Virginia waters do not exist).[65] Approximately 88 percent of these trips were on private, non-charter boats; the remainder, 242,000 trips, were on charter boats. Almost 70 percent of Maryland residents on private fishing boats were residents of Anne Arundel, Baltimore, and Prince Georges Counties and of Baltimore City. Of nonresident interviewees, 93 percent originated from four jurisdictions: Washington, D.C., Delaware, Pennsylvania, and Virginia.

The total harvest of finfish and crabs during the period May through October--the peak sport fishing season--amounted to 18.6 million pounds. As indicated in Table 3.12, which compares the sport and commercial harvest, overall sport activity is equivalent to a considerable percentage of the commercial harvest. Of the total commercial finfish catch, more than 2.7 million pounds were menhaden and gizzard shad, species not sought by sport fishermen.[65] If the commercial catch of these species is ignored, the sport catch was greater than the commercial finfish harvest.

Table 3.12 Chesapeake Bay Sport and Commercial Harvests, 1976

	Harvest (pounds)	
Category	Sport	Commercial
Finfish	15,434,600	17,581,700
Crabs	3,160,900	20,885,500
Total	18,595,500	38,467,200

Source: Maryland Department of Natural Resources, Fisheries Administration, "1976 Maryland Chesapeake Bay Sport Fishing Survey," June 1977.

One economic measure of the significance of sport fishing is the expenditure on variable costs. Variable costs are out-of-pocket payments for transportation, food, bait, boat gas, lodging, and boat rentals. These expenditures maintain the shore-support service industry for sport fishing in the Chesapeake Bay area. Based on Maryland's 1976 survey of fishing trips, average party size, resident/non-resident proportions, and adjusted variable cost data, the staff estimates that the total out-of-pocket expenditures in Maryland for 1979 were $26.6 million (see Table 3.13). It should be noted that the calculation does not take into account fixed costs,* which could easily double the staff's estimate.

Sport fishing in the Susquehanna River occurs from the vicinity of TMI to Havre De Grace (see Appendix E). The sport fishery resource in the river primarily serves local residents, although fishermen from the Baltimore area and from Pennsylvania are attracted to the river in sizable numbers. Much of the fishing effort is concentrated in the ponds, particularly Conowingo Pond, and near the tailwaters of the dams and hydroelectric facilities. Because of the variety of fish species (freshwater and anadromous, lotic and lentic, bottom and pelagic), the fishery resource offers the sport fishermen a variety of experiences.

3.6.4.3 Wildlife Resource Use**

In addition to Chesapeake Bay's importance to the state's commercial fishing industry and to sport fishing, the Bay's marshes and woodlands provide many thousands of acres of natural habitat

*Fixed costs include expenditures for rods, reels, boats, trailers, and clothing. However, such costs do not represent money spent on the fishing day nor do such costs necessarily represent monies retained within Maryland.

**Except as noted, information in this section is from Reference 52.

Table 3.13. Number of Boat Trips and Out-of-Pocket Expenditures by Sport Fishermen in Maryland, 1979[a]

Factor	Resident	Non-Resident	Total
Angler trips in private boats	1,503,017	263,687	1,766,704
Variable cost/angler (dollars)	9.71	16.08	10.66
Annual expenditure (dollars)	14,594,295	4,240,087	18,834,382
	Half Day	Full Day	Total
Charter boat trips	12,477	35,883	48,360
Variable cost/ boat trip (dollars)	121.32	175.24	161.32
Annual expenditure (dollars)	1,513,710	6,288,137	7,801,847

[a]Variable cost data for 1976 were inflated by 1.348 to account for change in the Consumer Price Index up to December 1979.

Source: H.J. Speir, et al., "1976 Maryland Chesapeake Sport Fishing Survey."

for a diversity of wildlife. The Bay area is in the path of the Atlantic Flyway and provides wintering areas and feeding grounds for migrating waterfowl. About 75 percent of the wintering population of Canada geese are attracted to the Delmarva Peninsula. Other waterfowl species attracted to the peninsula in large numbers include ducks, whistling swans, birds requiring the wetlands of the bay area for food and other habitat requirements, and a variety of game birds. The interspersion of forest and farmland and the presence of shoreline and wetlands provides a variety of food chains and habitat niches that support a diverse population of small mammals, reptiles, and amphibians.

The wildlife resources of the bay region are utilized by man in two basic ways. Consumptive utilization involves hunting and trapping to provide food and clothing for personal use and for sale. Nonconsumptive activities are undertaken purely for enjoyment and include bird watching, nature walking, and nature photography.

Although detailed information on the economic value of hunting and trapping activities is not available for the bay area, expenditures by sportsmen are thought to be significant. Based on data from the 1975 national survey of hunting and fishing, the typical migratory bird hunter in Maryland and Virginia spends about $147 annually for such items as license fees, rentals, food, lodging, gasoline, and guides.* Applying this figure to the total number of licensed hunters in Maryland and Virginia, total out-of-pocket expenditures would be more than $38.4 million annually in 1979 dollars. Maryland is second only to Louisiana in terms of the value of pelts and food

*Dollar values cited have been calculated by multiplying values cited in Reference 66 by 1.426 to account for changes in the Consumer Price Index up to December 1979. As in the estimate for sport fishing expenditures, the expenditures cited here for hunters do not include fixed costs.

trapped. Nutria and muskrat--most of which are trapped around the Chesapeake Bay--support a $3 million industry in Maryland.[67]

With respect to nonconsumptive activities, the 1975 survey indicated that the total number of person-days engaged in bird watching, nature walking, and photographing wildlife was 30 percent more than the total number of person-days spent fishing and hunting.

3.6.4.4 Boating

Since World War II, there has been a large growth in use of the bay by water-oriented recreationalists, such as power boat and sailboat owners. Population growth, urban development around the bay, increased disposable income and leisure time, and improved road systems have been responsible for increased boating activity. Although specific boat ownership and use data are not available, the bay is second to Long Island Sound in the intensity of boating use.[67] Persons participating in nonconsumptive activities spend large amounts of money on food, lodging, and rentals; however, estimates of these expenditures do not exist.

3.6.4.5 Summary

The preceeding sections described the Chesapeake Bay as a socioeconomic resource. However, data on landed values of commercial fisheries, employment in fish harvesting and processing, sport fishing expenditures, and the consumptive and nonconsumptive utilization of wildlife only begin to dimension the character, even the economic character, of the Bay. Dollar values can only inadequately express the subjective enjoyment of the recreational boater, hunter, and fisherman or the waterman's sense of independence. Nor can economic multipliers adequately portray the thousands of small businesses, families, and communities that support the hunting and fishing activities on the Bay.

It is also clear that a comprehensive and exact accounting of the economic values associated with Chesapeake Bay fishing and hunting activities--even if such an accounting were possible--would be a small percentage of the value of agricultural products and even less significant when compared with the value added in the manufacturing sector. However, the socioeconomic significance of the Chesapeake Bay lies not in economic comparisons but rather in the area's distinctiveness. The fishing industry, the waterman, the waterfowl hunting activity, and the small rural communities existing on the Bay's resources represent a unique culture and tradition that enhance the diversity of Maryland and Virginia. And the benefit of such diversity may well be impossible to measure.

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4. MAINTENANCE OF THE REACTOR IN SAFE CONDITION

4.1 OBJECTIVES AND ACTIONS

The objectives of maintaining the TMI-2 reactor in a safe condition can be summarized as: achievement of a thermally stable primary system in which the decay heat from fission products is continually being removed; maintenance of subcriticality of the reactor core; and confinement of the radioactivity within the reactor building.

Beginning in late April 1979, natural recirculation was used to remove the decay heat from the reactor core to steam generator "A", where subsequent cooling by the secondary water transferred the heat to the atmosphere. Throughout most of 1980, the natural recirculation was periodic as the decay heat level decreased. A new forced-circulation system, the mini-decay-heat-removal system, was proposed by Met-Ed for use in placing the reactor under a long-term cooling mode in which temperatures can be more effectively controlled. This system has been installed, but is now retained as an additional available system. Since January 5, 1981, the reactor has been cooled by a loss-to-ambient cooling mode which has been shown to be adequate to cool the reactor at its present decay heat rate. This cooling mode transfers the decay heat directly to the reactor building atmosphere from the reactor system through natural heat loss.

Subcriticality of the reactor is being ensured by the maintenance of sufficient boron in solution in the reactor primary coolant. The one operable source range neutron detector is used to monitor subcriticality. A small amount of control rod material is believed to have melted during the accident. The shutdown margin available at TMI-2 is estimated to be about 15% $\delta k/k$.*

The potential for recriticality under various hypothetical circumstances has been examined independently by several groups. Based on these analyses[1-5] it can be concluded that with 3500 ppm of boron in the primary coolant, the reactor can be maintained in a subcritical state even in the total absence of other control materials. The most probable (although very unlikely) cause of recriticality was found to be boron dilution, which would be a slow enough process that any approach to criticality can be detected and remedied.

4.2 MONITORING OF REACTOR AND REACTOR BUILDING

The reactor and the reactor building are being monitored by instruments measuring the reactor and building temperature and pressure and the water level in the sump inside the building. As of December 1980, five entries had been made into the reactor building. Radiation level readings, equipment inspections, and some repair work have been performed. The chemical analysis of the reactor coolant water continues to be checked weekly, and an inventory balance on the reactor coolant water is currently performed every four hours. One instrument channel is still functional for monitoring the low neutron flux level and the licensee is working on making a redundant instrument operational. In-core thermocouples, hot-leg and cold-leg resistance temperature devices, and pressure gauges are available to monitor temperature and pressure inside the reactor coolant system.

4.3 DECAY HEAT REMOVAL

The term "decay heat" refers to thermal energy generated by radioactive fission products and other in-core materials after the shutdown of a reactor. The decay heat power of the TMI-2 reactor at the time it was first shut down a few seconds after 4:00 a.m. on March 28, 1979, was 160,000 kW. By January 31, 1980, it had decayed to 200 kW and continued to diminish to about 43 kW by February 1, 1981.

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*This means that there are approximately 15% too few neutrons to sustain nuclear chain reaction at a constant rate (see Ref. 1).

4.3.1 Decay-Heat-Removal Mechanisms Employed Since the Accident

From the onset of the accident until 8:00 p.m. March 28, 1979, when reactor coolant pump 1A was restarted, heat removal from the reactor core was inadequate and was effected primarily by releases of primary water to the reactor building through the pressurizer relief valve that was stuck open. Between 8:00 p.m. on March 28, 1979, and April 27, 1979, decay heat was removed through steam generator "A" by forced circulation with a reactor coolant pump (see Fig. 1.1).[3] From April 27, 1979, through December 1980, decay heat was removed by natural convection circulation of the primary water through steam generator "A" in a steaming mode. Since January 5, 1981, the reactor has been cooled by heat loss from the reactor coolant system loops and reactor vessel to the reactor building air, with the building air cooled by the building cooling system.

4.3.2 Available Decay-Heat-Removal Systems

There are several methods by which decay heat can be removed. Four of the preferred modes are (1) the loss to ambient cooling mode, (2) the Mini-Decay-Heat-Removal System (MDHRS), (3) the long-term "B" steam generator cooling system, and (4) steaming through steam generator "A". All four methods have been approved by the NRC and are available to Met-Ed as alternative means by which decay heat removal may be effected. In addition to the above methods, the normal decay heat removal system is also available if needed.

4.3.2.1 Long-Term "B" Steam Generator Cooling System

Following the accident, provisions were made for long-term decay heat removal through steam generator "B". Additional pumps, piping, and heat exchangers were added to the steam generator "B" system to facilitate heat removal for indefinite periods of time.

4.3.2.2 Mini-Decay-Heat-Removal System

The MDHRS has been installed as an additional method of transferring decay heat from the fuel in the reactor system to the nuclear service water system.[1] It also may be used during defueling operations when the reactor coolant system will not be configured to maintain forced cooling. The system includes two pumps and two heat exchangers, arranged in a manner that will permit independent operation and thereby provide redundant decay heat removal capability. Each heat exchanger has the capability to remove the total decay heat. Two of the valves that connect the reactor coolant system to the MDHRS (see Sec. 2.1.1.2) have been opened. The remaining isolation valves are closed and will be opened just before the system will be used. The MDHRS is operational and is available for use if needed.

4.3.2.3 Loss-to-Ambient Cooling Mode

On November 6, 1980, a test of the loss-to-ambient cooling mode for controlling temperatures in the reactor was initiated. The turbine bypass valve from the "A" steam generator to the condenser was closed, isolating the reactor cooling system from all active cooling modes. The reactor was then cooled by heat losses from the system to the air inside the reactor building. The test was terminated after a month on December 9, 1980; the core and reactor cooling system temperatures had increased by about 10°F. The technical specifications were then modified to recognize loss-to-ambient as an acceptable means for long-term cooling of the reactor core.

The loss-to-ambient cooling mode was further tested during December 1980. Operating procedures were then prepared, and loss-to-ambient cooling was subsequently adopted as the primary means by which the reactor core would be cooled, with the long-term "B" steam generator and MDHRS also available as alternative cooling modes. Since January 5, 1981, loss-to-ambient cooling has been in effect.

4.3.2.4 Steaming to Steam Generator A

Since January 5, 1981, this mode has been discontinued as the primary mode for core cooling; however, return to this method of decay heat removal is still feasible and acceptable to the NRC.

4.3.2.5 Decay Heat Removal System

The Decay Heat Removal (DHR) System has not been used during the postaccident period, but is still an available mode of cooling. The MDHRS was designed to take the place of the DHR System

since the high-flow-rate DHR System is not necessary at the very low decay heat levels. Use of the DHR System is not expected under any likely circumstances, but is available for operation.

4.4 EFFLUENTS AND RELEASES TO THE ENVIRONMENT

4.4.1 Normal Releases

4.4.1.1 Gaseous Releases

From the time when steam generator "A" started operating in the steaming mode (April 27, 1979) until June 28, 1980, when the reactor building atmosphere was vented, about 66 to 80 Ci of Kr-85 had been leaking out of the TMI-2 reactor building every month.[6] In the steaming mode of steam generator operation, the turbine side of the steam generator is maintained in a partial vacuum by the plant air ejectors. Consequently, the pressure difference between the reactor building atmosphere and the turbine side of the steam generator enhanced leakage of Kr-85 from the reactor building through the packing of various steam valves to the secondary system. The Kr-85 gas and other gases (nonradioactive) were subsequently discharged from the secondary system through the auxiliary building ventilation system to the environment. The technical specification amendment accompanying the NRC order to permit the controlled purging requires that the dose to maximally exposed individuals during the purge be limited to: (a) 15 mrem skin dose, (b) 5 mrem total body dose and (c) 20 percent of the limits in (a) and (b) above may not be exceeded over any one-hour period. Those technical specification requirements were met.

From mid-July 1980 through November 1980, under the authority of the Interim Release Criteria, the reactor building was purged 13 times, sometimes in conjunction with the various entries that have been made. The largest single release was made August 14-15, 1980, when about 84 Ci were purged. The average release rate from August to December 9, 1980, was about 67 Ci per month, which is equivalent to about 15 Ci per week.

In addition to releases during periodic purging, small amounts of Kr-85 have been released from other sources within the plant (e.g., the auxiliary building ventilation system). The release rates are near or at equipment detection limits and are less than one curie per month.

4.4.1.2 Routine Operational Liquid Wastes

TMI Unit 2 is similar to other commercial nuclear power plants in that reactor support facilities, e.g., the auxiliary and fuel handling building, have special floor drain systems to collect miscellaneous liquids generated within these facilities. These drain systems do not collect TMI "accident water" (as defined in Section 1.6.2.1) but receive initially nonradioactive water, such as, river water inleakage, that becomes contaminated by radioactive material that is present in the buildings. This type of liquid radwaste is commonly referred to as "Routine Operational Liquid Wastes" (ROLW). As a result of cleanup efforts, the volume and radioactivity content of ROLW at TMI is now similar to that of a normally operating power reactor. The only difference is that the source of the relatively low levels of radioactive contamination in the drain system is the March 28, 1979, accident, as opposed to fuel leakage and activated corrosion products at normally operating power plants. Table 4.1 is a summary of the expected sources, volumes and curie content of untreated ROLW.

To date, there has been only a limited amount of ROLW generated at TMI Unit 2. The initial AFHB cleanup water has been processed as accident water because of its concentration of radioactive materials. Between March 28, 1979, and September 1980, ROLW from the contaminated drain tanks (see Table 4.1) was transferred to the Unit 1 EPICOR I radwaste system for processing and routine discharges. In September 1980, the Unit 1 EPICOR I system was disconnected from Unit 2. As a result, all ROLW is presently being collected and stored in Unit 2.

It is expected that as cleanup operations progress, the amount of ROLW generated will approach the amounts listed in Table 4.1. This waste will be treated in a typical radioactive liquid waste treatment system similar to that found in any power reactor. For planning purposes, it is assumed that an ion exchange system consisting of particulate filters followed by two demineralizer beds in series will be employed. Based on past performance of such systems, it is expected that the system decontamination factor will be at least 1000 for isotopes other than tritium. Tritium will not be removed by the system. The radioactivity content of the treated water would be less than that at a normally operating nuclear power plant and is listed in Table 4.2. It is anticipated that this treated water will be discharged after sampling and analysis to assure compliance with all regulatory criteria.

Table 4.1. Routine Operational Liquid Waste Sources

Source	Average Radioactivity Concentration (μCi/mL)[a]	Estimated Volume (gallons per day)
AFHB Floor Drains	10^{-2}	100
•River Water Leakage from Pump Seals		
•Air Conditioning Condensation		
•Closed (non-radioactive) Cooling Water System Leakage		
•Demineralized Water (non-radioactive) Flush Water		
AFHB Contaminated Drain Tank	10^{-4}	50
•Personnel Showers		
•Secondary Plant Sampling		
•Chemistry Laboratory Drains (low-level, e.g., glassware washing)		
•Instrument Repair & Calibration Shop		
Service/Personnel Access Facility[b]	10^{-4}	100
•Chemistry Laboratory Drains		
•Personnel Showers		
•Air Conditioning Condensation		
•Demineralized Water (non-radioactive) Flush Water		

[a]Includes an estimated average tritium concentration of 1×10^{-4} μCi/mL. The remainder has an approximate isotopic distribution of Cs-137 (89%), Cs-134 (9%) and Sr-90 (2%).

[b]Proposed facility.

Table 4.2. Release to the Susquehanna River of Treated Routine Operational Liquids

Radionuclide	Amount of Release (Ci)	Average Concentration Prior to Discharge (μCi/mL)	Average Concentration Entering River[a] (μCi/mL)	Licensee's Lower Limit of Detection (μCi/mL)
H-3	3×10^{-2}	9×10^{-5}	3×10^{-8}	3×10^{-7}
Cs-134	1×10^{-4}	4×10^{-7}	1×10^{-10}	6×10^{-9}
Cs-137	1×10^{-3}	4×10^{-6}	1×10^{-9}	6×10^{-9}
Sr-90	3×10^{-5}	9×10^{-8}	3×10^{-11}	1×10^{-9}

[a]Based on dilution factor of 3400 from service water discharge.

The operation of the ROLW treatment system will produce a relatively small amount of solid waste. It is expected that two 100 ft.^{3} spent demineralizer liners containing approximately a total of one curie of radioactivity will be generated per year. These should be disposed at a low-level radioactive waste disposal site.

4.4.2 Accident Scenarios and Associated Releases

There are two broad categories of core-related accidents that could result in the release of additional radioactive fission products from the damaged fuel in the reactor core. The first is sufficient overheating or mechanical damage (fracturing) to lead to the escape of some of the radioactive fission products still held within the core. Most of the remaining fission products are still trapped within the fuel particles in the core and would require very high temperatures (on the order of those reached during the original accident) to be released from the fuel. However, there may be small pockets of more readily released fission products (e.g. Kr-85 in a small gas bubble) that could be released by mechanical damage. The second broad category of core-related accidents is an inadvertent restart of the reactor with the associated generation of new fission products and heat. This latter category is referred to as a recriticality accident. Recriticality and accidents related to overheating of the whole core are discussed in the sections that follow.

In addition, the staff has considered the leakage of the sump water from the reactor building through the ground and into the river. This accident is also discussed.

4.4.2.1 Recriticality

Some neutrons, from extraneous sources and the spontaneous fission of uranium, are present in the core even when the reactor is shut down. These neutrons do cause some uranium atoms to fission; however, the rate of fission when the reactor is shut down is minute. For the reactor to restart (recriticality), the rate at which new neutrons are supplied by fission must be greater than the rate at which they are removed by capture in the core or by leakage from the core and capture in the material surrounding the core. Thus, to keep the reactor shut down (subcritical), it is necessary to ensure a high neutron-removal rate. Prior to the accident the neutron-removal rate was controlled by adjusting the concentration of boric acid in the reactor coolant and by adjusting the position of the silver-indium-cadmium control rods (both boric acid and the control rod material are strong neutron absorbers). In addition, fixed alumina-boron carbide (Al_2O_3-B_4C) absorber rods also were used to adjust the local neutron-removal rate.

Trace amounts of silver, indium, and cadmium have been detected in a primary coolant sample by laboratory analysis,[5] suggesting that some of the reactor control rods have been damaged because of melting. Precise information on their condition is not available. It has been speculated that any control material that may have melted probably has not left the core region, but has been redistributed in the fashion of resolidified candle wax drippings.[6] There also is a possibility that boron from the boric acid in the coolant might have precipitated in the core when portions of it were uncovered during the accident. Furthermore, it is likely that the fixed (Al_2O_3-B_4C) absorber rods in the less damaged portions of the core are reasonably intact because they have a high melting point (2040°C). Thus, while it is reasonable to expect that substantial quantities of the original neutron absorber materials remain in the core, there may be regions of the core with insufficient quantities of the fixed or movable absorber rods to maintain a subcritical condition without additional boric acid. Therefore, to ensure subcriticality, the amount of boric acid in the reactor coolant has been increased to about 3850 ppm and a new lower limit of 3500 ppm has been established. There is agreement in numerous criticality analyses[1-5,7,8] that the core would remain subcritical with a boric acid concentration of 3500 ppm in any physically possible geometry even if all the fixed and movable absorber rods were removed.

To ensure that the reactor remains subcritical throughout the decontamination program, it will be necessary to maintain control of the boric acid concentration in the reactor coolant system until the defueling is completed. Since the solubility of boron as boric acid is 4400 ppm at 35°F and 7100 ppm at 59°F, the possibility of boric acid precipitating out of solution because of a reduction in temperature can be ruled out altogether. Other means (such as chemical reactions or pH effects) by which boron concentration inside the primary system would be compromised are addressed in Reference 3; for example, large amounts of strong acid would have to be added before significant decreases in soluble boron concentration were observed. However, there are no foreseeable circumstances under which these conditions would occur.

The only concern indicated by the studies regarding recriticality in the TMI-2 situation is the introduction of water to the reactor core with a boron level of much less than 3500 ppm.[3,4,8] Calculations supported by experimental data from the Westinghouse Reactor Evaluation Center have shown that the introduction of 1000 ppm borated water in the outer region of the 2.96-percent-enriched square array of the fuel assemblies could result in recriticality.[4] Original undamaged core geometry has been assumed in those calculations. Lack of information on the current state of the core makes it difficult to accurately calculate the critical boron concentration. Calculations do show, however, that the introduction of underborated water could result in the core becoming critical.

The TMI-2 reactor was designed with the intention of using boric acid as a means to control reactivity (neutron balance in the core). Thus, prevention of a boron dilution accident was incorporated in the design through limiting the sources of pure water and the rate at which it could be added to the reactor coolant system. Although very unlikely, if a boron dilution event were to occur, several methods of detection are available to alert the operators so that the dilute water source could be terminated prior to core criticality. These methods include: excore nuclear instrumentation, periodic boron analysis of reactor coolant water, and pressure and temperature readings of the reactor coolant system. In addition, boric acid is normally used as a neutron absorber in the fuel transfer canal and the spent fuel storage pool. While the present boric acid levels are much higher than normal and the geometry of the core is not known, the basic approach of using boric acid to ensure shutdown is both a sound and normal practice. Control of the chemistry of the reactor coolant system and all sources of other water added to or mixed with the reactor coolant during the cleanup will prevent a dilution accident.

Initiation of a boron dilution event in the current reactor system is very unlikely. The standby pressure control system (SPCS), which is currently connected to the reactor coolant system to maintain RCS pressure and provide makeup water, is fully borated and sampled weekly for boron concentration. Boron concentration is maintained between 3000 and 4500 ppm. SPCS tank levels are monitored every four hours to determine the quantity of water injected to calculate a reactor coolant system leak rate (which is currently less than 0.1 gpm at about 100 psig reactor coolant system pressure). Other pumping systems connected to the RCS, such as makeup or decay heat removal systems, were previously borated or will be borated prior to startup so that system activation would not add deborated water to the RCS. A closed valved connection from the demineralized water system to the makeup pump suction does exist but is located in a high-radiation area to which access is administratively controlled. In addition, the power supply breakers to the makeup pumps are danger-tagged open to prevent inadvertent pump operation. Even if a pressurized demineralized water source were introduced to the reactor coolant system in its current configuration (i.e., filled solid with water), an increase in reactor coolant system pressure indication should alert the operator to terminate the source.

The present core inventory of radioactive fission products is very low. In fact, it is many orders of magnitude less than the inventory of the reactor during normal operation, which is used as a basis in hypothetical accident studies and for which public safety is assured. Most of the fission products produced in a recriticality accident would be extremely short-lived, i.e., they would decay away very rapidly. For example, the total amount of curies released to the atmosphere in the March 28, 1979, accident was greater than the total fission product inventory that would be in the current core ten hours after a hypothetical severe recriticality transient.[8] The reactor building is specifically designed to contain fission product inventories that are orders of magnitude greater than the current core inventory at containment pressures far greater than those which would exist during defueling. Nevertheless, protection of working personnel dictates that all safeguards be taken to preclude recriticality.

4.4.2.2 Core Cooling Failures

Since the accident, electric power for equipment used to remove decay heat from the core has been provided by circuits from offsite transmission networks and onsite power supplies. A 13.2-kV circuit from the Middletown junction substation and two redundant diesel generators were installed to increase reliability of power supplied to the TMI-2 system. In March 1980, Met-Ed proposed to the NRC to reconfigure the power system such that the two diesel generators and the 13.2-kV transmission line would be replaced by the existing 230-kV grid system. This system has now been approved by NRC and has been put into effect. Three separate, redundant, and independent sets of combustion turbines located in three geographical locations would be used to supply the grid. Consequently, the probability of complete loss of power needed to sustain decay heat removal operations is extremely small.

An accident scenario involving a leak in the reactor vessel which leads to uncovering and overheating of the core is discussed in Section 10.4. At the present decay heat levels, ample time is available to take corrective action if such a condition develops. In addition, with the low heat-generation rate and low system pressure, it would not be difficult to provide sufficient water to prevent overheating. However, if overheating is assumed to occur and all the remaining Kr-85 and cesium were released to the reactor building, the total activity from this hypothetical release would be about the same as the activity inside the reactor building prior to purging. Given the long time available to take corrective actions (many hours to several days), the reactor building, if occupied at the time, could be evacuated and isolated before any significant release from the core had occurred.

4.4.2.3 Leakage of Reactor Building Sump Water

The largest amount of contaminated water currently on the site is the 700,000 gallons of water in the bottom of the reactor building. This water contains an estimated 500,000 Ci of radionuclides. It is postulated that if this water should leak through the steel-lined concrete base of the reactor building, it would ultimately reach the Susquehanna River. As discussed below, such an accidental release is considered highly unlikely. A detailed discussion is also provided in Appendix V.

The reactor building is constructed to prevent water in the sump from leaking to the environment. The reactor building was designed and fabricated to withstand an internal pressure of 60 psig and to withstand major earthquakes. The foundation mat for the reactor building is 11½ ft thick and rests on bedrock. The reactor building has a carbon steel liner, 3/8 inch thick on the sides, 1/2-inch-thick dome, and 1/4-inch-thick base. A concrete slab 2 ft thick was poured above the base liner plate. The exposed face of the liner is coated with a prime and finish coat of phenoline 368. An additional leakage barrier is provided by a 40-mil-thick PVC wasteproofing membrane which is installed over the outer surfaces of the foundation mat.

In the unlikely event that water from the reactor building sump should leak through these barriers, several methods for early detection and control of this leakage are available. One indication of leakage would be a decrease in the sump water level, which is monitored daily. The detection of radioactivity in the tendon gallery and its sump would also be an indication of a leak. The tendon gallery is located outside the steel liner but in the perimeter of the foundation base mat. Detection of radioactivity in the site monitoring wells (see Sec. 11) would also indicate leakage from the reactor building sump. A final indication of leakage would be through the detection of radioactivity in the Susquehanna River from the licensees' environmental monitoring program and at the water intakes from the City of Lancaster through its radiological monitoring program.

Leakage from the reactor building sump could be controlled by any of several measures. Upon detecting a leak from the reactor building sump, the remaining water in the sump could be pumped into various available storage tanks in accordance with the licensee's existing contingency plan for this action. Another method for controlling such leakage would be to pump the contaminated water out of the monitoring wells which are located around the reactor building. A further method for controlling the release of activity from this postulated leakage would be the installation of a grout curtain between the reactor building and the Susquehanna River.

For the purposes of analyzing the transport of Sr-90 and Cs-137 from a reactor building sump leak, it is conservatively assumed that the water is released into the ground over an area the size of the entire base of the reactor building. Such a postulated release could drain only that portion of the contaminated water that lies above the water table. To simplify the calculation, this volume (470,000 gallons) is assumed to enter the groundwater instantaneously. For tritium, whose analysis depends on a somewhat different set of phenomena, the entire 470,000 gallons was conservatively assumed to be released instantaneously from a point at the center of the building.*

Water containing tritium would begin entering the river after a minimum of 350 days and continue for about 130 days.

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*The rapid rates of release from the containment are, of course, highly unrealistic but were chosen for computational expediency. If no consideration is given to interdiction, the resulting concentrations would be fairly insensitive to the rate of release to the ground.

After the initial holdup in the ground, the wave front of the Sr-90 and Cs-137 would reach the river in 23 years and 284 years and would continue to enter the river over periods of about 8.5 and 140 years, respectively. By the time the Sr-90 reached the river, radioactive decay would have reduced the total available activity to 2660 Ci. When the Cs-137 reaches the river, its total activity will have decayed to 416 Ci. The maximum annual average concentration during a given year would be close to the peak concentrations for these two nuclides.

The release of Sr-90 and Cs-137 to the Susquehanna River would continue over a long period if left unchecked. Since doses are usually calculated over a period of at least one year, the logical choice of a stream flow would be the reciprocal mean flow of 12,600 cubic feet per second (the derivation of this value is discussed in Section 3.4.1). All downstream drinking water users on the Susquehanna River are located far enough downstream that total mixing of the effluent across the channel would be expected. Travel time to downstream users would be negligible compared to that of the groundwater pathway. The peak radionuclide concentrations in the Susquehanna River, based on the annual average flow, would be 4.05×10^{-8} μCi/mL for Sr-90, 5.1×10^{-10} μCi/mL for Cs-137, and 5.2×10^{-7} μCi/mL for tritium. Furthermore, the peaks of these three radionuclides would occur at different times because of sorption. Maximum permissible concentrations (MPC) for unrestricted drinking water from 10 CFR Part 20 are 3×10^{-7} μCi/mL for Sr-90, 2×10^{-5} μCi/mL for Cs-137, and 3×10^{-3} μCi/mL for tritium. The calculated river concentrations are thus orders of magnitude below MPC.

4.5 ENVIRONMENTAL IMPACTS

4.5.1 Occupational Doses

During the decontamination of the reactor building and primary cooling system, equipment vital to monitoring and control of the reactor may be serviced and repaired. Tasks such as the repair of the malfunctioning low neutron flux level instrument channel and servicing of the building fan coolers will result in substantial doses to workers. The details of these tasks and their associated occupational doses are discussed in later sections. Only those tasks that are directly related to monitoring the condition of the reactor and its attendant systems will be addressed in this section.

The operations associated with the maintenance of the reactor in safe condition have included monitoring the primary system for boron concentration, water chemistry, temperature, and pressure; monitoring of the neutron level in the core; and monitoring of the temperature and pressure in the reactor building. The temperature, pressure, and neutron-level readings can be made from the reactor control room. To measure the boron level and water chemistry, weekly samples of the primary coolant have been taken and analyzed.

The radiation levels in the reactor control room are essentially at background levels, and personnel in the control room would not be expected to incur an occupational dose. In sampling and analyzing the primary coolant, the occupational dose has been about 20 to 30 mrem per sample.

For purposes of calculating occupational dose, it is assumed that only one person carries out the sampling and analyzing of the primary system water and that the sampling frequency is once per week. Using the higher dose of 30 mrem per sample, the dose in 13 weeks (one quarter) would be 390 mrem.

The maintenance activities would be needed until the primary system is defueled. This is estimated to take three to five years. Assuming the longer time of five years, an accumulative occupational dose of about 8 person-rem can be expected. The added probability that the individual worker would die of cancer from this dose would be 1 in 950. The probability of an additional genetic effect in the offspring of the worker would be 1 in 480.

The operation of a liquid radwaste treatment system for routine operational liquid wastes (ROLW) will result in some slight increase in occupational doses. Based on past experience with similar systems at both TMI and other locations, it is estimated that this dose will be significantly less than one (1) man-rem per year.

4.5.2 Offsite Doses

The dose estimates presented here for maintaining the reactor in a safe condition are based on Kr-85 source terms described in Section 4.4.1. On the basis of historical data described in that section, releases of Kr-85 are assumed to be made at the level of 67 Ci/month, which is equivalent

to an annual level of approximately 800 Ci/year. The estimate of total-body dose to the maximally exposed individual offsite due to release of this amount of Kr-85 is expected to be less than 10^{-3} mrem. The 50-mile total-body population dose for this realease is estimated to be 3×10^{-2} person-rems. These individuals and population doses are based on the external exposure pathway. Lung doses are expected to be higher than total-body doses by approximately a factor of 3, and skin doses are expected to be higher by a factor of about 80 over total-body doses.

The dose estimates from the release of treated routine operational liquid wastes are based upon the source terms discussed in Section 4.4.1.2. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The dose estimates to the maximally exposed individual are 0.0015 mrem per year total body dose and 0.002 mrem per year bone and liver dose. The downstream total body population dose is estimated to be less than 0.02 man-rem per year.

4.5.3 Postulated Accident Effects

No accident having immediate offsite consequences has been postulated. However, if a recriticality accident were to occur, cleanup of the resulting contamination would be necessary. This cleanup effort would probably cause effects similar, although smaller, to those estimated in the remainder of this PEIS.

4.5.4 Psychological-Socioeconomic Effects

Maintenance of the reactor in a safe shutdown condition will continue until the primary system is defueled. The staff estimates the period of this maintenance to be from three to five years. For the majority of people in the community, maintenance of the reactor in a safe condition will be understood to be an essential procedure in the process of decontaminating Unit 2, and will not exacerbate existing anxiety levels. Some members of the community may believe that the possibility of accidents involving offsite releases will increase with any delay of cleanup. For others, the long duration of this required maintenance will lead to the conclusion that maintenance will actually be long-term waste disposal on the site. In either case, the length of the maintenance period is expected to increase the probability of chronic anxiety. The staff is unable to ascribe any other resulting socioeconomic impacts to maintenance procedures.

The release of treated routine operational liquids from TMI Unit 2 may result in increased anxiety among members of the public who utilize downstream water. The impact may be insignificant if the public understands that the treatment and controlled discharge of these wastes is routine at a normally operating nuclear power reactor, e.g., Peach Bottom. The impacts may be greater if large segments of the public erroneously believes that the discharge is a release of accident water. The psychological-socioeconomic effects of release of processed accident water are discussed in Section 7.2.5.5.

Reactor core accidents hypothesized by the staff fall into two broad categories: (1) overheating or mechanical damage leading to an escape of radioactive fission products presently in the core, and (2) the inadvertent restart of the reactor with the associated generation of new fission products and heat. In both instances, the physical effects would be limited to the reactor building and would not impact offsite areas or population. Accidents during the maintenance period, although not directly affecting the public, would nonetheless confirm for a segment of the community the inability of Met-Ed and the NRC to manage the technical process of cleanup. This perception had previously been identified in postaccident studies as being an important cause of stress, therefore, any accident would likely lead to a measurable increase in community stress.

References--Section 4

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3. Memorandum from D. Cokinos to D.J. Diamond, "Recriticality Calculations for TMI," Brookhaven National Laboratory, May 18, 1979.

4. E.W. Barr et al., "TMI-2 Post-Accident Criticality Analysis," General Public Utilities Service Company, TDR-049, August 31, 1979.

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7. Memorandum of D.J. Malloy to J.B. Heineman, "Some Theoretical Limiting Constraints for TMI-2 Recriticality," Argonne National Laboratory, April 4, 1980.

8. C.J. Mueller, Argonne National Laboratory, Memorandum to Distribution, "Review of Draft - An Evaluation of the Potential for and Consequences of Recriticality During Cleanup and Defueling at TMI-2," November 18, 1980.

5. BUILDING AND EQUIPMENT DECONTAMINATION

The interiors of the auxiliary and fuel handling buildings (AFHB) and of the reactor building (RB), as well as the equipment in those buildings, were severely contaminated by radioactive material during and shortly after the accident on March 28, 1979. The auxiliary building contains tanks, pumps, piping, and other equipment used to process and store water for the reactor and primary cooling system and to treat radioactive wastes. The fuel handling building, which is adjacent to the auxiliary building and separated from it by a common wall, contains fuel handling and storage equipment. The general layout of these buildings is shown in Figures 0.1 through 0.3 of Appendix 0. The reactor building contains the reactor pressure vessel (RPV), which holds the fuel elements, and the primary system (steam generators, reactor cooling pumps, pressurizer tank, and associated equipment).* The building is a cylindrical structure about 170 ft high and 140 ft in diameter with reinforced concrete walls 4 ft thick.

The radioactive contamination in the AFHB and reactor building must be removed and disposed of in a safe manner in order to permit defueling and maintenance work that may be needed to ensure continued safe shutdown. In this section, the procedures for decontaminating the buildings are outlined, and the effluents and releases to the environment, as well as the environmental impacts that might occur as a result of these decontamination activities, are discussed.

The decontamination operations may be divided into two phases. The objective of the initial phase is to reduce radiation levels to an extent that the reactor fuel can be removed. This initial phase is necessary regardless of whether the facilities are decommissioned or refurbished for reuse, and the manner in which it must be carried out is largely independent of the final disposition of the facilities. In the final phase, decontamination operations will continue until radiation levels are as low as reasonably achievable (ALARA). The manner in which the final phase is implemented will depend on whether the facilities are decommissioned or refurbished (see Sec. 2.1).

The target radiation field and surface contamination levels are given in Sections 5.1 and 5.2 for the AFHB and reactor building, respectively.

Decontamination of the AFHB started shortly after the accident and was about two-thirds complete as of September 1, 1980. Decontamination activities since that time have been limited due to the licensee's financial situation (see Sec. 1.3). Decontamination of the reactor building had not yet started.

Precautions must be taken to protect the workers from excessive exposure to radiation and to avoid releases of contaminated material that might adversely affect the environment or endanger public health and safety. The special clothing and breathing apparatus used to protect workers from radioactive contamination are shown in Figures 5.1 and 5.2, which are scenes from actual decontamination operations in the auxiliary building. The amount of radioactive contamination decreases with time even if nothing is done. The rate depends on the half-life of the radionuclide. For two of the commonly occurring radionuclides, Cs-137 and Sr-90, this decrease is small, less than 3 percent per year. It is somewhat greater for Cs-134, about 29 percent per year, and quite large for Sr-89, which decreases 0.1 percent of its original amount in a period of a year.

The sequence of decontamination operations that must be carried out may be summarized as follows. Debris and heavy deposits, such as sludge, must be removed; then the exposed surfaces must be cleaned. Loose dirt can be removed by vacuuming or by hosing down the surfaces. Dirt that adheres too firmly to be removed by these means can be removed by high-pressure water jets or by scrubbing. Wiping of surfaces with specially designed wipes (similar to household dust cloths)

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*Removal of the fuel from the reactor pressure vessel (defueling) and decontamination of the inside of the primary system are treated in Section 6.

Figure 5.1. Wet Mopping of TMI-2 Auxiliary Building. Wet mopping is one technique used to decontaminate portions of the Auxiliary Building. Workers are shown wearing special clothing and breathing apparatus for protection from radioactive contamination. (Official TMI Photo)

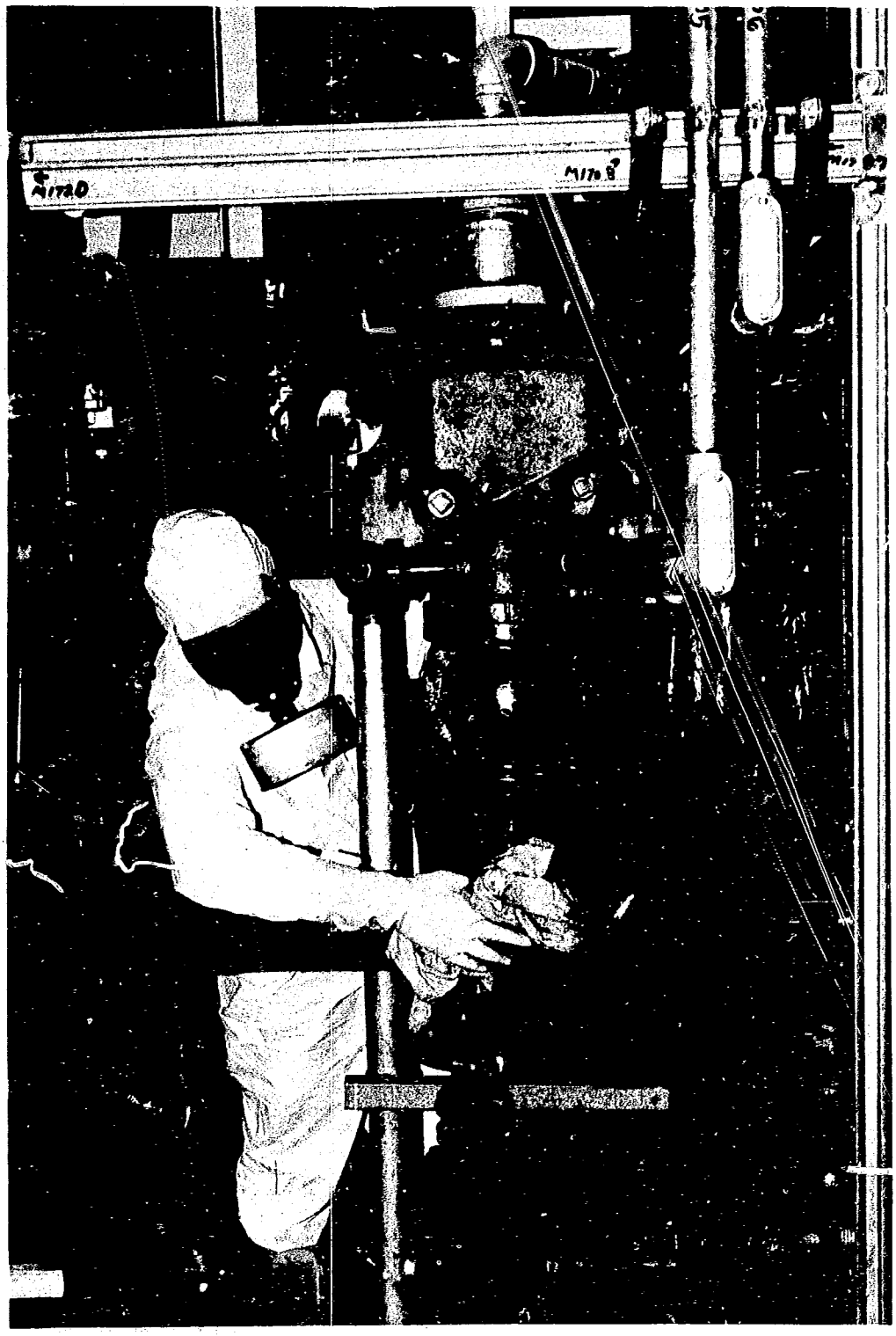

Figure 5.2. Auxiliary Building Equipment Cleanup. In this photo, a worker follows typical procedures for decontaminating equipment and piping systems in the TMI-2 Auxiliary Building. Protective clothing and breathing apparatus are worn as safety precautions. (Official TMI Photo)

also is a commonly used procedure. Finally, for contamination that has penetrated into the surface, like a stubborn stain, methods that remove some of the surface are necessary.

5.1 AUXILIARY AND FUEL HANDLING BUILDINGS

The overall objectives of the AFHB decontamination effort are to permit access without restriction because of surface or airborne contamination, reduce radiation exposure from gamma sources to ALARA levels, and prevent recontamination from other activities or system leaks. The following guidelines are applied to determine whether these objectives have been satisfied:

- Smearable $\beta + \gamma$ surface contamination is less than 1000 dpm/100 cm^2,[1]
- Airborne contamination is less than the 10 CFR Part 20 limits for restricted areas,[2] and
- General radiation levels are at plant design values--generally 0.4 mR/hr.

These objectives are largely determined by the need to use the equipment in the AFHB for defueling operations and are, therefore, applicable regardless of whether the facilities are decommissioned or refurbished. It is expected that they will be attained before defueling operations begin in the reactor building.

In addition to these general objectives, there also is a need to clean certain systems and items of equipment that may be needed in the efforts to decontaminate the reactor building and the primary coolant system. These AFHB systems and equipment include:

- Reactor bleed holdup tanks and associated equipment
- Reactor coolant pump water cooling and seal water systems
- Miscellaneous waste holdup tanks and associated equipment
- Coolant evaporator system
- Degasification system
- Fuel transfer ports and machines
- Fuel storage pool and handling cranes
- Canal cooling and purification systems
- Other handling equipment, such as cranes
- Make and purification demineralizers and filter.

Decontamination of the AFHB was started by May 1, 1979, and is expected to be far enough along by the time that decontamination of the reactor building begins to permit use of these systems (see Fig. 1.4). Decontamination work is expected to continue at a reduced level of effort through the summer of 1982.

5.1.1 Status and Specific Considerations

AFHB decontamination operations considered in this section are (1) cleaning of radioactive material from the interior surfaces of the building and from the surfaces of equipment, and (2) removal of radioactive sludge deposited in tanks and pipes and in the building sump.

5.1.1.1 Building and Equipment Surfaces

Surveys of the general access areas (corridors and normally nonrestricted areas) shortly after the accident showed radiation levels of 150 to 500 mR/hr in the fuel handling building and 50 to 5000 mR/hr in the auxiliary building. At certain locations in the general areas (referred to as "hot spots") and in individual areas (cubicles) containing contaminated filters, demineralizers, tanks, and pumps the radiation levels were much higher--hot spots up to 125 R/hr were measured in the access area of the auxiliary building and levels exceeded 1000 R/hr in some cubicles, such as the reactor coolant bleed holdup tank cubicles.[3]

When cleanup operations began, about 510,000 ft^2 of surface in the AFHB required decontamination. As of September 1, 1980, about 340,000 ft^2 had been decontaminated, with an expenditure of 500,000 person-hours of effort by personnel with assignments that required entry into contaminated areas in the AFHB. The radiation dose received by these workers during this interval was about 250 person-rem. These totals include support personnel (e.g., radiation technicians and maintenance and construction workers) as well as decontamination workers.

The amount of radioactive contamination removed from the AFHB prior to September 1980, excluding only the radionuclides in the accident water, was estimated by the licensee to be 128 Ci. The radioactivity in some of the more highly contaminated cubicles is not known; on the basis of the limited data available, the staff estimates that the radioactive contamination remaining in the AFHB on September 1, 1980, consisted of less than 60 Ci in the form of plateout and deposits on building and equipment surfaces and less than 9000 Ci in the form of sludge in the auxiliary building sump and tanks.

Decontamination of building and equipment surfaces initially was started in areas with low radiation levels. The decontamination team left many high-radiation areas, such as the reactor coolant bleed tank cubicles, until later because it is first necessary to remove the radioactive liquids from the tanks and piping. After these tanks and pipes are flushed and the filters changed, radiation levels are much lower and decontamination of surfaces can proceed with much less radiation exposure to personnel.

The general access areas in the AFHB had been decontaminated by April 1980; however, some of those areas have to be recleaned periodically because of recontamination. For example, installation of new equipment needed for various tasks has resulted in some contaminated material being tracked into previously decontaminated areas. In addition, leakage of barrels containing contaminated industrial detergent at one time was causing similar problems, but this has been corrected. The movement of contamination has required the routine monitoring of previously decontaminated areas and repeated decontamination of some areas. It is expected that some additional decontamination will be needed periodically in the general access areas (corridors, stairwells, etc.) because of airborne dispersion and tracking by workers from areas still undergoing decontamination. Another factor affecting decontamination efforts is the increased potential for development of leaks from systems that have not received routine maintenance because of contamination. This problem will become more acute with time and is continuing to receive attention.

Decontamination of the cubicles in the AFHB began in February 1980 and is expected to continue through the spring of 1982; after then a reduced level of effort will be needed to maintain the decontaminated levels. The cubicle areas tend to be more difficult to decontaminate because of special shielding requirements and higher contamination levels. As of September 1, 1980, many of the tanks, filters, and much of the piping in the AFHB still contained radioactively contaminated liquids. As indicated in Appendix O, of the 57 cubicles requiring decontamination, 6 had been completely decontaminated (except for fluid transfer and filter changes in 2 of the 6), 32 had been partially decontaminated, 15 required only light decontamination, and 6 had received no decontamination. The makeup and purification demineralizer cubicles are likely to be the most severely contaminated in the AFHB. The contamination consists primarily of radionuclides that were deposited in the inline filters and demineralizer resins during and immediately following the accident. Based upon gamma surveys by the licensee, each of the two demineralizers could contain about 2000 Ci. Transfer of fluids, changing of filters, removing of ion exchange resins, and/or flushing of lines still were needed in 22 of the 57 cubicles.

The completion date for decontamination of the AFHB will depend on the number of persons assigned to the effort. This number was severely reduced during September 1980, to the point that very little work beyond essential maintenance was being done. A reliable estimate for the completion date cannot be given until steps are taken to resolve financial difficulties that are (as of January 1981) limiting expenditures for decontamination work. An estimate of the work effort that will be required to complete the AFHB decontamination is given in Section 5.1.3.2.

5.1.1.2 Sludge

Some of the contamination in the auxiliary building is in the form of sludge, which requires special attention because of its bulk and high radioactivity content. The sludge is mostly cement residue (from an unfinished part of the floor and from floor drains and drain piping) with an admixture of ion-exchange resin from small amounts of resin carried over during demineralized water system flushing operations prior to the accident. During the accident, which caused flooding to a depth of about 6 inches on the lower level of the building, a large portion of the sludge was washed into the 9000-gallon sump. From there the sludge was pumped into the sump tank and the miscellaneous waste holdup tank (see Figs. 0.1 through 0.3, Appendix O, for the locations of these tanks). Smaller amounts of sludge may be present in the other tanks in the auxiliary building, but this has not been confirmed by direct examination. Pumps, valves, and lines also may contain small amounts of sludge. The estimated total volume of wet sludge in the tanks and

the sump in the auxiliary building is about 150 ft^3. For sludge with a density of 1.6 g/mL (100 lb/ft^3), the total weight of wet sludge to be removed is on the order of 15,000 pounds.* The total radioactivity in the sludge is estimated to be about 9000 Ci.

The sump is 14 ft wide by 24 ft long. It is 3 ft deep at one end and slopes to 4 ft deep at an outlet at the other end. The sludge is about 2 inches deep near the outlet pipe. At the opposite (shallow) end it is 4 to 6 inches deep. The total volume of sludge in the sump is estimated to be about 90 ft^3. Debris also are present in the sump. A 6-inch hose, a 2.5-inch firehose, tie rods and a temporary sump pump have been noted.[4] Radiation levels from 3 R/hr to 5 R/hr were measured at different locations in the sump cubicle prior to January 1980. Radiation levels in the sump cubicle can be controlled to some extent by pumping out the contaminated water in the sump and replacing it with processed water from which most of the contamination has been removed. Cleanup of the sump cubicle had not been started as of January 12, 1981.

Estimates by the licensee indicate that the miscellaneous waste holdup tank contains 10 to 15 ft^3 of sludge. This tank is being used as a feed tank for the EPICOR II water decontamination operations (see Sec. 7). During the period August 1979 to April 1980, the radiation level near the tank rose from about 5 R/hr to about 30 R/hr because of the accumulation of sludge as contaminated liquids were transferred in and out of the tank. The continuing need to use the tank may delay its desludging and decontamination.

The volume of sludge in the sump tank also is quite uncertain because the interior is not visible; it is thought by the licensee to be on the order of 5 to 10 ft^3. The radiation level at one end of the tank is about 20 R/hr. The total sludge in the remaining tanks is estimated by the licensee to be about 35 ft^3 or less. These tanks include the three reactor coolant bleed tanks (which constitute about three-quarters of the total liquid storage capacity in the auxiliary building and probably contain about 10 ft^3 of sludge per tank[4]), the spent resin tank, the concentrated waste tank, and other tanks identified in Figures 0.1 through 0.3 of Appendix 0. Desludging of a tank cannot begin until most of the liquid within it has been processed or removed. Liquid from reactor coolant bleed tanks A, B, and C has been processed, and flushing the tanks with processed water has decreased the radiation levels and removed some of the sludge.

5.1.2 Methods Used and Alternative Methods Considered

5.1.2.1 Methods Used for Building and Equipment Surface Decomtamination

Appropriate combinations of well-known methods have been used in the decontamination of building and equipment surfaces in the AFHB. The following methods have seen at least limited use, with the first six used predominantly:

- Removal of all nonessential items, such as wood, tools, hoses, cords, and loose equipment.
- Dry vacuuming of dry floors and equipment (piping, valves, cable trays, etc.) with a high efficiency particulate air (HEPA) filter on the vacuum exhaust.
- Low-pressure hot water misting and washing.
- Using high-pressure water jets on floors, tanks, piping, and valves.
- Wet vacuuming of industrial detergent after hand scrubbing.
- Manual wiping with disposable towels or oil-impregnated wipes.
- Removing strippable coatings on floors, walls, portable shields, and other surfaces.
- Electrochemical decontamination of tools and small equipment.
- Freon cleaning and Freon ultrasonic cleaning of electrical equipment and tools.
- Concrete removal by means of pneumatic impact devices.

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*The volume of dewatered sludge is smaller than the volume of wet sludge by a factor of about 4 to 5; the weight is smaller by a factor of about 2.

All of these methods, except perhaps the water jet, require the work crew to be relatively close to the contamination. In areas of high concentrations of airborne radioactive particulates, personnel must wear respiratory protectors, such as air packs or filter respirators. Workers also must wear special clothing for protection and to control the spread of contamination.

Various combinations of decontamination methods are selected by decontamination personnel to minimize exposure and maximize effectiveness. The decontamination personnel rely on experience and testing to determine the best method to use for each particular task.

5.1.2.2 Methods Used and Considered for Sludge Removal

The procedure now being used for removing sludge from tanks and piping in the AFHB starts with replacement of the inlet and outlet filters, as required. The system then is flushed several times, and the loose sludge remaining in a tank is collected with a portable recirculating vacuum filter system (RVFS) with disposable filter cartridges. Sludge caked to the walls of tanks can be dislodged with a high-pressure water jet and then collected in the RVFS. Standpipes (drain pipes that extend above the bottom of the tank) may complicate the desludging effort.

The alternatives considered by the staff for removing sludge from the AFHB sump and tanks may be divided into the following categories:

- Dissolve the sludge by chemical means so that it can be removed as liquid waste,
- Remove the sludge by mechanical means (scooping and scraping), or
- Resuspend the sludge so that it can be removed as a slurry.

Dissolution of the sludge, which consists largely of calcium sulfate and various silicates, would require strong acids, such as concentrated hydrochloric (muriatic) acid, that would increase the hazards for decontamination workers and damage the underlying surfaces and the drain hardware without providing any significant advantages. The process is not efficient and usually leaves an intractable residue; there are no practical means for dissolving cement, the primary constituent of the sludge, on a large scale. Dissolving the sludge would greatly complicate subsequent problems of processing the water to remove dissolved radioactive contamination for disposal as solid waste. This approach was not, therefore, considered further.

If mechanical removal were done by hand using shovels and scrapers, the radiation exposure to the workers would be higher than for the other alternatives. Mechanical devices for scooping and scraping that would permit semiremote or remote operations in restricted spaces are not, insofar as the staff is aware, available as standard equipment and thus would have to be designed, constructed, and tested. This design and development effort could lead to delays of unpredictable duration. Other methods, such as resuspension (including hose washing), are simpler and more effective and require no development work; hence, mechanical removal methods were not considered further.

The following alternative methods for removal of the sludge by resuspension have been considered:

- Resuspension and removal of the sludge by means of a portable RVFS.
- Resuspension by flushing and/or backflushing and entrapment of the sludge on the inline filters using existing piping and pumps.

The second resuspension alternative is very effective for tanks containing small amounts of sludge. The worker radiation dose that would be incurred in changing the inline filters on which the sludge was collected would be essentially the same as for the RVFS system. However, the agitation would be less than for the first alternative; hence, sludge removal would be less effective. Additionally, there would be a greater likelihood of spreading the sludge to other parts of the system. A high-pressure water jet also could be used to loosen and remove caked sludge from surfaces and, if necessary, to assist in resuspending the sludge when agitation from recirculation or from the vacuum system is insufficient for this purpose.

The water remaining after removal of the sludge from the slurry could either be returned to the tanks and sump or stored in some other tank until it could be processed to remove dissolved radionuclides.

5.1.3 Details of Feasible Methods and Facilities

5.1.3.1 Details of Methods Used

Application of the decontamination procedures listed in Section 5.1.2.1 proceeds in a logical sequence from removal of tools, equipment, and other loose items and debris to methods for removing loose contamination that does not adhere strongly to surfaces (for which spraying and washing is effective), to methods for removing fixed contamination that adheres to the surface so strongly that scrubbing or chemical solvents are needed. Some of the methods used for decontaminating surfaces in the AFHB are described below. Further discussion of decontamination methods may be found in Section 5.2.3.1.

Removal of Nonessential Items

The decontamination crews have found tools, loose equipment, barrels, boxes, staging, cables, hoses, wood pallets, and other miscellaneous items in many areas. Rather than decontaminate these items in place, the crews have moved them to staging areas for cleaning and eventual storage or disposal. Some of the items have been disposed of as low-specific-activity waste. Most of the items can be handled by cleanup personnel. Some items with fixed contamination have been stored for future use in contaminated areas. Large pieces of equipment may be decontaminated in place. Worker exposure comes from the contaminated items when they must be moved, as well as from the contaminated environment.

Dry Vacuuming of Dry Floors and Equipment

In areas where dry dust has accumulated, dry vacuuming can be effective. Dry vacuuming does not work well on crusted deposits; therefore, it is used primarily in areas where dust has not been wetted or crusted. The vacuuming involves the use of a specially equipped machine with a HEPA filter in the exhaust stream. Radioactive particles are retained in the filters. Worker exposure from airborne activity may be increased by the vacuuming activity even though most of the particles are picked up by suction. Dry vacuuming has been used in the AFHB only for removing the normal accumulation of dust from equipment that was above the flood line and did not, therefore, become contaminated during the accident.

Low Pressure Water Washing

Low pressure hot water misting and washing is used to dissolve chemicals on the walls, floors and components. This method is effective in dissolving dried boron deposits and controlling airborne contamination. It is used prior to the high pressure water jet spray.

Use of High-Pressure Water Jet

A high-pressure, low-flow-rate, water-jet spray system can be used to remove surface contaminants. The system is effective and fast in removing contaminants and reducing personnel exposure time. Large areas can be cleaned quickly and thoroughly.

Two water-jet units are available onsite at TMI-2 for decontaminating the AFHB. One operates at a moderate pressure of 1200 psi and the other at a high pressure of 10,000 psi. The water flow is relatively low, about 7 gpm for the 1200 psi unit. As of January 2, 1981, only the moderate pressure jet had been used for decontaminating the AFHB.

Use of the water jet was limited prior to March 1980 because of restricted water usage. It is now being used extensively with water processed by EPICOR II. Use of this processed water, rather than additional water pumped into the facility, has reduced the water inventory buildup and permits extensive use of the water jet. Tests of the atmospheric contamination caused by use of processed water have been conducted with the water jet in closed quarters. Under the worst conditions (i.e., with the atmosphere saturated with water vapor), the concentration of tritium in the air would be below the level specified in 10 CFR Part 20 as the safe limit for workers.[2]

Wet Vacuuming

Wet vacuuming has been the primary method for decontaminating areas where contaminants adhere tightly to surfaces. The method involves scrubbing with water and industrial detergents and then wet vacuuming the resulting solution. The wash solution is stored in barrels until it can be solidified for disposal. There have been limitations on the types of cleaning compounds used

since their effect on the solidification process has not been fully evaluated. To date, only one commercial detergent has been used. The scrubbing is a slow and tedious process that brings workers into close contact with contaminated wash solution.

Manual Wiping

Manual wiping, a worker-intensive technique, may be used to remove dust and accumulated contamination that cannot be vacuumed. Disposable towels or oil-impregnated wipes ordinarily are used. This technique requires workers to be close to contaminated areas; thus the exposure may be higher than for other techniques, such as vacuuming, for a given level of contamination. Manual wiping is normally used only after radiation levels have been reduced by gross decontamination using other methods. This method is also used for contolling contamination in areas previously contaminated.

Removing Strippable Coatings

This method involves the application and subsequent removal of a strippable coating. As the coating is removed, it takes with it the surface contamination. Strippable coatings are useful on portable shielding, making it easy to decontaminate. It involves close worker proximity to contamination, but there is less likelihood of the contamination being spread to other surfaces, such as clothing or gloves, than for manual wiping. Strippable coatings are commonly applied to decontaminated areas to facilitate subsequent decontamination if recontamination should occur.

Electrochemical Decontamination

Electrochemical decontamination is an electropolishing procedure used on metal objects to remove a thin layer of the exterior surface and attached contamination. The method employs a tank containing an acid solution and a low-voltage, high-current source. At TMI this method can be used only for small objects because the electrochemical decontamination tank is about 3 ft by 3 ft by 2 ft deep. Small tools and parts can be cleaned in a very short time.

The phosphoric acid in the decontamination tanks is recirculated through a filter that accumulates much of the contaminated solids removed from the surface. The major limitations of this method are that the objects must be metal, they must be small, and they must be removable.

The electrochemical decontamination apparatus is set up in a facility removed from the AFHB. Vapors from the facility are circulated through HEPA filters to limit radioactive releases to negligible amounts.

Freon Cleaning and Freon Ultrasonic Cleaning

This decontamination method involves the use of Freon, either by spray, brush, or ultrasonic bath, to clean electrical tools and small, intricate parts. The ultrasonic bath using Freon as a fluid will remove most contaminants, but only small parts can be cleaned in this manner. Larger electrical components, such as motors and switchgear, can be decontaminated by spray cleaning with Freon. Precautions are taken to ensure adequate ventilation in order to minimize inhalation of Freon vapor by workers. About five gallons per week in the form of vapors are removed by the facility ventilation system. The Freon cleaning and ultrasonic Freon cleaning equipment is located in the same facility as the electrochemical decontamination equipment and uses the same ventilation system.

Concrete Removal

The floors and walls in the AFHB are dense, uncoated concrete. At some locations the contaminated water was in contact with uncoated concrete for up to a year and the contamination diffused into the surface. Removal of a layer of the surface was the only way to eliminate this contamination. This was done by means of a pneumatic impact device. The maximum surface removal required was 3/16 of an inch.

5.1.3.2 Work Effort Required

As of September 1, 1980, about 280,000 person-hours had been expended on AFHB decontamination by decontamination workers, i.e., those workers who plan, prepare for, and carry out the actual decontamination work, such as operating the water jet and scrubbing and wet vacuuming.[5] An

additional work effort of about 220,000 person-hours was expended by other workers with assignments that required entry into the contaminated areas in the AFHB. These assignments included radiation surveys and monitoring by radiation technicians, engineering and construction work, maintenance, and plant operations. The work effort by personnel with assignments that did not require entry into contaminated areas (such as clerical workers, planning staff, etc.) is not included. During early stages of the work, about 30 to 50 percent of the work time of workers with assignments in the contaminated areas was spent working inside the building; the remaining time was spent in putting on and taking off protective clothing and gear, planning, training, and similar activities. The licensee expects, as a consequence of experience and improvements in organization, to be able to maintain a performance level of 60 percent productive in-building work for current and future operations.

Decontamination work had been completed on about two-thirds of the total surface area (about 340,000 ft^2 of a total of 510,000 ft^2) by September 1, 1980. The average work effort per unit area was about 1.5 person-hours/ft^2 during this time. If this ratio does not change during the remainder of the decontamination effort, the total work effort for decontaminating the AFHB by workers with assignments in the contaminated areas will be about 750,000 person-hours. The staff estimates that 40 percent to 50 percent of this time will be spent in contaminated areas inside the AFHB.

Some of the more difficult tasks, such as sludge removal and decontamination of the sump cubicle, have been left until last. The work effort per unit area will be greater for these tasks. However, experience gained during the first 16 months of decontamination work should decrease the work effort per unit area required for decontamination tasks of comparable difficulty. Taking both of these considerations into account, the staff regards the estimate of 750,000 person-hours to be a reasonable one.

The schedule for completing decontamination of the AFHB will depend on the level of work effort. If the work effort by decontamination workers could have been maintained at a level of two 12-hour shifts per day, five days per week, with a work force of 19 persons per shift (as originally planned by the licensee), with a comparable level by other workers with in-building assignments, decontamination could probably have been completed by the end of 1981.[5] However, financial difficulties forced the licensee to reduce the level of effort by decontamination workers to one 8-hour shift per day, five days per week, with a work force of 8 in September 1980. The current level of effort (as of January 1981), which has been maintained since early in September 1980, limits decontamination work in the AFHB to maintenance and decontamination of selected critical areas. The current schedule of the licensee shows a completion date of about September 1982 (see Fig. 1.4). This implies an appreciable increase from the current level of effort some time during 1981.

The number of workers that would, on the average, be needed to provide 250,000 person-hours of work effort over a period from May 1, 1979 to September 1, 1982, assuming that the workers spent an average of 40 hours/week on the job, would be about 85. This is the average number of workers that would have to be employed in order to carry out the work that involves assignments in contaminated areas in the AFHB. The actual number at any time will fluctuate as the level of work effort fluctuates. Support workers who do not have to enter contaminated areas are not included.

5.1.4 Effluents and Releases to the Environment

5.1.4.1 Normal Operations

Effluents and releases associated with decontamination of the AFHB and equipment include solids, liquids, gases, and airborne particulates.

The solid wastes include debris removed during initial cleanup operations, contaminated filters from dry vacuuming operations, solid residue removed from comtaminated liquids by liquid processing operations, and contaminated laundry and materials (e.g., wipes) used for decontamination operations. The activity of the radioactive waste generated by decontamination operations up to September 1, 1980, (not including EPICOR II liners generated by water processing) was estimated to be 128 Ci.[5] Further information concerning releases from solid waste handling may be found in Section 8.

Liquids from use of the water jet from use of the water jet, from wet vacuuming, and from desludging operations will be processed to remove dissolved radioactive contaminants. The processing operations and the alternatives for the ultimate disposition of the processed liquids are discussed in Section 7.

The estimates of concentrations of effluents and releases are based on the licensee's schedule prior to September 1980, which called for completion of AFHB decontamination work by December 1, 1981. According to this schedule, the total elapsed time for decontaminating the AFHB was 30 months and September 1, 1980, was the approximate midpoint in time between the start and finish of the AFHB decontamination work. Any extension of that schedule would lead to a decrease in the concentrations; hence, the values for the release concentrations given below are still valid upper bounds for the concentrations that may be expected under the revised schedule (Fig. 1.3-1), which extends the completion date by almost a year. The total releases are independent of the schedule.

During the period from May 1, 1979, when decontamination of the AFHB had started, to September 1, 1980, when about two-thirds of the 510,000 ft^2 of contaminated surface had been cleaned, the releases of airborne contamination through the AFHB building ventilation systems from decontamination operations did not exceed technical specification limits.* The area that must still be cleaned is equivalent to only half the area decontaminated during the prior 15 months. However, the levels of contamination (see Tables 0.1 through 0.3 in Appendix 0) are higher, so the release rate of airborne effluents might be essentially the same during a subsequent 15-month decontamination period.

Estimates of the normal airborne releases that are expected to occur during a 15-month period following September 1, 1980, are summarized in Table 5.1. The desludging operations are expected to take only about three months; therefore, the calculations of airborne release concentrations for desludging operations are based on this time interval rather than 15 months.

Tritium releases during normal operations are primarily from evaporation of the processed water used for water jet and other washing operations. The ventilation system fans will be on when workers are inside the building; hence water vapor in the building atmosphere will be released to the outside atmosphere when work within the building is in progress. The processed water has a typical tritium concentration of 0.13 μCi/mL. The staff estimates that up to about 2 percent of the water used for water jet operations will be lost by evaporation; this will generate about 0.1 Ci of tritiated water vapor that will be vented to the outside atmosphere by the building ventilation system.

The principal normal airborne releases from surface decontamination operations result from wet vacuuming operations. During wet vacuuming operations, 0.1 percent of the material which may contain up to 60 Ci of plateout and deposits on building and equipment surfaces, is estimated to be released to the building atmosphere. This material passes to the air-cleaning system, which consists of two stages of HEPA filtration.

The staff has based its calculation of airborne releases that might be generated by desludging operations on experience acquired in complex chemical processing of nuclear fuels indicating that about 0.01 percent of the total material processed may become airborne.[6] This estimate is judged by the staff to be high relative to that expected in the desludging operations.

The auxiliary building and the fuel handling building have separate air filtration systems, each consisting of two trains in parallel. Each train consists of four filters in series: a prefilter, a HEPA filter, a charcoal filter, and another HEPA filter. The maximum flow rate for the auxiliary, building is 65,000 cfm; the maximum flow rate for the fuel handling building is 36,000 cfm.

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*The technical specifications limit the releases of I-131 and particulate activity with half-lives greater than eight days in gaseous effluents to an instantaneous release rate of 0.3 μCi/s and to 0.024 μCi/s when averaged over any calendar quarter. Most of the contribution to particulate contaminants from the AFHB is from Cs-137, Cs-134, and Sr-90. The only significant gaseous contaminants are H-3 and Kr-85. The technical specifications limit the instantaneous release rates of H-3 and Kr-85 to 3×10^4 μCi/s and 4.5×10^4 μCi/s, respectively. The technical specifications also limit the release rates of H-3 and Kr-85 to 4.8×10^3 μCi/s and 7.2×10^3 μCi/s, respectively, when averaged over any calendar quarter. When both isotopes occur together, the individual release limits are less (see 10 CFR Part 20, Appendix B).

Table 5.1. Airborne Releases of Principal Radionuclides Expected to Occur from Decontamination Operations in the Auxiliary and Fuel Handling Buildings during a 15-Month Period after September 1, 1980[a]

Radionuclide[b]	Surface Decontamination	Desludging	All
	Concentrations of Releases[c] (μCi/mL)		
H-3	5×10^{-11}	1.9×10^{-12}	5×10^{-11}
Cs-137	2.8×10^{-14}	2.8×10^{-12}	2.8×10^{-12}
Cs-134	4.7×10^{-15}	5.5×10^{-13}	5.5×10^{-13}
Sr-90	3.0×10^{-16}	3.8×10^{-13}	3.8×10^{-13}
Sr-89[d]	3.0×10^{-7}	6.3×10^{-14}	6.3×10^{-14}
	Total Releases (Ci)		
H-3	0.1	4.5×10^{-4}	0.1
Cs-137	5.1×10^{-5}	6.7×10^{-4}	7.2×10^{-4}
Cs-134	8.6×10^{-6}	1.3×10^{-4}	1.4×10^{-4}
Sr-90	5.6×10^{-7}	9.0×10^{-5}	9.1×10^{-5}
Sr-89[d]	5.6×10^{-8}	1.5×10^{-5}	1.5×10^{-5}

[a]Releases during prior decontamination operations (May 1, 1979, to September 1, 1980) did not at any time exceed technical specification limits.

[b]The duration of the release is assumed to be 15 minutes; the flow rate for the building ventilation sysltems are 65,000 cfm for the auxiliary building and 36,000 cfm for the fuel handling building.

[c]The concentrations for surface decontamination operations are calculated averages over a period of 15 months. The concentrations for desludging operations are calculated averages over a period of three months. The concentrations for all operations are those that also would be applicable during the desludging operations. The auxiliary building and fuel handling building have independent ventilation systems and separate HEPA filter systems with maximum flow rates of 65,000 cfm and 36,000 cfm, respectively. The ducts from both systems join into a common duct to the plant stack at a point past the HEPA filter systems.

[d]The releases for Sr-89 are based on estimates of the amounts of Sr-89 present on September 1, 1980. About 99.98 percent of this amount will disappear by radioactive decay by December 1, 1981.

An overall penetration factor of 10^{-3} has been used in calculating the releases shown in Table 5.1.* NRC Regulatory Guide 1.140, which develops guidelines for operating nuclear power-plants, specifies a very conservative penetration factor of 10^{-2} (corresponding to 99 percent efficiency) for the entire exhaust system and then only if the HEPA filters test in-place to an efficiency of 99.95 percent or greater. The Regulatory Guide gives no additional credit for HEPA

*The penetration factor is the fraction of material entering a filter that passes through the filters. Thus, for a given filter, the penetration factor equals 1 minus the filter efficiency when the filter efficiency is expressed as a fraction.

stages in series. Application of these criteria to the cleanup would be unnecessarily conservative in that actual penetrations through a single stage of HEPA filters are much less than one percent (even for particle sizes having maximum penetration) and in that further substantial reductions of penetration are achieved with a second stage of HEPA filters. Using a conservative penetration factor of 0.001 for each of two HEPA stages (1×10^{-6} overall), the values shown in Table 5.1 (except for H-3) would be lowered by a factor of 1000.

5.1.4.2 Accident Scenarios

Possible accidents during surface decontamination activities would include spills, rupture of the HEPA filter attached to a vacuum cleaner, or rupture of a HEPA filter in the building ventilation system. The spilling of liquids inside the AFHB would not result in immediate release to the environment and probably would not result in increases in ultimate releases to the river, to the atmosphere, or as solid waste.

The only pathway to the river for a liquid spill inside the AFHB would be through a leak in the building structure and subsequent percolation through the rock and soil to the river. A hypothetical accident for such a liquid spill and leakage is covered in Section 7.4.2.

The rupture of a HEPA filter in the building ventilation system would increase the airborne release rate described under normal operations (Sec. 5.1.4.1) until the failure was detected (in about 15 minutes[7]) and the ventilation system secured. An analysis of similar failures in a fuel reprocessing plant[7] indicates that 1×10^{-3} of the filter inventory could be released to the airstream. The resultant concentrations of the principal radionuclides released due to rupture of a HEPA filter are listed in Table 5.2.

The accidents that could occur during desludging activities are similar to those that could occur during surface decontamination. Estimates of releases that could be expected as a consequence of desludging operations if a HEPA filter were to rupture are given in Table 5.2.

5.1.5 Environmental Impacts

5.1.5.1 Occupational Doses

From May 1, 1979, through September 1, 1980, about two-thirds of the AFHB was decontaminated. The cumulative whole-body dose received by decontamination workers (the workers who operate the water jets and do the scrubbing, wet-vacuuming, wiping, and similar decontamination operations) was 142 person-rems. The highest dose received by an individual decontamination worker over this period was 2.5 rem. Cumulative and average individual decontamination worker doses are given in Table 5.3.

Other personnel with assignments that required entry and work in contaminated areas in the AFHB included radiation technicians (who survey and monitor radiation levels and accompany each work team to monitor radiation levels and provide assistance if needed) engineers, construction workers, maintenance workers, and operations personnel. The total cumulative dose during the period from April 27, 1979 through September 1, 1980, for all personnel with assignments that required them to enter contaminated areas within the AFHB is estimated by the staff to be about 250 person-rem.

The staff estimates, on the basis of extrapolations from experience during the period from April 27, 1979, to September 1, 1980, that the cumulative dose for decontaminating the remaining one-third of the contaminated surfaces in the AFHB will be in the range of 125 to 300 person-rem, for a total cumulative whole-body occupational dose of 375 to 550 person-rem. The lower bound is based on the assumption that the occupational dose per unit area of surface decontaminated will remain the same. The upper bound is based on the assumption that it will increase by a factor of 3 because some of the more difficult and highly contaminated areas have been left until last. The factor 3 is based on an extrapolation of the trend in the average dose per worker per month.

The expected number of additional cancer mortalities in a work force exposed to a cumulative whole-body dose in the range of 375 to 550 person-rem is 0.05 to 0.07 for a work force of 85 individuals. The average added probability that an individual worker would eventually die from cancer would range from about 1 in 1200 to 1 in 1700 if he/she participated in the AFHB decontamination work for the full duration of the cleanup effort. The expected number of additional genetic effects in the offspring of the exposed workers ranges from 0.10 to 0.14.

Table 5.2. Airborne Releases from HEPA Filter Failure during Decontamination of the AFHB

Radionuclide[a]	Releases from HEPA Filter Failure	
	Surface Decontamination	Desludging
	Concentrations of Releases[b] (μCi/mL)	
Cs-137	1.2×10^{-9}	2.4×10^{-8}
Cs-134	2.0×10^{-10}	4.7×10^{-9}
Sr-90	1.3×10^{-11}	3.3×10^{-9}
Sr-89[c]	1.3×10^{-12}	5.4×10^{-10}
	Total Releases (Ci)	
Cs-137	5.1×10^{-5}	6.7×10^{-4}
Cs-134	8.6×10^{-6}	1.3×10^{-4}
Sr-90	5.6×10^{-7}	9.0×10^{-5}
Sr-89[c]	5.6×10^{-8}	1.5×10^{-5}

[a]Other fission products are present; however, on the basis of analysis of dissolved radionuclides in the reactor building sump water, for which Sr-89, Sr-90, Cs-134, and Cs-137 constitute greater than 99 percent of the MPC contribution (see Appendix J), and which has a radionuclide distribution similar to that of the AFHB sump water, the contributions of each of the other radionuclides to airborne releases are expected to be less than 0.1 percent and have not, therefore, been listed.

[b]The duration of the release is assumed to be 15 minutes; the flow rate for the building ventilation systems are 65,000 cfm for the auxiliary building and 36,000 cfm for the fuel handling building.

[c]The releases for Sr-89 are based on estimates of the amounts of Sr-89 present on September 1, 1980. About 99.98 percent of this amount will disappear by radioactive decay by December 1, 1981.

Table 5.3. Whole-Body Occupational Doses Received by Workers during Decontamination of the Auxiliary and Fuel Handling Buildings[a,b,c]

Period	Average Dose per Worker (mrem)	Number of Workers	Cumulative Dose[d] (person-rem)	Number of Months per Period	Average Dose per Worker per Month (mrem)
April 27 - June 30, 1979	158	183	28.9	2	79
July 1 - September 30, 1979	104	192	20.0	3	35
October 1 - December 31, 1979	121	174	21.1	3	40
January 1 - February 26, 1980	182	82	14.9	2	91
February 27 - April 28, 1980	365	52	19.0	2	183
April 29 - September 1, 1980	958	40	38.3	4	240
Highest individual dose - 2.5 rem					
Total cumulative dose - 142 person-rem					

[a]Memo from W.K. Lehto to J.H. Opelka, Argonne National Laboratory, "Compilation of Information and Data from Visit to TMI-2 on May 6, 1980," May 13, 1980.

[b]From T.R. Block, "Auxiliary and Fuel Handling Building Decontamination Program," presented at an information briefing by Met-Ed Co. at Crawford Station, Middletown, PA, September 23, 1980.

[c]The doses listed are for decontamination workers only; radiation technicians, engineers, and maintenance and operations personnel are not included. The total cumulative dose for all personnel with assignments that required entry into the contaminated areas of the AFHB is estimated by the staff to be 250 person-rem.

[d]Cumulative doses for each period; the total cumulative dose is the sum of the numbers in the column.

5.1.5.2 Offsite Doses

The estimates for offsite doses to an individual subjected to the maximum exposure from AFHB decontamination and desludging are listed in Table 5.4. These estimates are based on the source terms developed in Section 5.1.4.1 and listed in Table 5.1. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3. The total-body population dose received by the human population within a 50-mile radius from normal AFHB decontamination activities is estimated to be 0.02 person-rem.

Table 5.4. Dose Estimates for the Maximum Exposed Individual for AFHB Decontamination and Desludging Activities

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	4.5×10^{-5}	5.3×10^{-4}	4.6×10^{-5}
	Ground Shine	8.8×10^{-5}	8.8×10^{-5}	8.8×10^{-5}
	Vegetable Use	3.1×10^{-3}	1.3×10^{-2}	2.0×10^{-3}
	TOTAL	3.2×10^{-3}	1.4×10^{-2}	2.1×10^{-3}
Nearest milk goat	Inhalation	5.4×10^{-5}	2.0×10^{-4}	3.0×10^{-5}
	Ground Shine	8.4×10^{-5}	8.4×10^{-5}	8.4×10^{-5}
	Goat Milk Use	1.5×10^{-3}	1.1×10^{-2}	1.1×10^{-2}
	TOTAL	1.6×10^{-3}	1.1×10^{-2}	1.1×10^{-2}
Nearest cow and garden	Inhalation	4.9×10^{-5}	5.9×10^{-4}	5.1×10^{-5}
	Ground Shine	1.3×10^{-4}	1.3×10^{-4}	1.3×10^{-4}
	Vegetable Use	4.5×10^{-3}	1.9×10^{-2}	2.9×10^{-3}
	Cow Milk Use	5.7×10^{-4}	2.9×10^{-3}	2.2×10^{-3}
	TOTAL	5.2×10^{-3}	2.3×10^{-2}	5.3×10^{-3}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups: adults, teenagers, children and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden are for children, and the dose estimates for the nearest milk goat are for adults per total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 mile north, and nearest cow and garden = 1.05 mile east.

5.1.5.3 Postulated Accident Effects

The type of accident for which dose estimates are made here is rupture of a HEPA filter. This accident scenario is described in Section 5.1.4.2, and the source terms are listed for it in Table 5.2. Doses are estimated here for a HEPA filter failure occurring during the surface decontamination operation and during desludging operations. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The dose estimates for individuals subjected to the maximum exposure are listed in Tables 5.5 and 5.6. The significance of these doses is discussed in Section 10.4.

Table 5.5. Estimates of Offsite Doses to the Maximum Exposed Individual Caused by Failure of a HEPA Filter During Surface Decontamination Operations in the AFHB

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	2.9×10^{-5}	1.0×10^{-4}	4.7×10^{-5}
	Ground Shine	3.5×10^{-4}	3.5×10^{-4}	3.5×10^{-4}
	Vegetable Use	2.4×10^{-3}	1.2×10^{-2}	7.8×10^{-3}
	TOTAL	2.8×10^{-3}	1.2×10^{-2}	8.2×10^{-3}
Nearest milk goat	Inhalation	2.9×10^{-5}	1.0×10^{-4}	4.7×10^{-5}
	Ground Shine	3.5×10^{-4}	3.5×10^{-4}	3.5×10^{-4}
	Goat Milk Use	5.7×10^{-3}	4.0×10^{-2}	4.7×10^{-2}
	TOTAL	6.1×10^{-3}	4.0×10^{-2}	4.7×10^{-2}
Nearest cow and garden	Inhalation	2.0×10^{-5}	6.9×10^{-5}	3.2×10^{-5}
	Ground Shine	3.5×10^{-4}	3.5×10^{-4}	3.5×10^{-4}
	Vegetable Use	2.4×10^{-3}	1.2×10^{-2}	7.8×10^{-3}
	Cow Milk Use	1.4×10^{-3}	6.0×10^{-3}	6.0×10^{-3}
	TOTAL	4.2×10^{-3}	1.8×10^{-2}	1.4×10^{-2}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calculated for four age groups: adults, teenagers, children and infants. The highest dose estimates for each age group are listed. The total-body dose estimates for all locations are for adults. For the bone and liver doses, the the estimates for the nearest garden and nearest cow are for children, and the estimates for the nearest goat are for infants.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 mile north, and nearest cow and garden = 1.05 mile east.

Table 5.6. Estimates of Offsite Doses to the Maximum Exposed Individual Caused by Failure of a HEPA Filter during Desludging Operations in the AFHB

Location	Pathway	Dose (mrem)[a]		
		Total-body	Bone	Liver
Nearest garden[b]	Inhalation	6.4×10^{-4}	9.0×10^{-3}	6.3×10^{-4}
	Ground Shine	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}
	Vegetable Use	1.7×10^{-1}	7.3×10^{-1}	1.0×10^{-1}
	TOTAL	1.8×10^{-1}	7.4×10^{-1}	1.1×10^{-1}
Nearest milk goat	Inhalation	8.6×10^{-4}	3.8×10^{-3}	4.6×10^{-4}
	Ground Shine	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}
	Goat Milk Use	8.7 × 10-2	6.2×10^{-1}	6.5×10^{-1}
	TOTAL	9.2×10^{-2}	6.3×10^{-1}	6.6×10^{-1}
Nearest cow and garden	Inhalation	4.4×10^{-4}	6.2×10^{-3}	4.3×10^{-4}
	Ground Shine	4.7×10^{-3}	4.7×10^{-3}	4.7×10^{-3}
	Vegetable Use	1.7×10^{-1}	7.3×10^{-1}	1.0×10^{-1}
	Cow Milk Use	2.1×10^{-2}	1.1×10^{-1}	8.1×10^{-2}
	TOTAL	2.0×10^{-1}	8.5×10^{-1}	1.9×1^{-1}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three-organ doses are listed in this table. Doses were calculated for four age groups: adults, teenagers, children and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden are for children, and the dose estimates for the nearest milk goat are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 mile north, and nearest cow and garden = 1.05 mile east.

5.1.5.4 Psychological-Socioeconomic Effects

The foregoing analysis indicates that the health effects of decontaminating the AFHB are expected to be negligible under normal operations. If the planned actions are understood and believed safe by the general public, the majority of people living in the vicinity of TMI will not experience additional psychological distress or adverse socioeconomic impact. Once initiated, this phase of decontamination would be expected to result in a decrease in anxiety levels among those who are concerned that cleanup was unnecessarily delayed.

The accident scenarios described in Section 5.1.4.2 involve spills, rupture of the HEPA filter attached to a vacuum cleaner, and failure of the building filter system. These hypothesized accidents could have psychological impact. Although the accident scenarios would have negligible radiological consequences, some segment of the local community would believe the hypothetical accident to be threatening, with a resulting increase in the symptoms of anxiety. The level of anxiety would be expected to increase in relation to believed threats, NRC's credibility, the level of controversy or uncertainty, and adverse media coverage. Accidents involving airborne releases, whether or not such releases reach offsite areas, could result in short-term consumer avoidance of agricultural products, particularly dairy products.

5.1.5.5 Other Environmental Effects

Discharge of about five gallons per week of Freon vapor to the atmosphere could occur if Freon is used to clean equipment. Such a release is negligible compared to the global releases of Freon and would have no significant effect in the local area nor on the global ozone layer.

5.1.6 Economic Costs

Based on the number of person-hours of work effort that was required for the portion of AFHB decontamination completed as of September 1, 1980, (about two-thirds of the total) the staff estimates that the total direct costs for decontamination of the AFHB will be between about $16 million and $22 million. A cost range is presented to allow for uncertainties in the effort required to complete the remaining third of the decontamination. The cost of desludging operations constitutes less than 5 percent of the total.

The costs for AFHB decontamination are presented in Table 5.7. The majority of the cost is for direct labor associated with decontamination and with maintaining the decontaminated areas.* Estimated equipment costs include items such as the recirculation vacuum filtration system, filter cartridges, filter processing equipment, special casks, shielding, etc. The bases of these cost estimates are described in Appendix K. Costs of waste packaging and handling, and waste transportation are covered in Sections 8 and 9, respectively.

Table 5.7. Economic Costs for AFHB Decontamination (thousands of dollars)

Item	Best Case	Worst Case
Labor	15,450	21,150
Equipment & Materials	450	650
TOTALS	15,900	21,800

5.2 REACTOR BUILDING

The initial objective of the reactor building decontamination is to establish and maintain radiological conditions (general area radiation, airborne gaseous and particulate activities, and surface contamination levels) that will permit reactor defueling and primary system decontamination to proceed. Defueling is a prerequisite for decommissioning or refurbishment. The following judgmental values have been applied as guidelines to determine whether this initial objective has been achieved:

- Reactor building sump water processed and the sludge removed.
- General area radiation exposure rates less than
  - 30 mR/hr below the 305-ft elevation
  - 10 mR/hr at the 305-ft elevation
  - 10 mR/hr average at and above the 347-ft elevation, with exposure rates to about 5 mR/hr at the defueling location. The lower exposure rates at the defueling station could be achieved by suitable placement of shielding and additional local cleanup.

*Direct labor is here defined as the labor of decontamination and other workers with assignments that require work in radiation zones (see Sec. 5.1.3.2).

Smearable contamination on building surfaces and equipment approximately 3000 to 4000 dpm/100 cm^2 exclusive of hot spots.

Processing of the primary system water and flushing and decontamination of the primary cooling system may be necessary in order to bring the radiation exposure rates down to 30 mR/hr below the 305-ft elevation.

The foregoing guidelines will not be affected by the outcome of the decision on future use of the site. The cleanup operations that must be carried out in order to achieve the initial decontamination objectives are also, for the most part, independent of the outcome.

5.2.1 Status and Specific Considerations

Preliminary reactor building decontamination operations that have taken place during the period prior to January 1981 include purging of the Kr-85 from the atmosphere in the reactor building, followed by preliminary entries into the building in order to map radiation fields, measure surface contamination, and inspect the extent of any damage to the building and equipment. These operations were prerequisites for the remaining building and equipment decontamination operations, which would have to be carried out regardless of subsequent decisions on the choice of cleanup alternatives or the decision on future use of the site. The operations were also necessary to ensure continued safe shutdown of the reactor. These operations, and the information that was gained from them regarding the condition of the building and equipment, are described in this section.

5.2.1.1 Reactor Building Atmosphere

Significant quantities of radioactive fission products and particulates were released into the reactor building atmosphere as a consequence of the accident on March 28, 1979 (see Table 5.8). The total atmospheric radioactivity in the reactor building just prior to the purge was about 44,000 Ci. Nearly all of this was krypton-85 (Kr-85) gas. The contribution from tritium (in the form of HTO vapor) was less than 3 Ci, and the contribution from all other radionuclides in the atmosphere was less than 0.001 Ci. By July 11, 1980, the atmospheric contamination had been reduced to a level such that full protective equipment is no longer required by workers. The sequence of events by which this reduction occurred are described below.

In consideration of the licensee's specific request dated November 13, 1979, for permission to remove the krypton gas, the NRC staff prepared and published a draft environmental assessment in March 1980 on decontamination of the building atmosphere and then a final environmental assessment (see Appendix C) in May 1980. On June 12, 1980, the Nuclear Regulatory Commission, following its own review of the staff recommendations and comments from the public, the Governor of Pennsylvania, and many other organizations and individuals, issued a Memorandum and Order (CLI-80-25)[8] which authorized Met-Ed to conduct a controlled purge of the reactor building atmosphere, commencing no sooner than June 22, 1980. The Commission also issued on June 12, 1980, its Order for Temporary Modification of Operating License No. DPR-73,[9] which established offsite dose limits for the purge.

Purging of the reactor building atmosphere by venting to the outside atmosphere started on June 28, 1980, and continued until the morning of July 11, 1980. The radioactivity vented during this period was estimated to be in the range of 38,000 to 50,000 Ci of Kr-85, with a median value of 44,000 Ci.* Additional purges were made on August 1, 8, 14-15, and 22, September 19-20, October 10 and 15, November 8 and 12, and December 4 and 9, 1980, in order to vent the Kr-85 released by offgasing from the water in the reactor building sump and from other sources. The venting was done to facilitate manned entries for inspection, sampling, and testing of reactor building conditions. Releases during August ventings were less than 60 Ci each, except for the purge of August 14-15, which resulted in releases of less than 84 Ci. The releases during the months of September, October, November, and December were 27, 15, 12, and 7.5 Ci, respectively.

The Kr-85 concentration immediately prior to purging was about 0.8 μCi/mL. The purging operations reduced the airborne radioactivity in the reactor building by a factor of about 10^7 to the point that the concentrations were less than the limits for worker exposure stated in Appendix B, of 10 CFR Part 20. Some radioactivity remains because of resuspension of radionuclides from building surfaces by air currents, outgassing of sump water and equipment, and from inleakage from the primary coolant system. Periodic purging is needed to prevent the airborne radioactivity from rising above the limits noted above.

Table 5.8. Composition of the Atmosphere in the Reactor Building Shortly before the Purge on June 28 - July 11, 1980

Constituent	Half-Life (years)	Activity Concentration (μCi/mL)	Total Activity in Reactor Building Atmosphere (Ci)	Volume in Reactor Building Atmosphere[a] (L)
Air	-	-	-	5.7×10^7
H_2O[b]				
at 20°C	-	-	-	1000 (liquid)
at 40°C	-	-	-	3000 (liquid)
Kr-85	10.8	0.8[c]	44,000	31 (gas)[d]
HTO[b,e]				
at 20°C	12.3	1.7×10^{-5}	1	7×10^{-7} (liquid)
at 40°C	12.3	5×10^{-5}	3	2×10^{-6} (liquid)
Cs-137	30	$< 1 \times 10^{-9}$[f]	$< 6 \times 10^{-5}$	$< 10^{-6}$ (solid)
Cs-134	2.1	$< 1 \times 10^{-9}$[g]	$< 6 \times 10^{-5}$	$< 10^{-6}$ (solid)
I-129[h]	1.7×10^7	$< 6 \times 10^{-9}$[i]	$< 4 \times 10^{-4}$	$< 10^{-6}$ (solid)
Sr-90	28.8	$< 1 \times 10^{-9}$[g]	$< 6 \times 10^{-5}$	$< 10^{-6}$ (solid)

[a]The reactor containment building internal volume is about 5.7×10^7 liters.

[b]The H_2O and HTO tritiated water were present in vapor form. A saturated vapor (100% relative humidity) at the temperature indicated was used as a basis for the calculations. The volumes listed in the last column are the liquid volumes that would be obtained if the vapor were separated out and condensed.

[c]Memo: J.T. Collins to B.J. Snyder and H.R. Denton, (NRCTMI Program Office Weekly Status Report" September 8, 1980.

[d]The volume of gas that would be obtained if all of the Kr-85 were separated out and stored in a container at room temperature and one atmospheric pressure.

[e]The tritium atoms released initially with the hydrogen gas exchanged with the hydrogen atoms in H_2O so that almost all of the tritium was present as HTO. The tritium activity in the atmosphere was calculated from the tritium activity in the sump water (1 μCi/mL (see J.A. Daniels, "Containment Sump Analyses," presented at the EPRI/DOE Seminar/Workshop on Decontamination/Dose Reduction Technology, November 17, 1979)) which gives the HTO/H_2O ratio, and the amount of water vapor in the reactor building atmosphere.

[f]Calculated from measurements of particulates absorbed in filters through which atmospheric samples are passed (reactor building particulate sample data, from continuous tabulation by the Metropolitan Edison Co., recorded January 30, 1980, and February 13, 1980).

[g]Calculated from the measured Cs-137 activity in the reactor building atmosphere and the measured Cs-134/Cs-137 and Sr-90/Cs-137 ratios in the containment sump (see "Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building Atmosphere," U.S. NRC, NUREG-0662, March 1980; J.A. Daniels, "Containment Sump Analyses," presented at the EPRI/DOE Seminar/Workshop on Decontamination/Dose Reduction Technology, November 27, 1979; and reactor building particulate sampling data, from continuous tabulation by Metropolitan Edison Co., recorded on January 30, 1980, and February 13, 1980).

[h]The I-131 radionuclide, which has a half-life of eight days, has decayed to an undetectable and completely negligible level.

[i]Calculated from the concentration of I-129 in the sump (see Daniels, cited above) and the atmosphere/sump concentration ratio for I-129. The latter was obtained from the atmosphere/sump ratio for I-131, which was determined from measurements prior to August 31, 1979, before the I-131 had decayed to unmeasurably low levels.

Environmental impacts that occurred as a consequence of the purge are described in Section 5.2.5.2.

5.2.1.2 Entries

Five successful entries had been made into the reactor building as of January 2, 1981. These entries were preceded by an attempted entry on May 20, 1980 (prior to purging of the reactor building atmosphere) which was aborted because the entry team was unable to open the inner door of personnel airlock No. 2. The inner airlock door was successfully opened on July 23, 1980, and two people entered the reactor building to conduct a general radiation survey on the 305-ft level and inspect for damage resulting from the accident. For the successive entries on July 23, August 15, October 16, November 13, and December 11, 1980, the size of the entry team was 2, 4, 12, 5, and 14 persons; entry times varied from 20 to 121 minutes; the total cumulative occupational exposure for all team members for each entry was 0.42, 1.03, 2.28, 4.11, and 5.9 person-rem; and the maximum whole-body dose received by any individual for each entry was 220, 340, 570, 460, and 650 mrem, respectively.[11,12] No skin exposure from beta radiation was detected until the fourth and fifth entries, when the maximum beta skin exposures were 170 mrad and 65 mrad, respectively. The entry team members wore protective clothing for all entries. On the first entry breathing air was supplied by individual tanks of compressed air; on succeeding entries, air from the containment building was drawn through individual HEPA filters by battery-powered fans into pressurized full-face masks.

The entry teams made radiation measurements, took samples of surface radioactivity by wiping and scraping surfaces, inspected for damage, took color photographs of the interior, and removed a few small loose items for subsequent laboratory analysis. During the third entry, essential maintenance on the No. 1 personnel airlock and on equipment used for monitoring from outside the reactor building was carried out. During the fourth entry, a high-volume air sample was taken and tests were made to determine the effectiveness of three different hands-on surface decontamination procedures. During the fifth entry the reactor pressure vessel head was surveyed from inside the empty refueling pool and the polar crane was inspected. The results[11] from these activities are summarized in Sections 5.2.1.3 through 5.2.1.5. Typical views of the inside of the reactor building, taken during the entries, are shown in Figures 5.3 and 5.4.

Before the start of decontamination, additional entries will be needed on a continuing basis to obtain more information on the distribution of radioactive contamination and radiation field intensities, and on the condition of the equipment. This information is needed to plan for the cleanup operations and for inspection and possible maintenance of instrumentation used to monitor the condition of the reactor. Procedures for subsequent entries are expected to be similar to those already used.

5.2.1.3 Building and Equipment Surfaces

Radioactive deposits (plateout) are present on most of the approximately 300,000 ft^2 of exposed building and equipment surfaces inside the reactor building. Measurements of plateout from samples obtained by wiping or scraping at 28 locations during the first and second entries yielded the following results for the plateout activity (in units of $\mu Ci/100\ cm^2$):** measurements at eight locations on the 305-ft elevation floor ranged from 0.3 to 47, with an average of 14; measurements at four locations on the 305-ft elevation walls ranged from 0.006 to 0.02, with an average of 0.01; measurements at two locations on the 347-ft elevation floor ranged from 6.4 to 6.9, with an average of 6.7; and measurements at four locations on the 347-ft elevation walls ranged from 0.01 to 0.04, with an average of 0.02. The remaining measurements of miscellaneous objects at ten locations, such as stairwell surfaces, tool chest, glass sample, steel plate, funnel, cable trays, entry door, etc., ranged from 0.04 to 29, with an average of 5.

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*The initial estimate of 57,000 Ci of Kr-85 was based on a conservative interpretation of measurements of small samples of the reactor building atmosphere obtained through penetrations in the outer wall of the reactor building. The values quoted above are based on measurements of the plant stack gas velocity and Kr-85 concentration in the vented gas. The uncertainty of ±6000 Ci is due to inherent difficulties in making accurate measurements of gas velocities in large vent stacks and the concentration of Kr-85 in the gas stream. These difficulties limited the accuracy to about ±14 percent.[10]

**An activity of 1 $\mu Ci/100\ cm^2$ is approximately equal to 9.3 $\mu Ci/ft^2$ or 2.2×10^6 dpm/100 cm^2.

Figure 5.3. View Outside the Inner Airlock Door of the Reactor Building Just Prior to Entry. The photograph shows Met-Ed's personnel in protective clothing and carrying communications and radiation detection equipment making final preparations prior to entering the inner door of the personnel access hatch of the reactor building at the 305-ft elevation (ground level).

Figure 5.4. View Inside the Reactor Building during Entry. A view of the entry team inspecting the condition of equipment and making radiation measurements on the 305-ft elevation (ground floor).

In a decontamination test conducted during the fourth entry, a small area on the floor of the 305-ft elevation with a beta-gamma surface activity of about 14 μCi/100 cm^2 was wiped with a towel saturated with demineralized water, then wiped with a dry towel, then wiped with a towel saturated with a detergent solution, then wiped dry again. After the first dry wiping the surface activity was reduced by a factor of 87; after the second dry wiping it was reduced by an additional factor of 2.4.

The plateout in the reactor building came from fallout as the steam escaping from the primary system condensed during and shortly after the accident. It was probably distributed much more uniformly on the walls and floor immediately after the accident. The staff infers that the much smaller plateout values on the walls is primarily due to flushing that occurred after the accident during the period that moisture evaporating from sump water heated by decay heat from the reactor was condensing on the walls and draining down on to the floors and back into the basement.

The principal radionuclides in the plateout are Cs-137 (83 percent) Cs-134 (13 percent) and Sr-90 (4 percent). Trace amounts of Co-60, Nb-95, Sb-125, and Ce-144 were measured at some locations. The Cs-137/Sr-90 ratios varied, depending on the location at which the sample was taken. Ratios as high as 53 and as low as 3 were measured, with an average of about 16. The variation is due to the nature and concentration of the liquids in contact with various surfaces and subsequent washing by condensation. The relative amounts of the three principal solid radionuclides are also different for different sources. The proportions for major sources at different locations are shown in Table 5.9.

The total plateout on the floors, walls, and ceilings on the 305-ft and higher elevations can be estimated from the numbers given above and the surface areas to be about 2 Ci. This does not include the radioactivity in drains and other locations not accessible for swipe measurements where contamination could accumulate. Many of the "hot spots" are due to such sources. Nor does the 2-Ci estimate include the plateout in the basement (i.e., the adherent film of contamination that would remain after the sludge, which consists of a layer of nonadhering particles, is removed.*) or inside the D rings. There are not sufficient data available as to the amount of radioactivity in drains or other inaccessible locations, in the basement, or inside the D rings, but the staff considers it unlikely that the total would be greater than 100 Ci; it probably is much less.

Most of the surfaces at the 282.5 ft (basement) level are submerged. The floor is covered to a depth of about 8 ft with about 700,000 gallons of sump water. This sump water and any sludge that has accumulated on the basement floor under the water must be removed before the surfaces in the basement can be decontaminated. The condition of the building and equipment surfaces in the basement and the amount of sludge present were not known as of January 2, 1981.

5.2.1.4 Radiation Levels

Plateout, the contaminated sump water, and suspended particles are the major sources of worker radiation exposure for both beta and gamma radiation. The general gamma radiation levels (excluding "hot spots") were measured during the first three entries in the range of 100 to 700 mR/hr at the 305-ft elevation (grade level) and 30 to 600 mR/hr at the 347-ft elevation. The average gamma radiation levels are estimated to be 500 mR/hr and 250 mR/hr at the 305-ft and 347-ft elevations, respectively. The staff estimates that the average gamma radiation level at the 305-ft elevation would drop to about 250 mR/hr if the basement were drained and decontaminated to the extent that radiation from the basement through the floor and drain openings was appreciably less than the contribution from plateout on the floor and walls at the 305-ft elevation. Measurements made of beta radiation levels by instruments in contact with the surface at various points on the walls and floors gave readings that ranged from 2 to 4 rad/hr.

Higher radiation fields occur at localized "hot spots" where concentrations of plateout occur and above the open stairwell, which is not protected by an intervening floor or wall from radiation

*Both sludge and plateout are removed from surfaces primarily by means that utilize water; the former by resuspension and the latter by going into solution or by detergent action. The important distinction between the two is that the former can be removed from the water by filtration; methods utilizing ion-exchange resins or minerals, or evaporation, are necessary for the latter.

Table 5.9. Relative Amounts of the Three Principal Radionuclides for Different Sources

Source	Radionuclide		
	Cs-137	Cs-134	Sr-90
Plateout in reactor building	83%	13%	4%
Dissolved solids in reactor building sump water	86	13	1
Filterable solids in reactor building sump water			
Sample A	8	1	91
Sample B	11	2	87
Primary system water	51	8	41
Dissolved solids in water in AFHB sump	81	16	3
Filterable solids in water in AFHB sump	79	16	5
Dissolved solids in water in AFHb sump tank	82	16	2
Filterable solids in water in AFHB sump tank	71	14	15

from the sump water. Typical measurements of the gamma radiation in such areas were 1-2 R/hr at the air coolers,* 2-5 R/hr at floor drains, 10 R/hr over the metal deck for the covered floor hatch, and 18 R/hr at the open stairwell. The highest measurement, obtained remotely, was 40-45 R/h at 5 to 7 ft from the sump water in the basement (below the 305-ft elevation).

5.2.1.5 Condition of Equipment

The containment spray system was actuated on March 28, 1979, following a hydrogen burn within the reactor building; hence, the system was operable at that time and may still be operable.[13] In addition, some of the building lights were activated during the second entry. Otherwise, there is no basis for assuming that any of the other reactor building equipment, services, or protective instrumentation needed for decontamination of the building surfaces and equipment would be operable at the beginning of decontamination operations. Provisions for supplying all of the required services and equipment by repair, replacement, or other means would be needed. It is believed that the reactor building equipment hatch at the 305-ft elevation can be used to transfer equipment to and from the reactor building before decontamination operations are begun. The status of certain key pieces of equipment is noted below.

The electrically-powered polar crane is assumed to be inoperable because of the adverse environment in the reactor building and because of the long period without operation or maintenance. During the fifth entry, visual observation from a platform immediately below the crane cab revealed considerable heat damage to electrical components.

The reactor building air cooling units, located adjacent to the equipment hatch at the 305-ft elevation, are operable but are grossly contaminated and will require extensive decontamination, shielding, or removal to reduce radiation levels for personnel working in the area. In-place repair or replacement would be required if these units were to be used for building cooling during cleanup. The extensive amount of air ducting at this elevation in the building may be highly contaminated and require special decontamination efforts, or replacement. (Some prior decontamination to reduce worker exposure would probably be needed even if the ducting were replaced.)

*The most detailed measurements around the air coolers were made during the third entry on October 10, 1980.

There was evidence at some locations within the building of the pulse of abnormally high temperatures and pressures which occurred during or after the accident: a dent in the enclosed stairwell door at the 305-ft elevation; a collapsed metal storage drum; a melted plastic telephone handset on the 347-ft elevation; and a charred sign. This damage probably occurred during the hydrogen burn. Aside from these observations, visual inspection did not reveal any obvious damage to the equipment at the 305-ft and 347-ft elevations.

Because the basement is flooded with 8 ft of contaminated water, most of the equipment there probably will have to be removed and disposed of.

5.2.2 Alternative Methods Considered

Alternatives other than those listed below may be suggested during the decontamination operations and should be considered if there is reason to believe that, within the ALARA concepts, they would reduce worker exposure, the amount of waste generated, decontamination time, or occupational hazards.

5.2.2.1 Building Decontamination

Decontamination below the 305-ft elevation (the "basement" level) and on the 305-ft and higher elevations (the "upper" levels) present different problems, primarily because of the 700,000 gallons of contaminated water that overflowed the sump and now covers the basement floor (the 282.5-ft elevation) to a depth of about 8 ft. Decontamination of the basement will involve pumping out the contaminated water and removing the sludge and debris, as well as decontaminating the building and equipment surfaces. Decontamination of surfaces is the major building decontamination task required for the upper levels. The alternatives for the basement and upper levels are considered separately below, but they are not independent of each other--some of the major alternatives concern coordination of decontamination tasks on different levels of the building.

Alternatives for Decontaminating the Basement

The contaminated water in the basement (the sump water) must be removed before other decontamination tasks in the basement can be started. After the water has been removed, removal of the sludge and debris and then decontamination of the underlying surfaces can proceed.

Alternatives for Removing the Sump Water. The alternatives for removing the sump water dffer in the rate of removal, transfer methods, timing, and clean-water backfilling of the basement.

Rate Alternatives. Three major alternatives that differ in the rate at which the water would be removed were considered by the staff: (1) rapid removal to a storage tank at some other location on the site; (2) removal at a rate that would permit the water to be processed before it was stored at some other location; and (3) a feed-and-bleed operation in which the water was returned to the sump after processing until the dilution reduced the contamination to a predetermined level.

The rapid removal alternative is considered by the staff to be the least desirable of the three alternatives. In order to accommodate all of the contaminated water in suitable existing tanks it would be necessary to utilize the Unit 2 reactor coolant bleed holdup tanks (231,750 gallons), the tank farm in spent fuel pool A of Unit 2 (110,000 gallons), the Unit 1 reactor coolant bleed holdup tanks (247,000 gallons), and spent fuel pool "A" of Unit 2 (320,000 gallons).[14] The precautions needed to avoid exposure to personnel engaged in other operations, the increased radiation levels in the storage areas, and the loss of spare tank capacity for managing the overall water inventory would hamper cleanup operations. The greater complexity (number of valves, pumps, etc.) and length of lines utilized compared to other alternatives would increase the transfer hazards. The storage hazards are also increased; there would be much less likelihood of leakage or spillage of the contaminated water (which has a radioactivity concentration of 0.75 Ci/gallon) if it could remain in the reactor building basement until it was processed. Use of Unit 1 facilities is very undesirable and would by transferring radioactive accident water increase the extent of contamination and increase potential hazards from accidental releases by significant amounts. For these reasons, rapid transfer of the contaminated water into existing storage facilities should be considered only as an emergency measure in case some unforeseen situation (such as major leakage from the building to the environment) made necessary the rapid removal of the contaminated water from the basement for temporary storage elsewhere.

Some of the preceeding objections could be eliminated by building new storage facilities specially designed to receive the contaminated water in the reactor building basement. Heavily shielded and leakproof tank, piping, and valves that would withstand earthquakes, tornados, and floods would have to be designed and constructed at some location outside any of the existing buildings but within a controlled ventilation structure. An extended period would be required for planning and construction, and even with the utmost care, the hazards from leakage and spillage from a tank located outside of the AFHB or reactor building would be much greater than for the alternative of leaving the water in the reactor building basement until the water was processed.

Rapid removal of a fraction of the sump water to one or more of the Unit-2 reactor coolant bleed holdup (RCBH) tanks is a less hazardous and more practical operation. There are three RCBH tanks, each with a capacity of 77,250 gallons, located in shielded cubicles in the auxiliary building. Even though this option does involve some increase in the spread of radioactivity and in the hazards of leakage or spillage, such spills would be limited to a part of the AFHB. This alternative of partial removal could be used (1) to permit the addition of water from other decontamination activities to the basement inventory without increasing the water level, or (2) to accelerate the basement draining schedule above that which would be possible if the other, slower water-removal alternatives were used.

Processing of the sump water as it is removed is considered by the staff to be a practical alternative. For a processing rate of 2.5 gallons per minute, which is the expected average processing rate for planned processing systems (see Sec. 7.1.3), the processing time for the water now in the basement would be about seven months. This time would be increased by one month for every 100,000 gallons of water added from decontamination operations at the 305-ft and higher elevations in the building.

If the feed-and-bleed alternatives were used to reduce the contamination in the sump water (other than tritium) by a factor of 10^6, which is the approximate reduction in dissolved radionuclides in the effluents from a processing system with decontamination factors comparable to EPICOR II, seven years would be required.* A variant of this alternative would be to use a feed-and-bleed operation for a shorter period of time to partially reduce contaminant concentrations and then implement a second alternative to complete the processing.

Transfer Alternatives. Four transfer alternatives for removing water from the reactor building basement were considered by the staff and judged to be practicable: (1) remove the water through the existing 4-inch-diameter sump pump discharge pipeline from the reactor building sump into the AFHB; (2) install a new submersible pump and discharge system in the sump or on the basement floor; (3) adapt the 18-inch-diameter sump discharge lines that feed the reactor building spray system and decay heat removal system to divert sump water to the liquid waste processing system, or (4) use a surface suction system inside the reactor building to transfer sump water to the tank farm.

The existing sump pumps in the reactor building basement cannot be used because they are under water and are not operable. The pump in Alternative 1 would be located in the auxiliary building and could discharge into a liquid waste collection header in the same area. The use of an existing pump, WG-P-1, located in the auxiliary building at an elevation of 281 ft was considered. Use of this particular pump presents two disadvantages: (1) when the sump water level dropped below 281 ft, air could be drawn in through the spray header and valve BS-V-1 (via air leakage), which could air-bind pump WG-P-1; and (2) sump water and sludge would pass through the reactor building spray pump, BS-P-1, and sludge would accumulate in this pump, thereby creating a significant source of radioactivity. A more practical variant of alternative 1 would be to use a pump similar to WG-P-1 in a reactor building spray pump vault, which is located in the auxiliary building at the 258½-ft elevation. This would eliminate the potential air-binding problem (because the pump would be below the bottom of the reactor building sump at 276½-ft) and sludge-buildup problem (because the BS-P-1 pump could be bypassed).

Timing Alternatives. The bounding cases for timing alternatives that involve basement decontamination operations are: (1) coordinate water removal and decontamination operations so that the water surface does not rise above the current level (which was 290.55 ft on January 8,

*If the water used for decontamination operations at higher building elevations were allowed to drain into the basement, this time would be increased.

1980, and rising at a rate of about 0.06 ft/month because of leakage of about 5000 gallons per month from the reactor cooling system); or (2) proceed with decontamination activities without regard to the water level in the basement.

A net increase in the amount of water in the basement would cause the water level to reach the following critical points: 40,000 gallons would bring the water level to the centerline of one of the valves, DH-VI-171, that controls the flow of water from the primary system to the mini-decay-heat-removal system (MDHRS); 115,000 gallons would bring the water level to the bottom of the Penetration 401,* 170,000 gallons would bring the water level to the bottom of equipment that operates DH-VI-171; and 1,270,000 gallons would bring the water level to the 305-ft elevation. The decay heat valves inside the reactor building that control water flow to the MDHRS are now open so that the MDHRS is operable. These valves might operate under water, although they were not designed for such operation, but they could not be closed if the water level reached the operating equipment. The only valve that would then be available to control flow to the MDHRS and isolate any leaks that might develop downline from the decay heat valves would be DHV-3, which is located outside of the reactor building.

Coordination of water removal with remote wash operations in such a manner that the amount of water in the basement did not increase by more than 100,000 gallons (in order to avoid flooding of penetration 401) and preferably less than 40,000 gallons, would be the major concern for the timing alternatives. A remote wash could add from 60,000 to 360,000 gallons of water to the sump in a period of a few days; hence, it might be necessary to remove up to 260,000 gallons before a remote wash could be initiated. The staff considers it unlikely that a remote wash will be done, for reasons discussed below under remote decontamination alternatives.

If water removal started before semiremote decontamination operations on the upper levels, the water level in the basement should not rise as a consequence of these operations. The water removal rate is expected to average 2.5 gpm. The water jet used in semiremote washing operations has a flow rate of 7 gpm, but it is operated intermittently; draining of water into the basement as a result of water jet use presents no problem relative to increasing the water level as long as the water jet is not turned on more than about 8 hours during each 24-hour period. If water removal were delayed until after semiremote decontamination operations were underway, it might be necessary to transfer some of the sump water to the reactor coolant bleed holdup tanks.

Other timing alternatives that would not directly affect decontamination operations in the basement are discussed below in connection with decontamination operations on the upper levels.

Backfilling Alternatives. Currently the sump water is the major source of radiation in the basement. After the water has been pumped out, the major source of radiation in the basement will be exposed deposits of sludge and other surface contamination. The two principal alternatives for reducing the radiation exposure from sludge and contaminated surfaces exposed by removal of the water would be to (1) proceed immediately with removal of the sludge and decontamination of the exposed surfaces in the basement, or (2) backfill the basement with clean or processed water in order to reduce radiation levels while decontamination work on other levels is in progress. The staff has estimated that the radiation levels at the underside of the floor at the 305-ft elevation from the exposed deposits in the basement could be up to three times less than the radiation levels from the sump water before it was removed. The radiation levels at the 305-ft and higher elevations are, therefore, expected to decrease uniformly as the water is pumped out and the basement is decontaminated. Backfilling the basement would delay cleanup of the basement and increase the amount of water that required processing.

The thickest deposits of sludge are expected to be in the reactor building sump, which is a pit 6 ft deep, 7 ft wide and 8 ft long with a capacity of 2500 gallons located in the basement floor

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*Penetration 401 is a penetration through the 4-ft-thick outer wall of the reactor building. It has a diameter of about 2.7 inches at the inside wall. It is normally closed with an airtight and watertight seal. It can be used to insert instruments or retrieve air or water samples; it is currently being used for instruments that measure the water level. It is proposed to be modified by adding a welded closure assembly to eliminate the potential for flooding through the penetration without eliminating the capability for making water-level measurements.[15]

just north of the center of the D-ring. Backfilling only the sump with clean or processed water to reduce exposure from this source would be a practical and desirable procedure that would not interfere with cleanup operations in the remainder of the basement area.

Alternatives for Removing the Sludge. The alternatives for removing the sludge from the reactor building are essentially the same as for the AFHB, except that in the reactor building the sludge might be deposited over a much larger area. However, neither the actual amount nor distribution of the sludge in the reactor building is known.

If it is assumed that the amount of sludge, which consists of particulate matter washed down from building and equipment surfaces, is proportional to the building and equipment surface area, and that conditions are comparable in the AFHB and reactor building, then the amount of wet sludge in the reactor building will be about two-thirds of the amount in the AFHB, or about 100 ft^3. This is probably an overestimate because almost all of the surfaces in the reactor building had been painted or otherwise finished in a manner that would facilitate decontamination, while a large fraction of the surfaces in the AFHB is unpainted concrete, which sloughs off more particulate matter. In addition, there was an unfinished part of the floor in the AFHB that contributed a large part of the cement residue, and there were ion-exchange resin beads present. There were no unfinished sections of floor in the reactor building and no means by which ion-exchange resins could contribute to the sludge.

The concentration of radioactivity in the sludge can be expected to be about the same as the concentration of radioactivity in solids filtered from the sump water. The radioactivity of filterable solids in the reactor building sump water was 1.6 Ci/ft^3 on July 1, 1980.[16] The isotopic distribution for the reactor building sludge is assumed by the staff to be the same as for the filterable solids in the reactor building sump water.*

Agitation of the sump water as it was being removed in order to resuspend as much of the sludge as possible and remove it with the water was considered by the staff; however, it appears unlikely that this method would be useful when the water level was high enough to cover the entire basement floor. Agitation of the water in the basement would have to be by remote means, and it would be impractical to provide sufficient agitation to remove sludge deposits from all surfaces in the basement as the water receded.

Agitation is expected to be most useful toward the end of the water removal operations when the only remaining water is in the sump, which may contain an appreciable accumulation of sludge. A water jet is expected to be the most effective means for washing the sludge from the remainder of the basement into the sump where it can be removed by any of the alternative methods discussed in Section 5.1.2.2.

Alternatives for Surface Decontamination. After the water and sludge had been removed from the basement, the alternatives for decontaminating the surfaces would be the same as those discussed below for decontaminating the surfaces at the upper levels, except that the remote wash alternative would not be applicable, and it might be desirable to use a high-pressure (10,000 psi) water jet in addition to the moderate-pressure (1200 psi) water jet used in the AFHB.

Alternatives for Decontaminating the Upper Levels (305-ft and Higher)

The alternatives considered by the staff for decontamination operations on the 305-ft and higher levels in the reactor building involve (1) coordination of tasks on the basement and upper levels; and (2) the methods used for surface decontamination.

Coordination Alternatives. The major coordination alternatives concern the timing of the start of decontamination operations on the upper levels relative to completion of initial cleanup operations in the basement. The reference time for distinguishing alternatives is the time at which cleanup of the basement has proceeded to the point that the contribution to the radiation fields on the upper elevations from sources in the basement is small compared to the contribution

*Two measurements of the filterable solids in the reactor building sump were available as of January 30, 1981; one reported in February 1980;[17] the other reported in July 1980.[16] The July 1980 measurements were on a sample taken from near the bottom of the sump and are expected to be more representative of the properties of the sludge.

from plateout on the walls and floors of the upper elevations. Tenfold reduction in the overall strength of radiation sources in the basement would be sufficient for this purpose and could be accomplished either by removing the water and sludge from the basement (except for the sludge in the sump itself, which could be shielded by backfilling or by other means) and washing down some of the most highly contaminated surfaces (if water and sludge removal alone were not sufficient) or by a feed-and-bleed operation at an average rate of 2.5 gpm over a period of 15 months (see discussion above under "Rate Alternatives"). This reduction of the radiation source strength in the basement by a factor of 10 or more, by whatever means, is referred to in this section as the "initial cleanup" of the basement.

The major coordination alternatives considered by the staff are: (1) defer the start of decontamination tasks on the upper elevations of the buildings until the initial cleanup of the basement is completed; or (2) begin decontamination of the upper elevations before initial cleanup operations in the basement have started. There is, of course, an intermediate range of alternatives in which decontamination at upper elevations would begin after initial basement cleanup had been started but before it had been completed. Another intermediate alternative would be an immediate preliminary decontamination of the floors alone (on which the plateout activity is 100 times larger than on the wells--(see Sec. 5.2.1.3), with further decontamination of the upper levels deferred until initial cleanup of the basement was completed. The alternatives (1) and (2) above represent bounding cases.

The only restrictions on coordination alternatives that seem appropriate in the light of information available at the time this document was prepared (other than ALARA considerations for overall planning) are that the water level in the reactor building basement should not be allowed to reach an elevation of 291.85 ft (which can be accomplished by not allowing the amount of water in the basement to increase by more than about 100,000 gallons), and that the reactor pressure vessel head should not be removed until about 300,000 gallons of sump water from the basement (together with any water that might be added from decontamination activities) have been processed. This amount of water, plus the 743,000 gallons of water already processed by EPICOR II, are needed for the spent fuel pools, cask pit, and fuel transfer canal. These facilities must be filled with about 1,040,000 gallons of water before the reactor pressure vessel head is removed and defueling operations commence. If defueling operations began before 300,000 gallons of processed water from the basement were available, it would be necessary to use a corresponding amount of clean water. This would cause an undesirable increase in the amount of tritium-contaminated water that would have to be stored or disposed of after cleanup operations were completed.

Surface Decontamination Alternatives. The primary decontamination task on the 305-ft and higher elevations of the reactor building is decontamination of surfaces. The methods available for this task may be classified as remote, semiremote, and hands-on categories.

Remote Decontamination Alternatives. In the staff's judgment and based on currently available information, remote decontamination is not an essential step in the cleanup of the upper elevations of the reactor building; however, in the event that a decision is made to carry out a remote decontamination step, the staff has examined two alternatives: (1) remote wash, and (2) use of robots.

The reactor building spray system, which can be operated remotely from outside the building, could be used to carry out a remote wash. This system is similar to the sprinkler systems used in public buildings, except that the flow of water is much greater. It can be modified for steam injection, although this involves some worker exposure because the spray nozzles must be changed. The staff considers the following alternatives to be practicable: (1) overhead spray deluge; (2) overhead steam injection through the building spray system; (3) a low-elevation saturated steam injection into the building atmosphere (which would also involve worker exposure during installation) and (4) a combination of the first and third alternatives.

The primary advantages of a remote wash is that it would provide an initial reduction in radiation levels with very little worker exposure. The disadvantages are that a large volume of water would be generated and only the dome area could be decontaminated; there is no spray system installed below the 347-ft elevation, which is the floor of the dome area. The walls in the dome area are much less contaminated than the floors, probably because they already have been partially washed by condensation from the building atmosphere (which has been at 100 percent humidity for much of the time); hence the decontamination factor for a remote wash may be lower than is usual (i.e., 2 to 10) for such operations.

The use of detergents or chemicals rather than processed water for the remote wash was considered by the staff, but will not be pursued further because the large amounts of detergent or chemicals required would severely complicate the eventual processing of the wash water.

The use of robots for remote decontamination was not considered further for the following reasons. The usefulness of industrial robots is limited because they are unable to move over obstacles; the field of view, resolution, and depth perception of their optical systems are limited; and the manipulative capacity of mechanical arms and grippers is limited. Robots specially designed for the TMI-2 environment would require considerable development and a long lead time. The staff does not, however, rule out the use of robots in specific circumstances when these disadvantages are not a severe handicap and when remote means are needed to cope with decontamination of high radiation areas where shielding is not feasible.

Semiremote Decontamination Alternatives. Semiremote decontamination operations require the presence of people inside the reactor building to operate the decontamination equipment. The use of appropriate equipment would limit occupational radiation exposure by allowing the workers to be farther from the radiation source and by decreasing the worker exposure time. The various methods could be used individually or in combination, or alternatively, one method might follow another on the same surface, with the choice depending on the success of the application of the first option. The options considered by the staff are: hose washdown, using low-pressure water; low-pressure hot water misting and washing; high-pressure water jet washdown; steam jet washdown using steam to propel other liquid cleaners; steam cleaning using steam alone; sandblasting; and vacuum blasting.

These alternatives are not regarded by the staff as being exclusive nor is it intended to exclude other methods, such as ice blasting or the use of foam. All of the methods listed are feasible and any of them may be used.

Some of the procedures that normally involve hands-on techniques could also be done semiremotely. For example, commercial equipment that can perform scrubbing and wet-vacuuming operations on floors automatically under remote control are available and may provide a practical means for reducing operator exposure for these operations.

Hands-on Decontamination. The hands-on phase of decontamination would be instituted when remote (if any) and semiremote decontamination methods had been productively exhausted. Most hands-on approaches are close in concept to industrial janitorial services--workers use wet cloths, brooms, scrub brushes, mops, vacuum cleaners, etc., to clean surfaces and equipment. These methods are the same as those used for AFHB decontamination (see Sec. 5.1.2.1).

Most of the surfaces in the reactor building are painted, and it is expected that washing and scrubbing with water or detergent, or, at most, removal of the paint film will be sufficient to reduce the plateout to acceptable levels. However, there are some areas (for example, part of the surface area of the wall in the enclosed stairwell) where removal of some of the material beneath the surface may be required. Some areas in the basement that have been immersed in contaminated water for almost two years also may require such treatment. Methods that remove a layer of material include the use of sandblasting, jackhammer, pneumatic or hydraulic impactor, or concrete spaller. The discussion in Appendix U is relevant to the foregoing on these techniques.

5.2.2.2 Equipment Decontamination

The equipment in the reactor building can be decontaminated by a variety of methods. The alternatives listed below are all feasible and have been used in previous decontamination operations (see Sec. 1.5). The alternative methods for decontamination of surfaces are the same as those discussed in Section 5.1.2.1.

- In-place decontamination. This alternative would involve use of hands-on techniques.
- Local demounting and reinstallation. This technique would incorporate disassembly before using hands-on techniques.
- Removal for decontamination outside of the reactor building. The equipment would be removed to a special decontamination area where it could be decontaminated remotely or with hands-on techniques.

- Disposal. This option would be appropriate where replacement was more cost effective than decontamination.

5.2.2.3 New Facilities Needed

A large-scale equipment handling and transfer capability in a containment service building would be required for temporary storage and movement of equipment and decontamination of the reactor building. The minimum requirements of this containment service building would be to serve as an extended barrier, as an airlock, and as an interface building. The alternatives considered by the staff for fulfilling these requirements are:

- A medium-size, medium-integrity, steel-frame, metal-sided building designed to provide proper control of contamination and to provide minimal processing area.

- A larger, medium-integrity building similar to that above designed to provide space that would permit more processing and ease congestion.

5.2.3 Details of Methods and Facilities

5.2.3.1 Descriptions of Methods and Facilities

Descriptions of those feasible methods for decontaminating the reactor building that were listed and described briefly in Section 5.2.2 and are most likely to be used are given below. Descriptions of the enclosures that would be needed to control the spread of contamination, details of the equipment and of the decontamination procedures for the equipment, and descriptions and diagrams of the containment service building needed to control access to the reactor building and provide a staging area also are given below. Examples of successful decontaminations using some of these techniques are given in Section 1.5.

Semiremote Decontamination

Semiremote decontamination techniques involve the use of equipment that permits the operators to stay some distance from the radiation source. Experience has shown that it is most effective to remove all loose materials and debris before these techniques are used (see Sec. 1.5). Hot spots are shielded either before or while the semiremote decontamination is being conducted and then are decontaminated later when the general background dose rates are lower.

For the semiremote decontamination methods, the use of a holdback-carrier solution consisting of one milligram of nonradioactive cesium per liter of reprocessed water is advantageous in large-scale cleanups. The holdback carrier works as follows: enough nonradioactive cesium is added to the base liquid so that the number of atoms of inactive cesium in the solution will far exceed the number of atoms of radioactive cesium on the surface. As the holdback-carrier solution is washed down a contaminated wall, the radioactive cesium on the wall is replaced by nonradioactive cesium. By using holdback carriers, the contaminant transferred to the wash is more likely to remain in solution than to be redeposited on the surface. Hose wash and high-pressure water jet operations are suitable for use of holdback-carrier solutions. Other semiremote operations considered are steam cleaning, sandblasting, and vacuum blasting.

Hose Wash. As a decontamination method, hose wash offers some advantages in terms of flow rate control, flow pattern, and directional properties. These factors are especially advantageous for decontamination of hard-to-reach areas. However, because of low impact forces, hose wash is less effective as a general decontamination technique than the other methods considered. In fact, if the surface being cleaned is covered with oil or grease, hose wash is ineffective. Depending on conditions, hose wash decontamination factors range from 2 to 100. Flow rates for hose wash are typically about 50 gpm.

The hose-wash flow pattern can be a solid stream, a spray fan, or surface irrigation. Solid-stream washing is used to forcefully penetrate and dissolve encrusted material and to transport pooled or piled contaminants toward drains. A portion of reactor building surface contaminants will be combined with borate films and sodium hydroxide to form "Borax", which is soluble in water. Hose wash is, therefore, a major cleanup tool in the building/equipment cleanup task and will be particularly effective on those surfaces that were not adequately cleaned by means of remote decontamination. The spray fan flow pattern provides wide area coverage with low splatter. Surface irrigation nozzle patterns provide a mild flush action with no splatter. The fan pattern is expected to provide rinse action following solid-stream wash. Surface irrigation may be used to gently flush contamination to lower elevations.

Low Pressure Hot Water Misting and Washing. In this method the water is applied by means of a nozzle that generates a fine mist. It is similar to a hose wash, with the difference that the use of hot water makes it more effective for surfaces covered with oil or grease and the use of a mist rather than a stream of water reduces the volume of water used.

Water Jet. High-pressure water jets* consist of a high-pressure positive-displacement pump with a lance or gun for delivery. The unit can produce water pressures as high as 15,000 psi, but pressures of less than 10,000 psi are more typical. Pressures of 1000 psi are common in low-pressure units. The high impact force of these units makes them effective tools for removal of hard-to-remove contamination.

The volume of water emitted per unit time is a function of the delivery pressure and the nozzle opening. However, the total volume of water required to complete a given task is approximately equal using either high- or low-pressure water jet units.

The selective use of nozzles improves decontamination efficiency by matching nozzle flow patterns to specific work tasks. For example, the slotted fan spray nozzle with its knife edge pattern is typically used for descaling applications, while the circular nozzle (using high pressures) provides a high impact stream at distances up to five meters. The higher pressures will descale paints and remove oxide films. High-pressure water jets create splatter and can redistribute contaminants to other locations on the same surface. This characteristic can be minimized by careful pressure selection. A water jet unit can be equipped with a nozzle brake for more precise control at the sacrifice of some delivery pressure. Water jets can be turret-mounted to reduce operator fatigue during long hours of use. Unit positioning may be assisted by crane or use of movable platforms on wheels.

While the water jet may yield high decontamination factors (typically exceeding 1000), ultimate deposition of scattered contaminants is not completely controllable, and at high pressures the potential for creating high concentrations of airborne radionuclides is strong. The use of lower pressures to improve control might extend the length of the immediate task, but as a tradeoff it will reduce unwanted side effects. The use of radiological confinement, such as temporary barriers, isolation, and collection sheets, is essential during high-pressure water washing.

Steam Jet. Steam-jet decontamination is similar to water-jet decontamination. Water or reagents are introduced into an injector system and mixed with steam. The water-steam mixture is expelled through a nozzle. The delivery pressure is controllable from atmospheric to very high, but is typically less than 1000 psi. The volume of water delivered is usually greater than that for water jet. The delivery rate is about 16 gpm at 250 psi and 19 gpm at 400 psi. Steam-jet application of decontamination reagents is best done at low velocity.

The paths of the waste solutions and removed contamination from steam-jet decontamination are the same as for the previously described hose and water-jet washing. The potential for greater liquid-waste generation must be balanced against the potential for a higher decontamination factor.

The heated solution may provide a higher decontamination factor and the lower pressure reduces splatter and subsequent dispersion of the removed contaminants. This technique provides better control than a water jet but may not provide the necessary abrasive force to remove contaminants. The driving unit is usually the water and steam pressure; whereas the water jet requires a positive displacement pump.

Steam-jet cleaning is best applied to cleanup of areas containing oil and grease deposits. To this end the steam jet may be more effective for cleaning fixed pump and oil-lubricated equipment, as well as areas near such components. Borate and sodium tetraborate deposits may be more easily cleaned or dissolved using this method as a result of the elevated temperature of the solution.

Steam Cleaning. The steam-cleaning method involves ejection of steam through a nozzle or lance directed toward the surface to be cleaned. Typical pressures are less than 500 psi. The generation of liquid waste is considerably less for this method than for the previously described

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*The high pressure water jet will sometimes be referred to merely as a "water jet."

methods; however, a means must be provided to flush or vacuum the loosened wet contaminants. At 250 psi the rate of water delivery is less than 2 gpm.

Steam-cleaning techniques are frequently used for degreasing and, therefore, may lend themselves to cleaning the same areas and components as discussed for steam jet cleaning. For effectiveness, the nozzle must be close to the surface, and as with the other methods, contaminant dispersion is the major disadvantage. Steam cleaning is especially effective over irregular surfaces and provides minimal runoff where other methods may cause dripping or collection of excessive amounts of water in areas where liquids are undesirable.

Sandblasting. Sandblast decontamination is an abrasive cleaning technique involving use of air pressure to drive sand particles at high velocity against a surface. Most often it is a dry technique with little or no liquids produced. The abrasive action can be adjusted by grit size, air pressure, and standoff distance to provide an action that will scour, peen, polish, or remove the surface material. Sandblasting is a rapid and effective decontamination method, but because of the abrasive action the method is limited to use on surfaces where roughing does not destroy the usefulness of the item worked on. A disadvantage of sandblasting is that a large volume of contaminated grits are generated and must be disposed of as radioactive waste. Typical uses for sandblasting are cleanup of odd-shaped equipment, paint removal, and cleanup of metal and porous surfaces such as concrete where deep contaminant penetration cannot be removed by liquid cleaning methods.

Abrasive action during sandblasting is somewhat dependent on the skill of the operator. Typically, sandblast air pressures for steel and concrete run up to 90 psi. Pressures of about 60 psi provide scouring action, and still lower pressures may be used on nonferrous metals. A sand grit size of 20 mesh is usually quite damaging to surfaces, while grits above 100 mesh create a particulate airborne contamination problem. Sandblasting techniques require confinement of the abrasives. Movable tenting is generally used to control dust. The contaminated sand may be collected and removed by vacuuming.

Wet sandblasting is milder than dry sandblasting and lends itself better to complex shapes; cracks can be irrigated when water is used as the driving force. The water cycle of wet sandblasting also acts as a rinse so that the generation of airborne radionuclides is low.

Vacuum Blasting. The techniques of vacuum blasting, with abrasives other than sand, is a proven method and provides for the reuse of the abrasive material. In most cases, the abrasives are steel shot or wire cuts. The action is harsh and very effective. Reduced pressures can be employed to scour rather than remove surfaces. The removed surfaces, contaminants, and pellets or cuts are simultaneously vacuumed up and separated by a cyclone separator. The contaminants are removed to a specially protected unit chamber. On metals, the action can have a scouring or peening effect. Since the unit can be floor mounted or hand held, its versatility lends it either to semiremote or hands-on methods of decontamination. Vacuum blasting is most useful when the unit can be held perpendicular to the surface being cleaned. On irregular surfaces, reflection of abrasives can be a problem and vacuum pickup is less efficient. The principle advantage of this method is control of blast dust, which is nearly all returned to the unit. Very little dust is produced when a metal abrasive is used, and the life of the abrasive material can be as much as 100 times that of sand. Confinement, tents, barriers, etc., are recommended and should be used as standard practice. Decontamination factors with vacuum blast exceed 1000.

Hands-on Decontamination

The hands-on effort should be minimized by first using semiremote decontamination methods to the maximum extent possible so as to minimize dose to the workers. The effective use of barriers, partitions, screens, curtains, and enclosures is desirable to separate work areas (clean from dirty), permit several activities to proceed simultaneously, and minimize recontamination.

General surfaces (such as floors) are expected to require scrubbing either by industrial floor scrubbers or by hand, followed by wet vacuuming, and possible detergent cloth wiping. A final reagent/rinse mopping then would complete the effort. Any liquid or solid waste generated would be collected and disposed of by solidification or compaction in a form acceptable for disposal.

Overhead areas may require damp scouring with reagents followed by rinses and cloth wipes. High-elevation work above floors probably will involve use of bosun chairs, scaffolding, and telescoping platforms to reach all surfaces. The area above the polar crane may be reached by using the crane beams as a staging platform.

Aids to provide more efficiency in the hands-on effort include industrial janitorial equipment and hand-held automobile scrubbers and polishers. Application of strippable coatings or foamed-on cleaning agents followed by wet vacuuming and wiping reduces scrubbing requirements. Generally, a final wiping to remove residual radioactivity (transfer-technique) is more effective than repeated rinses.

Areas where paint is removed may require immediate sealing after decontamination. Depending on the status of the reactor building decontamination operations, decontaminated floors either should be covered or kept clean with oil-cloth brooms.

To reduce the potential for resuspended radioactivity in the basement from migrating to other elevations, this level should be isolated by means of barriers placed over penetrations in the 305-ft-elevation floor. Portable blower systems with HEPA filters could be strategically placed at the 305-ft elevation with "elephant trunk" ducting inlets lowered through the grating areas. These blower systems would create a negative pressure (in relation to the 305-ft elevation), and the local air flow pattern would be from less to more contaminated areas.

Concurrent with the above task, eductors could be introduced into the water in the basement. The sediment/sludge would be agitated by the eductor system suction, and dishcarge from the system would pass through screens and shielded roughing filters to remove sediments and sludge. The main objective would be to remotely remove the sludge and particulate contamination from the basement with as small a contribution to the personnel dose as reasonably achievable. Equipment similar to that used to vacuum sludges from reactor fuel storage pools could be used, but the exposure of workers to radiation would be considerably higher.

An alternative to the agitator system is a high-pressure water jet coupled to suction dredges. The dredged solution would be recycled through a filter bank system and a portion of the solution used as the liquid supply for the water jet. This method requires an agitation pool at least one foot deep to provide shielding and to prevent resuspension. During the dredging operation, ceilings and walls could be decontaminated by semiremote techniques.

The details of methods used for concrete removal are described in Appendix U.

Enclosures and Barriers. Enclosures and barriers consisting of portable, semi-airtight screens and tents would be needed in order to inhibit the spread of contamination to pre-cleaned areas. The first radiological enclosure needed would be for isolating the encroachment area. This enclosure would be located at the equipment hatch. The equipment hatch location has a monorail available and is contiguous with the planned containment services building.

The enclosure erected for the encroachment area would be typical of other enclosures used in the reactor building except for size. The enclosure might be a waterproof room constructed of a fire-retardant wooden framework, with polyvinyl chloride roof and walls. The floor could be fitted and glued in place after the walls were up. The walls of such structures customarily have ample clear material for viewing windows inserted in opague wall panels. All seams would be double stitched, then coated with vinyl glue to cover stitch holes. Each compartment could be equipped with a blower and roughing filter so that the air would flow from the direction of clean areas to contaminated areas. An alternative would be to provide a blower fitted with high efficiency filters (HEPA) in the last of several connected compartments which, when actuated, moved air down the chain of compartments. This system would establish a pressure difference between compartments, keeping the dirty sections at negative pressure with respect to the cleaner areas. Air flow would then always be from clean to dirty, thereby inhibiting the spread of contamination.

If enclosures were needed in an area before decontamination could proceed, shielding might be required to protect the workers erecting the enclosures and those who would work inside. A low-density shielding material (e.g., rubber, wood) would shield out low-energy beta radiation. A thin layer of aluminum or steel would shield out higher-energy beta radiation. If reduction of gamma radiation were required, lead blankets, lead sheet, lead brick, or high-density concrete blocks would have to be inserted between the gamma source and the enclosure. Shielding materials should be packaged or covered to prevent their contamination. This would reduce the amount of solid wastes. Fabricated stack tanks or movable five-gallon containers filled with water might also serve as temporary shielding materials.

To reduce the possibility of airborne contamination from affecting other work zones, the 305-ft and 347-ft levels should be sealed from one another at the appropriate time. Floor gaps could be

sealed with fire-retardant materials, polyethylene, or waterproof tape. In areas where liquids are required to flow down the reactor building wall and through the floor gap, the seal could contain a sluicing system that could be opened. Blower systems with HEPA filters located on the 347-ft and 305-ft elevations would be needed to maintain the desired air flow rate and air direction (from areas of lower contamination to areas of higher contamination) between elevations and allow the reactor building cooling system to function effectively. Decontamination of the spaces below the 305-ft level then could proceed without recontamination of the upper levels. Semi-remote methods then could be employed to decontaminate ceiling and walls exposed above the sump. The sump would serve as a holding reservoir for spent decontamination liquids and reduce resuspension of airborne particles.

Equipment Decontamination Alternatives

In-Place Decontamination. In-place decontamination includes hosing, washing, brushing, or steam cleaning of equipment without removing it from the installed position. This is the fastest and least expensive method, and involves the least total radiation exposure to operating personnel; however, the method would likely ruin some of the equipment and would leave contamination in crevices, threaded connections, and areas to which adequate access is not possible.

Examples of equipment that probably could be decontaminated in place are sealed or enclosed motors above the flooded level that are in protected locations, such as the elevator drive systems at the 370-ft elevation level and motorized actuators between the 305-ft and 347-ft levels that would be protected from contaminated "rainfall" by the floor at the 347-ft level.

Local Demounting. This procedure would include the same methods as discussed above, but in addition would involve, as needed, demounting of the equipment or adjacent equipment to permit cleaning of otherwise inaccessible locations. Increased radiation dose would result from the additional time required for removing the components, especially if these procedures were complex. The equipment would remain demounted until a decision on ultimate use of the facilities were made. It would then be removed if the facilities were decommissioned, or remounted if the facilities were reused.

Equipment that could be decontaminated locally after demounting are enclosed devices like motors or actuators subjected to dripping contaminated water, or equipment such as handrails, simple seismic restraints, or hatch cover gratings.

Removal for Decontamination Outside of the Reactor Building. Components too complex or delicate for quick cleaning in place or local decontamination after demounting could be removed from the reactor building. A building is being established for this. It would provide storage for the equipment until the start of Phase III operations. It could be used for refurbishing equipment either for salvage (if the facilities were decommissioned) or reuse.

Equipment that would require removal from the reactor building include such items as instrument cabinets, cabling, hydraulic snubbers, and some electrical switchgear.

Disposal. Any component that had been submerged or contaminated to a point that the decontamination effort and resulting occupational radiation exposure would be excessive with respect to its replacement costs would be a candidate for disposal.

Likely candidates for the disposal category are items that have been submerged, such as ducting, cabling, and instruments at the 282-ft level, or items for which past experience indicated that decontamination was not feasible. Examples are crane lifting cables and exposed mirror insulation on the primary system.

For much of the equipment, detailed advanced planning is not possible because the nature of the contamination problems is not known and the effectiveness of in-place decontamination methods as they apply to specific items is not certain. An example is the cooling coils in the air-cooling equipment at the 305-ft level of the reactor building. These are certain to be highly contaminated because of their long-term operation in the highly contaminated building atmosphere; however, the primary contaminant is likely to be cesium, which is water soluble and might be removed without the need to disassemble the air-cooling units.

There are various pieces of equipment in each of these categories, but most will probably not be classifiable in advance of actual work because of the unknown contamination and condition of each component. Since considerable equipment would have to be removed from the reactor building,

regardless of whether it were decommissioned or refurbished, an extensive parts control and inventory system would be required. There would be extensive use of the facilities immediately outside the equipment hatch, so that some quick method of making a disposition decision (salvage or scrap) decision would have to be developed, or a holding area would be required at some place out of the way of the activity at the hatch. This would require packaging of the components for transport through noncontaminated areas, at the cost of additional handling and increased occupational dose.

Certain equipment recovery activities can be predicted at this time and are listed below by their location in the reactor building.

Details of Equipment at Different Levels

430-Ft Elevation--Polar Crane Level. The polar crane would have to be decontaminated and at least partially refurbished early in the reactor building decontamination program to permit its use to move heavy equipment between levels in the building, including installation of an auxiliary crane and small boom cranes to aid in the decontamination effort. Additional activities likely to be required are removal of the trolleys on the fuel handling bridge, removal of the shield blocks above the reactor, removal and handling of the reactor coolant pump motors, cable replacement, and handling of any casks or shielding required during the decontamination. The effort required might vary from refurbishment of only the bridge drive components in place to complete replacement of the trolley. The polar crane is needed for cleanup and defueling activities, regardless of whether the facilities are decommissioned or reused.

367-Ft Elevation--Top of Shield Walls. Additional lifting and handling equipment (in the form of small boom cranes and possibly a bridge crane with rails) would need to be installed at this level before most of the work involving defueling or work on the primary system inside the shield walls could start.

347-Ft Elevation--Main Working Floor Level. This elevation would have to be cleared of all unnecessary equipment to make room for handling of components being moved out of the building and for laydown of components during defueling operations. Refurbishment or modification of the fuel handling bridges at this elevation might also be a high priority task.

305-Ft Elevation--Equipment Hatch Level. The area around the equipment hatch and a path to the vertical access hatches on the south side of the building would have to be cleared and established as relatively low-radiation-level areas early in the decontamination effort since most equipment transfer and personnel entry and exit would be at this location. The reactor building air cooling units located on this level immediately adjacent to the equipment hatch probably are highly contaminated and would require extensive decontamination, shielding, or removal. Refurbishment in place or replacement would be required in order to provide building ventilation. The extensive amount of air ducting at this elevation also might require special effort if it is highly contaminated. Most essential building services would be brought in through the equipment hatch or spare reactor building penetrations at the 305-ft level.

282-Ft Elevation--Sump Level. Because about 8 ft of highly contaminated water now floods the sump level, most of the equipment at this level probably would have to be scrapped.

Associated Facilities

Large-scale equipment handling and transfer capability would be required for decontamination and decommissioning or refurbishment of equipment and the reactor building. The 23-ft diameter equipment hatch on the southwestern side of the reactor building at ground level is the only feasible exit/entry point for these transfers. Since it would be necessary to use this hatch before the inside of the reactor building is decontaminated, an enclosure of some type would be required on the outside of the equipment hatch to serve as an interface between the reactor building and the outside environment. The minimum requirements for this service building would be as follows:

- Serve as an extended contamination barrier so that contaminated items can be brought out of the reactor building for staging, decontamination, examination, and packaging.
- Serve as an airlock so that material and personnel transfer through the equipment hatch is independent of uncontaminated transfers in and out of the containment service building.

Serve as an interface building to permit rigid control of equipment, supplies, and personnel entering and leaving the contaminated zone.

The containment service building also would include other facilities needed for working with contaminated material and equipment, such as a laundry facility for decontamination of clothing, equipment repair areas, and areas for temporary storage of contaminated waste, and also related facilities that do not involve contaminated material, such as a personnel access facility and a work-effort management and control facility.

Minimum-Size Medium-Integrity Building. A conceptual design for a minimum-size building to meet the facility requirements is shown in Figure 5.5. It would include an airlock outside the equipment hatch. The airlock would have to be large enough to accommodate the largest piece of equipment, including its transport device. This might be a reactor coolant pump motor, the fuel handling bridge trolley, or the auxiliary bridge crane that has been proposed for installation to aid in defueling efforts. Beyond the equipment hatch airlock would be another isolation area large enough to permit loading any of the equipment onto a truck or perhaps a rail car. Equipment would be packaged in the equipment hatch airlock, transferred to the loading area and verified as being free of surface contamination before the building's exterior doors could be opened. Interlocking would be required to ensure that the airlock/loading area door could not be opened when any of the exterior doors were open. Some limited decontamination or equipment survey and evaluation work could be conducted in the air lock, but this would obstruct personnel access and material transfer through the hatch. To preclude contamination of the previously clean areas and equipment, contaminated items would not be brought into the loading area before packaging. Most equipment would need to be transported to other facilities for decontamination, refurbishment or disposal.

Contaminated water, as well as particulate solids that were suspended in the water, would be pumped out through pipelines into the AFHB. The solids would be filtered out, immobilized, and packaged for final disposal in the AFHB. Sludge and other wet solids that were not removed in this manner would have to be packaged in a form suitable for moving through the hatch and staging area without introducing any appreciable increase in the radiation exposure. This could be accomplished either by immobilizing and packaging the wet solids within the reactor building so that they would be in the final packaged form before being moved through the equipment hatch, or by using shielded, watertight containers to move the wet, unprocessed solids through the equipment hatch to a location outside the reactor building where they could be immobilized and packaged for final disposal.

A steel-frame, metal-sided building could be built at the equipment hatch to meet the minimum facility needs identified above. Such a building might be amenable to complete decontamination and, hence, disposal as uncontaminated waste after all decontamination operations had been completed. It would need to be seismically isolated from the containment building and could be designed for Uniform Building Code (UBC) seismic and wind loading conditions. Building integrity would be sufficient to permit staging and processing of reasonable quantities of contaminated materials, but versatility would be low because of the minimal space provided.

Larger Medium-Integrity Building. A larger version of the building described above could be constructed to provide more space and greater versatility. A diagram of a larger building of the same general construction is shown in Figure 5.6. The building would cost more, but should not take significantly longer to design and construct. The additional space would permit more of the processing of contaminated material to be performed in the building; this, in turn, would decrease the effort that would have to be expended in packaging contaminated materials for transfer to processing operations elsewhere. Use pressure and congestion should also be more controllable because of the greater versatility and increased space of the building. More storage space for equipment would also be available while waiting for a decision on decommissioning versus restart.

The conceptual design has a large work area between the equipment hatch airlock and the various loading and shipping areas. This would permit significant waste staging and processing to occur within the contamination control boundary, but outside the high radiation levels of the reactor building and away from the congested area immediately outside the equipment hatch. Since there would be less packaging for transport and unpackaging for processing, contaminated packaging waste generation and occupational radiation dose should be less.

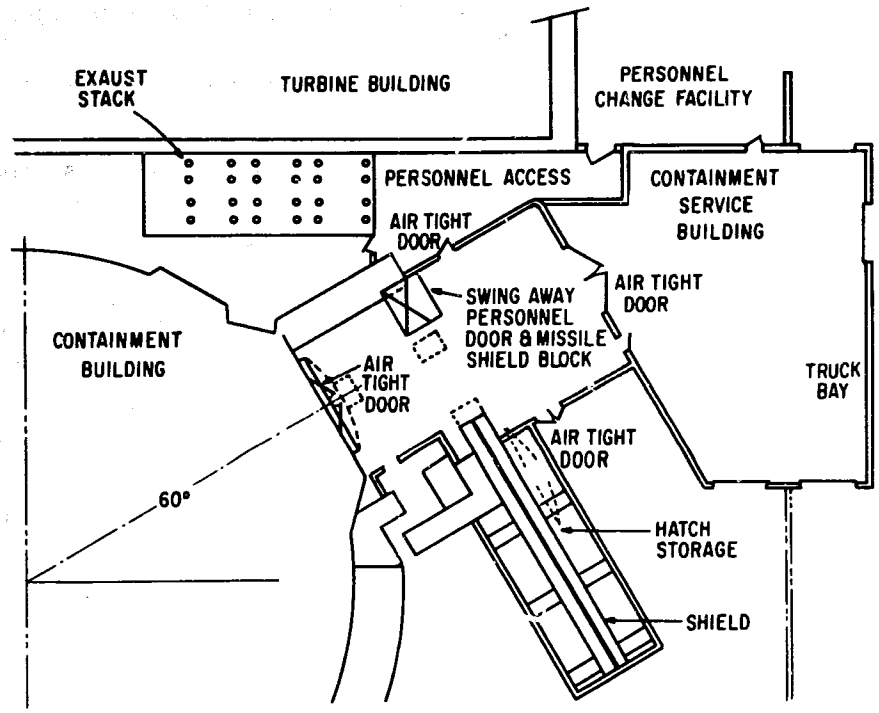

Figure 5.5. Minimum Size Containment Services Building.

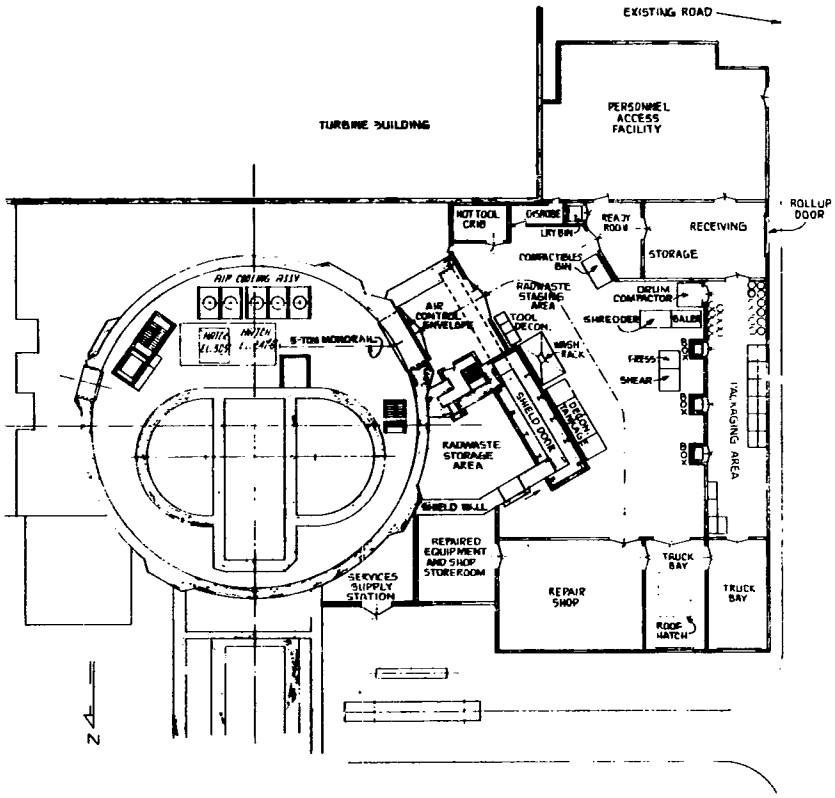

Figure 5.6. Larger Containment Services Building.

5.2.3.2 Work Effort Required

Decontamination of the reactor building will require a coordinated effort between decontamination crews and support personnel from several disciplines. The work effort estimates presented in this section include decontamination workers and workers with assignments that require entry into the reactor building (radiation protection, engineering, construction, maintenance, and plant operations personnel with in-building assignments), but not workers with assignments that do not require entry (personnel engaged in training, suitup assistance, supply clerical, planning, management, and similar assignments that do not require entry). The staff assumes that for personnel with assignments that require entry into the building, the proportion of time spent in productive work on the inside will be about half of the total.* The remainder of the time will be spent outside the building in putting on and removing protective clothing and equipment, planning and instruction sessions, and similar activities.

Work effort estimates were obtained by two methods: (1) by a detailed analysis of the times required for individual decontamination tasks, and (2) by extrapolation from experience in the AFHB.

The detailed analysis led to an estimate of approximately 240,000 person-hours for the in-building work. The staff assumes that an approximately equal amount of time will be expended by the same workers in activities outside the reactor building; hence, the estimated work effort for decontaminating the reactor building is about 480,000 person-hours.

Many of the time estimates for individual tasks were quite conservative. Estimates of certain tasks (e.g., semiremote and hands-on decontamination operations) using alternative, less conservative, procedures gave lower estimates. On the basis of such judgmental considerations, the staff infers that the work effort for decontaminating the reactor building probably will not be less than about 300,000 person-hours.

The total accountable work effort by workers with assignments inside the AFHB amounted to about 1.5 person-hours for each square foot of surface decontaminated (see Sec. 5.1.3.2). This number is based on the known work effort by decontamination workers and an estimate that each person-hour of decontamination work (setting up and operating the water jet, scrubbing, wet vacuuming, etc.) required an additional 3/4 hour of support operations (radiation surveys, monitoring and stand-by duty by radiation control personnel, and plant operations, maintenance, and construction activities). The proportion of support to decontamination activities will be greater in the reactor building because of the work involved in providing access to surfaces that cannot be reached from existing structures,** constructing barriers and rerouting ventilation to prevent spread of decontamination in the large open areas, providing essential building services (e.g., additional lighting, electrical outlets and water for decontamination operations), and managing the materials and large pieces of equipment that must be moved. The staff assumes that this additional work may increase the amount of support effort to as much as 2½ hours of support work for each hour of decontamination work, which would increase the work effort per unit of decontaminated area from 1.5 to 3 person-hours/ft^2. This leads to an estimate of 900,000 person-hours of work effort for cleaning up the 300,000 ft^2 of contaminated building and equipment surfaces inside the reactor building. The staff regards this estimate as an approximate upper bound.

The preceeding estimates do not include the work effort for defueling operations or for cleanup of the primary system.

The staff assumes, on the basis of the schedule provided by the licensee (Fig. 1.4), that work on initial decontamination of the reactor building (i.e., decontamination to the level at which defueling activities can start) will extend over a period of at least 18 months and, in view of the uncertainty in the work effort that will be required and in the choice of water removal alternatives, could take three times as long. On the same basis, completion of decontamination activities to the point where ALARA criteria have been attained will require at least 36 months

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*The licensee's objective is to maintain a 60 percent productive in-building effort; on the basis of experience in the AFHB, the staff considers an estimate of 50 percent to be more realistic.

**Essentially all of the decontamination work in the AFHB has been to date on floors and walls within easy reach; ceilings and upper walls have not yet been decontaminated.

and will probably take much longer. Decontamination work will not be complete when Phase II operations begin. A useful estimate of the time that will be required to complete decontamination to ALARA levels cannot be given at this time because of the following unknown factors: the delays that will be introduced by post Phase II operations, the contamination that will be generated by these operations, and the level of effort, which is largely controlled by the rate at which funds can be made available to support the decontamination work. The staff assumes, for calculational purposes, that decontamination work performed in connection with Phase I and II operations will be carried out over a period that may range from 18 to 54 months. This assumption allows for the possibility that decontamination work that goes beyond the initial objectives described in the introduction to Section 5 may be deferred.

5.2.4 Effluents and Releases to the Environment

5.2.4.1 Normal Operations

To date, the largest release to the environment from the reactor building due to cleanup activities was the 44,000 Ci of Kr-85 that was vented to the outside atmosphere during the period from June 28, 1980, to July 11, 1980. Most of the other radionuclides present in the reactor building atmosphere before purging (approximately 1 Ci of tritium in the form of tritiated water vapor (HTO), less than 4×10^{-4} Ci of I-129 and less than 6×10^{-5} Ci of each of the radionuclides Sr-90, Cs-137, and Cs-137--see Table 5.8) were removed by the HEPA filters in the venting system.

Releases of the Kr-85 gas were controlled to limit the cumulative maximum individual offsite doses to the public resulting from the purge to less than the annual dose design objectives (15 mrem to the skin, 5 mrem whole-body) of Appendix I to 10 CFR Part 50.

The offsite dose that resulted from the purge is given in Section 5.2.5.2.

Periodic releases are made of much smaller amounts of Kr-85 generated by outgassing, primarily from the contaminated water in the basement. The dates of the releases, limits for August and monthly totals for September through November are given in Section 5.2.1.1. Venting is now being done on a monthly basis with one release a few days before each entry and another during the entry period. After decontamination operations are under way, continuous venting may be desirable to minimize worker exposure during decontamination operations. The staff assumes, on the basis of the data given in Section 5.2.1.1, that 15 Ci/month is a conservative upper bound for the Kr-85 releases from the reactor building during building decontamination operations. The releases decreased each month during the period from August 1980 through December 1980, and will probably be around 1 Ci/month before July 1981.

Other effluents and releases associated with decontamination of the reactor building and equipment include solids, liquids, and airborne particles.

Solids generated by decontamination activities in the reactor building come from sludge debris and contaminated equipment removed from the the building, contaminated filters from vacuuming operations, solid residue removed from contaminated liquids by liquid processing operations, and contaminated laundry and materials (e.g., wipes) used for decontamination operations. The effluents and releases from these materials are associated with waste disposal operations which are treated in Section 8.

Decontamination activities within the reactor building initially will not result in any direct releases of liquid to the environment. The volumes of liquids that will be used for decontaminating the building and equipment surfaces inside the reactor building are estimated by the staff to be zero to 370,000 gallons of water for remote wash operations 70,000 to 230,000 gallons of water for semiremote wash operations and 30,000 to 50,000 gallons of detergent solution, about 11 percent liquid detergent of unknown composition* and 89 percent water. The staff assumes that the most likely volume of liquids that will be generated by all building and equipment surface decontamination operations is 150,000 gallons of water and 40,000 gallons of detergent solution. The staff estimates that at least 90 percent of the plateout in the reactor building will be

*The detergent that was used for scrubbing operations in the AFHB was Radiac Wash, a proprietary chemical solution that contains ethylenediaminetetraacetic acid (EDTA) and other chemicals. This same detergent probably will be used for scrubbing operations in the reactor building.

removed in the water, and most of the remainder will be removed in the detergent solution. Processing and disposal of these liquids are discussed in Section 7. The operations are expected to last 18 to 54 months.

If the water in the basement were not removed prior to building decontamination operations or isolated by a vapor barrier from the rest of the building, most of the tritium released from the reactor building would be tritiated water vapor from evaporation of the water in the basement. The maximum possible release of 2500 Ci of tritium would occur if all of the water in the basement were removed by evaporation. If saturation conditions (100 percent humidity) could be maintained with the building ventilation system in operation (at 50,000 cfm), the entire 700,000 gallons of water would evaporate in about a year. (This would correspond to an evaporation rate of about 0.001 pounds of water per square foot per minute.) The concentration of tritium in the air under these conditions would be 2×10^{-5} μCi/mL, which is larger than the guidelines for restricted areas in 10 CFR 20 by a factor of 4.

These limiting conditions are unrealistic and unattainable; however, the staff condiders an evaporation rate of about 20 percent of this limit to be a reasonable bounding case if the building ventilation system were in operation and no special precautions were taken to prevent evaporation of the water. If this situation continued for a period of 18 months, approximately 750 Ci of tritium in the form of tritiated water vapor would be released to the atmosphere. Under these conditions the average concentration of tritium in the reactor building atmosphere would be 4×10^{-6} μCi/mL.

The staff estimates that the amount of tritiated water that will evaporate and be released from semiremote operations will be about 2 Ci. This represents a lower bound to the release of tritium that would occur if the water in the basement were removed prior to decontamination or effectively isolated. Under these conditions the average concentration of tritium in the reactor building atmosphere would be negligible.

Total airborne radioactivity discharged to the environment as a consequence of decontamination operations in the reactor building is estimated by the staff to be about 10^{-4} Ci (excluding H-3). The ventilation flow rate is 50,000 cfm. The air cleaning system contains two stages of HEPA filtration. A penetration factor of 10^{-3} has been used in calculating the releases (see Sec. 5.1.4.1). The release concentrations of principal radionuclides are listed in Table 5.10.

Table 5 10. Airborne Releases of Principal Radionuclides Expected to Occur from Decontamination of Surfaces in Reactor Building under Normal Conditions

Radionuclide[a]	Total Release (Ci)	Concentrations (μCi/mL)
H-3	2 to 750	1×10^{-8} to 4×10^{-6}
Cs-137	8.1×10^{-5}	7.4×10^{-14}
Cs-134	1.4×10^{-5}	1.3×10^{-14}
Sr-90	5.0×10^{-6}	4.5×10^{-15}

[a]Other fission products are present; however, on the basis of analysis of dissolved radionuclides in the reactor building sump water, for which Sr-89, Sr-90, Cs-134, and Cs-137 constitute greater than 99 percent of the MPC contribution (see Appendix J), and which has a radionuclide distribution similar to that of the AFHB sump water, the contributions of each of the other radionuclides to airborne releases are expected to be less than 0.1 percent and have not, therefore, been listed.

These estimates are regarded as upper bounds. If a more realistic estimate, based on the use of a penetration factor of 0.001 for each of the two filters in series (i.e., a total penetration factor of 10^{-6}) were used, all numbers in Table 5.10 would be reduced by a factor of 1000.

The staff estimates the amount of plateout on readily accessible surfaces on the 305-ft and higher elevations of the reactor building to be 2 Ci. An uncertain additional amount, assumed by the staff to be less (probably much less) than 100 Ci, is present in less accessible locations, such as in floor drains and in the basement (counting only the contaminated film that will remain after the sludge has been removed) and inside the D rings. This plateout will be dissolved or suspended in the decontamination liquids and removed with them. For computational purposes, the staff conservatively assumes that about 0.1 percent of the surface plateout on the 305-ft and higher elevations may become airborne during decontamination operations (primarily during water jet operations). The fraction of contamination from less accessible locations on the 305-ft elevation and above that becomes airborne will be less; the staff has made the conservative assumption that an equal contribution to airborn contamination might occur from these sources. Conditions in the basement are largely unknown. In view of the relatively small volume of the basement (less than 5 percent of the total volume of the reactor building) and area (about 20 percent of the total area), the staff assumes that the contribution to airborne contamination from the basement area will not exceed the contribution from the 305-ft and higher elevations.

The sludge, which has been estimated to have a wet volume of about 100 ft^3 and a radioactive content of about 160 Ci (see subsection in Sec. 5.2.2.1 on alternatives for removing the sludge), will be removed primarily by resuspension in the sump water before it is pumped out. The contribution of this operation to airborne releases will be negligible compared to other contributions. It may be necessary to dislodge some fraction, up to about 5 percent of the total, by use of a water jet. The staff again has conservatively assumed that up to 0.1 percent of this residue might become airborne during this operation.

5.2.4.2 Accident Scenarios

The types of accidents that could occur during reactor building decontamination include a liquid spill, either before or after processing the liquid, dropping of a solid waste package, a fire in a barrel of contaminated trash, and failure of HEPA filters in the ventilation system. Effluents and releases from accidents associated with processing the liquids from the reactor building decontamination are discussed in Section 7. Effluents and releases from accidents in the handling of solid wastes are discussed in Section 8.

A failure of one of the HEPA filters in the ventilation system would increase the airborne particulate release described in Table 5.10 until the ventilation system is secured (15 minutes). Based on the analysis of Section 7, the accident releases are shown in Table 5.11.

Table 5.11. Estimated Airborne Releases from a HEPA Filter Failure during Decontamination of the Reactor Building Surfaces

Radionuclide[a]	Total Release (Ci)	Concentrations (μCi/mL)
Cs-137	8.1×10^{-5}	3.8×10^{-9}
Cs-134	1.4×10^{-5}	6.6×10^{-10}
Sr-90	5.0×10^{-6}	2.4×10^{-10}

[a]Other fission products are present; however, on the basis of analysis of dissolved radionuclides in the reactor building sump water, for which Sr-89, Sr-90, Cs-134, and Cs-137 constitute greater than 99 percent of the MPC contribution (see Appendix J), and which has a radionuclide distribution similar to that of the AFHB sump water, the contributions of each of the other radionuclides to airborne releases are expected to be less than 0.1 percent and have not, therefore, been listed.

5.2.5 Environmental Impacts

5.2.5.1 Occupational Doses

The same two methods used to estimate the work effort were used to estimate the occupational dose that will be incurred by workers who must enter the reactor building to carry out decontamination operations: (1) by extrapolation from experience in the AFHB, and (2) by a detailed analysis of the doses expected from individual decontamination tasks. The first of these methods gave the higher work effort estimate. For the occupational dose the reverse was true; the first method gives the lower occupational dose estimate. The staff infers that this result may be due to more careful attention by the licensee to the use of strategies that would minimize worker exposure and less careful attention to efficient deployment of the work force.

Experience in the AFHB up to September 1, 1980, gave a cumulative personnel dose per unit area of surface decontaminated of 0.8 person-rem/ft^2. Some of the more difficult cubicles with higher radiation fields (e.g., the sump cubicle) have been left until last; the staff estimates that these cubicles may increase the dose to as much as 1.1 person-rem/ft^2 by the time the AFHB decontamination is completed (see Sec. 5.1.5.1). This dose rate is based on the estimate that 3/4 hour of support work inside the AFHB was required for each hour of decontamination work. The staff estimates that about 2.5 hours of support work will be required for each hour of decontamination work in the reactor building. This would increase the dose per unit area from 1.1 person-rem/ft^2 to 2.2 person-rem/ft^2. The initial radiation fields in the AFHB were higher than the initial radiation fields in the reactor building on the 305-ft and higher elevations; however, in view of the unknown (and probably much higher) radiation fields to which workers will be exposed during decontamination operations in the reactor building basement, the staff assumes that the AFHB and reactor building radiation levels are, on the average, comparable. On the basis of these considerations, the estimated occupational dose for decontamination and related work inside the reactor building is 660 person-rem.

A detailed analysis of the occupational dose for the individual tasks involved in decontaminating the reactor building leads to an initial estimate of 22,000 person-rem for the total occupational dose for decontaminating the reactor building. This is an initial bounding estimate, and for two reasons, it is much higher than the actual expected dose. First, the work effort estimates are very conservative. Estimates of the work effort required for semiremote wash, scrubbing, and wet vacuuming operations on the 305-ft and higher elevations by alternative and probably more realistic methods gave much smaller work-effort and occupational dose estimates for these tasks. On the basis of these considerations, the staff believes that 15,000 person-rem would be a more realistic upper bound even if no credit was allowed for the use of effective shielding and work location strategies.

A considerable reduction in occupational dose can, however, be gained by such strategies as the use of shielding and careful planning to choose work stations where the exposure is minimum during the decontamination operations. Experience in the AFHB has shown that the actual exposure is less than what would be calculated from the product of the radiation field and work duration alone by a factor of 8 for decontamination work, and in some cases as much as 100 for construction and maintenance work.[18] Arguments can be given to support either the assumption that the gain will be larger for the open spaces in the reactor building or that it will be smaller; the staff has made the conservative assumption that it will be smaller and assumes that an overall factor of 5 is reasonable.

On the basis of the preceding considerations, the staff believes that 3000 person-rem is a probable upper bound for the cumulative occupational dose that will be incurred in decontaminating the reactor building.* Bounding values for the estimated cumulative occupational dose are, therefore, 660 and 3000 person-rem.

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*If a factor of 3 were used to compensate for the conservatism in the dose estimates for the individual tasks (which is consistent with the alternative decontamination task assessments), and a factor of 10 were used to take into account the gain by use of a well-planned decontamination strategy, the estimated occupational dose would be reduced to 733 person-rem, which is close to the estimated lower bound obtained by extrapolation from the AFHB experience.

The occupational dose includes an estimated 20 to 60 person-rem due to inhalation of tritiated water vapor. The tritiated water will be produced by evaporation from the water in the basement, which has a tritium concentration of about 1 μCi/mL. The estimate for the dose from inhalation is based on the expectation that the water will remain in the basement for about one year, and that during this time, 50,000 to 150,000 person-hours of work effort will be spent in areas in which the tritium concentratin from tritiated water vapor will be about 2×10^{-6} μCi/mL.*

The estimated bounds for the work effort and duration of the decontamination operations were 300,000 to 900,000 person-hours and 18 to 54 months, respectively (see Sec. 5.2.3.2). The staff assumes that these two sets of numbers correspond; i.e., an average rate of expenditure of effort of 50,000 person-hours per quarter for a period of 6 to 18 quarters is assumed.

Based on the cumulative occupational doses and work effort given above, the staff estimates that the average dose rate for individual workers lies in the range from 0.73 mrem/hr to 10 mrem/hr. At the lower rate, a total cumulative dose of 660 person-rem would be incurred with a work effort of 900,000 person-hours over a period of 18 quarters, for which the worker contingent would be 100 workers if a 40-hour work-week were maintained. At the higher dose rate (10 mrem/hr) a total cumulative dose of 3000 person-rem would be incurred with a work effort of 300,000 person-hours over a period of six quarters, for which a contingent of 500 workers limited to 100 hours per quarter would be needed.

The expected number of additional cancer mortalities in this work force will range from 0.09 to 0.4. The average added probability that a worker in this group would eventually die of cancer would range from about one in 1100 to one in 1300 if he/she participated in the reactor building decontamination work for the full duration of the cleanup effort. The expected number of additional genetic effects in the offspring of these exposed workers ranges from 0.2 to 0.8.

5.2.5.2 Offsite Doses

The largest offsite dose that is a consequence of decontamination operations in the reactor building is that which occurred during the period from June 28 to July 11, 1980, when the Kr-85 was purged from the reactor building atmosphere.

The point of maximum exposure during the purge was at a location about 0.4 mile from the site in an east-southeast direction. If a person had remained at this location throughout the purge he/she would have received a beta skin dose of 4.5 mrem and a whole-body gamma dose of 0.05 mrem.

Dose estimates for decontamination operations in the reactor building subsequent to September 1, 1980 are based on the source terms developed in Section 5.2.4.1 and listed in Table 5.10. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3. The dose estimates are listed in Table 5.12 for the lower tritium range in Table 5.10, and Table 5.13 is for the upper tritium range in Table 5.10. The 50-mile whole-body cumulative population dose received by the human population during these activities is estimated to be 0.02 to 6 person-rem. The lower value is for the lower tritium range in Table 5.10, and the upper value is for the upper tritium range in Table 5.10.

5.2.5.3 Postulated Accident Effects

No accidental releases occurred during purging of the reactor building atmosphere. The type of accident for which dose estimates are made here is rupture of a HEPA filter during reactor building decontamination. This accident scenario is described in Section 5.2.4.2 and the source terms are listed for it in Table 5.11. Doses are estimated here for a HEPA filter failure occuring during the surface decontamination operation in the reactor building. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses is discussed in Section 10.4. The dose estimates to the maximum exposed individual are listed in Table 5.14.

*In order to maintain the average atmospheric concentration at this level or below, it will be necessary to arrange barriers and ventilation patterns so that the fraction of water vapor in the reactor building atmosphere that comes from the water in the basement does not exceed about 10 percent of the saturation concentration at 75°F.

Table 5.12. Dose Estimates for the Maximum Exposed Individual for Reactor Containment Building Decontamination for Lower Estimated Tritium Release[a]

Location	Pathway	Dose (mrem)[b]		
		Total-Body	Bone	Liver
Nearest garden[c]	Inhalation	1.4×10^{-4}	3.1×10^{-5}	1.4×10^{-4}
	Ground Shine	9.5×10^{-6}	9.5×10^{-6}	9.5×10^{-6}
	Vegetable Use	6.7×10^{-4}	8.0×10^{-4}	6.9×10^{-4}
	TOTAL	8.2×10^{-4}	8.4×10^{-4}	8.4×10^{-4}
Nearest milk goat	Inhalation	7.3×10^{-5}	1.2×10^{-5}	7.5×10^{-5}
	Ground Shine	9.1×10^{-6}	9.1×10^{-6}	9.1×10^{-6}
	Goat Milk Use	3.0×10^{-4}	2.8×10^{-4}	5.7×10^{-4}
	TOTAL	3.8×10^{-4}	3.0×10^{-4}	6.5×10^{-4}
Nearest cow and garden	Inhalation	1.6×10^{-4}	3.5×10^{-5}	1.6×10^{-4}
	Ground shine	1.4×10^{-5}	1.4×10^{-5}	1.4×10^{-5}
	Vegetable Use	8.0×10^{-4}	1.2×10^{-3}	8.4×10^{-4}
	Cow Milk Use	2.6×10^{-4}	2.8×10^{-4}	4.6×10^{-4}
	TOTAL	1.2×10^{-3}	1.5×10^{-3}	1.5×10^{-3}

[a]Dose estimates in this table are based on the lower range of tritium in Table 5.10.

[b]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden are for children the dose estimates for the nearest milk goat for total-body are for adults and for bone and liver are for infants, and the dose estimates for the nearest cow and garden are for children.

[c]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 5.13. Dose Estimates for the Maximum Exposed Individual for Reactor Containment Building Decontamination for Higher Estimated Tritium Release[a]

Location	Pathway	Dose (mrem)[b]		
		Total-Body	Bone	Liver
Nearest garden[c]	Inhalation	5.2×10^{-2}	3.1×10^{-5}	5.2×10^{-2}
	Ground Shine	9.5×10^{-6}	9.5×10^{-6}	9.5×10^{-6}
	Vegetable Use	1.8×10^{-1}	8.0×10^{-4}	1.8×10^{-1}
	TOTAL	2.3×10^{-1}	8.4×10^{-4}	2.4×10^{-1}
Nearest milk goat	Inhalation	2.7×10^{-2}	1.2×10^{-5}	2.7×10^{-2}
	Ground Shine	9.1×10^{-6}	9.1×10^{-6}	9.1×10^{-6}
	Goat Milk Use	2.0×10^{-1}	1.1×10^{-3}	2.1×10^{-1}
	TOTAL	2.3×10^{-1}	1.1×10^{-3}	2.4×10^{-1}
Nearest cow and garden	Inhalation	5.7×10^{-2}	3.5×10^{-5}	5.7×10^{-2}
	Ground shine	1.4×10^{-5}	1.4×10^{-5}	1.4×10^{-5}
	Vegetable Use	2.0×10^{-1}	1.2×10^{-3}	2.0×10^{-1}
	Cow Milk Use	8.0×10^{-2}	2.8×10^{-4}	8.1×10^{-2}
	TOTAL	3.4×10^{-1}	1.5×10^{-3}	3.4×10^{-1}

[a]Dose estimates in this table are based on the upper range of tritium in Table 5.10.

[b]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden are for children the dose estimates for the nearest milk goat for total-body are for adults and for bone and liver are for infants, and the dose estimates for the nearest cow and garden are for children.

[c]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 5.14. Estimates of Offsite Doses to the Maximum Exposed Individual Caused by Failure of a HEPA Filter during Decontamination of the Reactor Building

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	4.2×10^{-5}	5.4×10^{-4}	7.5×10^{-5}
	Ground Shine	5.5×10^{-4}	5.5×10^{-4}	5.5×10^{-4}
	Vegetable Use	1.1×10^{-2}	4.7×10^{-2}	1.2×10^{-2}
	TOTAL	1.2×10^{-2}	4.8×10^{-2}	1.3×10^{-2}
Nearest milk goat	Inhalation	6.9×10^{-5}	2.3×10^{-4}	5.5×10^{-5}
	Ground Shine	5.5×10^{-4}	5.5×10^{-4}	5.5×10^{-4}
	Goat Milk Use	9.7×10^{-3}	6.6×10^{-2}	7.6×10^{-2}
	TOTAL	1.0×10^{-2}	6.7×10^{-2}	7.7×10^{-2}
Nearest cow and garden	Inhalation	2.9×10^{-5}	3.7×10^{-4}	5.1×10^{-5}
	Ground shine	5.5×10^{-4}	5.5×10^{-4}	5.5×10^{-4}
	Vegetable Use	1.1×10^{-2}	4.7×10^{-2}	1.2×10^{-2}
	Cow Milk Use	1.9×10^{-3}	1.1×10^{-2}	9.7×10^{-3}
	TOTAL	1.3×10^{-2}	5.9×10^{-2}	2.2×10^{-2}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The dose estimates for the nearest garden and for the nearest cow and garden locations are for children, and the dose estimates for the nearest milk goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

5.2.5.4 Psychological-Socioeconomic Effects

As indicated in Section 5.2.3.2, the staff estimates that from 300,000 to 900,000 person-hours would be required to complete the reactor building decontamination process. Assuming a 4.5-year schedule--winter of 1982 to spring of 1986--the number of workers employed at any given time for work on decontaminating the reactor building is estimated to average 100, with a peak of 200 in 1983-1984.

Although the manpower requirement for reactor building and AFHB decontamination are large relative to other cleanup operations, the community level impacts will not be significant. Most workers, particularly those involved in the actual decontamination work, will be subject to high job turnover rates because of occupational dose limitations. These relatively transient workers will not move into the community with their families, but will also seek to "double-up" in hotel, motel, and rooming house accommodations. Moreover, because of their transient status, such workers will tend to spend money only for necessities and recreation, retaining most of their incomes for their families and for spending in their hometown areas. Finally, relocating workers will seek to minimize housing and transportation costs by selecting locations that offer acceptable amenities. For most workers, the greater Harrisburg area will provide adequate housing choice and community amenity levels. Because of the small size of worker housing demand relative to the population of the Harrisburg area, the staff judgment is that in-moving workers will not impose a significant impact on the housing market or on community facilities.

Considering the procedures to be undertaken and the offsite releases under normal operations and under hypothesized accident conditions, the staff believes that the possible psychological and socioeconomic impacts during reactor building decontamination would be similar to those described in Section 5.1.5.4.

5.2.6 Economic Costs

In addition to the work effort described in Section 5.2.3.2, costs will be incurred for such items as: (1) construction and operation of the containment service building, (2) installation of essential services (electrical power, steam, air and localized ventilation), (3) refurbishment of equipment (including the polar crane and building coolers), (4) materials (chemicals, shielding, venting, filters, etc.), (5) equipment (jib cranes hoisting and rigging gear, hydrolasers, casks, steam cleaners etc.), (6) technical services to prepare the detailed procedures for implementation of the decontamination, and (7) training of personnel.

Numerous alternatives have been discussed, and in reality, the reactor building cleanup probably will entail a combination of these alternatives and can therefore be estimated as a range of costs. For the purpose of estimating costs, the staff has examined two cases: (1) a case in which extensive shielding is required and greater than expected difficulty is encountered in removing the contamination from surfaces and equipment (worst case) and (2) a case in which conditions are essentially as expected and supported by recent surveys (best case).

The staff estimates that the costs for reactor building decontamination will range from $25 million for the best case to over $60 million for the worst case. The principle catagories of these costs are listed in Table 5.15. The assumptions used to make these estimates are given in Appendix K.

Table 5.15. Economic Costs for Reactor Building Decontamination (thousands of dollars)

Item	Best Case	Worst Case
Labor	13,400	40,100
Equipment	3,500	5,900
Facilities	8,500	17,000
TOTAL	25,400	63,000

References--Section 5

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2. Code of Federal Regulations, Title 10, Part 20, Office of the Federal Register.

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4. Memorandum from B.A. Irving and V.F. Bouchard, Vikem Industries, Inc., to J. Renshaw, GPU/Met Ed, October 27, 1979.

5. T.R. Block, "Auxiliary and Fuel Handling Building Decontamination Program," presented at an information briefing by Metropolitan Edison Co. at Crawford Station, Middletown, PA, September 23, 1980.

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9. Order for Temporary Modification of Operating License No. DPR-73, issued by the U.S. Nuclear Regulatory Commission on June 12, 1980, relative to the Technical Specifications for Three Mile Island Unit 2.

10. "Three Mile Island Unit 2 Recovery Quarterly Progress Report for the Period Ending September 30, 1980", Section 4 Metropolitan Edison Co./General Public Utilities, 1980.

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12. Memorandum from T.L. Gilbert to W.K. Lehto, Argonne National Laboratory, "Entry Data," December 8, 1980.

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14. Letter (TLL 541) from R.C. Arnold, Met-Ed/GPU, to H.R. Denton U.S. Nuclear Regulatory Commission, November 4, 1980.

15. Letter (TLL 684) from G.K. Hovey, Met-Ed/GPU, to L.H. Barrett, U.S. Nuclear Regulatory Commission, December 23, 1980.

16. D.O. Campbell, E.D. Collins, L.J. King and J.B. Knaver, "Evaluation of the Submerged Dimineralizer System (SDS) Flowsheet for Decontamination of High-Activity-Level Water at the TMI-2 Nuclear Power Station," Oak Ridge National Laboratory, ORNL/TM-7448, July 1980.

17. R.E. Brooksbank and W.J. Armento, "Post-Accident Cleanup of Radioactivity at the Three Mile Island Nuclear Power Station," Oak Ridge National Laboratory, ORNL/TM-708, February 1980.

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6. REACTOR DEFUELING AND PRIMARY SYSTEM DECONTAMINATION

The operations leading to and including removal of the damaged reactor core from the pressure vessel and subsequent decontamination of the reactor primary cooling system (RCS) components (reactor pressure vessel, steam generators, pressurizer, pumps and associated piping) are considered in this section. The activities discussed are:

- Handling of the radioactive water in the primary cooling system during reactor defueling and primary system decontamination (Sec. 6.1). The water processing and disposal alternatives are discussed in Section 7.
- Inspection of reactor system components prior to defueling and decontamination to determine the condition of the primary cooling system and reactor core (Sec. 6.2).
- Removal of the reactor pressure vessel (RPV) head and internal components to gain access to the reactor core (Sec. 6.3).
- Removal of the reactor core fuel elements (Sec. 6.4).
- Decontamination of the primary system components to remove fuel debris and other particulate matter from the system and to reduce fission product plateout on internal surfaces (Sec. 6.5).

6.1 HANDLING OF PRIMARY COOLANT SYSTEM WATER

Processing and treatment of the primary water must precede and continue during the operations leading to reactor defueling and primary system decontamination (see Sec. 7 for alternatives and details). Following the initial decontamination of the primary water, inventories will vary depending upon the particular phase of the defueling and decontamination activities. Appendix F discusses potential for water reuse to minimize increasing the total inventory of contaminated water.

The various phases of water handling for reactor defueling and primary system decontamination are:

- Preparations for reactor head removal
- Removal of RPV head and internals
- Reactor defueling
- Primary system flush and drain
- Decontamination of primary system components.

6.1.1 Preparation for Reactor Head Removal

Following initial primary water processing (see Sec. 7.1.5) the primary system would be drained from an initial inventory of 96,000 gallons to about 26,000 gallons. Primary system circulation would be maintained by existing TMI-2 piping systems. The primary water chloride concentration would be reduced to acceptable levels prior to opening the reactor coolant system to air.

6.1.2 Removal of RPV Head and Internals

Either after or during reactor pressure vessel (RPV) head removal, the fuel transfer canal (FTC) would be filled with primary-system-grade water to the normal refueling level. Filling the FTC would add about 360,000 gallons of water. Circulation of this water would be maintained by existing TMI-2 systems. Should the radionuclide concentration increase above the target value of

0.01 μCi/mL (exclusive of tritium), additional water processing would be conducted using either one of the alternatives described in Section 7.1.2 or the existing purification systems or a supplemental purification system.

6.1.3 Reactor Defueling

After removal of the upper internals (see Sec. 6.2) the spent fuel pool (SFP) would be filled, if not already filled, and connected to the FTC. The SFP contains about 700,000 gallons of water. This would increase the total inventory of water to about 1,100,000 gallons. Circulation of this water would be maintained using existing TMI-2 systems. Should the radionuclide concentration increase above the targeted value of 0.01 μCi/mL (exclusive of tritium), additional water processing would be conducted using either one of the alternatives described in Section 7.1.2.

6.1.4 Primary System Flush and Drain

After completion of reactor defueling, the SFP will be isolated from the FTC and the latter will be drained. At this point, the primary system water inventory will have been reduced to about 26,000 gallons. The RPV head then will be installed and the primary system will be filled with water to bring the inventory up to 96,000 gallons. The flushing and draining operations then will be initiated (see Sec. 6.5). The staff assumes that during the flush and drain operations, an additional 150,000 gallons of water will be added as makeup water during the flushing operations, to provide about 2½ system volumes. Processing of this water is covered in Section 7.

6.1.5 Decontamination of the Primary System Components

After completion of primary system flush and drain, the primary system will once again be filled (96,000 gallons) and chemical decontamination will be initiated. Depending upon the process selected, between 100,000 and 500,000 gallons of water will be required. The processing of this water is discussed in Section 7.1.3. The physical operations and alternatives are discussed in Section 6.5.

6.2 REACTOR COOLANT SYSTEM (RCS) INSPECTION

Since current knowledge about the condition of the TMI-2 reactor core and reactor coolant system is limited, it is necessary for workers to enter the reactor building to inspect components before plans can be finalized for primary system decontamination and reactor defueling. Inspections will be coordinated with the reactor building decontamination (Sec. 5.2), removal of the RPV head (Sec. 6.3), defueling (Secs. 6.3 and 6.4), and primary coolant system decontamination (Sec. 6.5). The objectives of these inspections would be to determine the condition of the core and provide ongoing monitoring of important reactor coolant system parameters during the succeeding steps in the dismantling and cleanup process.

6.2.1 Status and Specific Considerations

The data available concerning the status of the TMI-2 reactor do not provide sufficient information to determine the extent of damage in the reactor core. The best information on the possible internal condition of the reactor was derived from attempts to reconstruct the progress of the accident, based on data gathered during the accident.[1,2] This information leads to the conclusion that a large fraction of the fuel rods have ruptured and there has been oxidation of Zircaloy in the core (about 50% of the total core inventory of Zircaloy, i.e., fuel cladding, control-rod guide tubes, and instrument tubes, has oxidized).

There has been structural damage in the core as deduced from fission product release, hydrogen generation, core thermocouple readings, core pressure drop, and a significant system pressure transient. The reactor pressure vessel, internals, and head have been exposed to high temperatures, as indicated on incore thermocouples and deduced from the behavior of the neutron detectors. The high temperatures may have caused fragmentation of some of the fuel pellets. This fragmented fuel may have been distributed throughout the core region and could obstruct the view needed for visual inspection and defueling. The temperature of parts of the core during the accident may have been sufficiently high to have caused some melting and fusing of steel end fittings and spacer grids on the fuel assemblies. There is, however, no indication that significant melting of the uranium oxide fuel pellets has occurred. Care must be taken during in-vessel activities to avoid unnecessary scattering of damaged fuel. There may have been some hydriding of the Zircaloy in the core; however, large-scale hydriding of the Zircaloy in the core is not

expected, and any zirconium hydrides that did form should not be present as finely divided particles. (Finely divided zirconium hydride is known to be pyrophoric and could thus become a hazard if present during the cleanup operation.) Thus, a pyrophoric hazard is highly unlikely. In any case the core will be kept constantly covered with water throughout the defueling so that even if significant hydriding exists, pyrophoric events are not possible.

6.2.2 Alternative Methods Considered

The number of inspections required will depend on the condition of the core. Since the environmental impact of the examinations is expected to be small relative to the impact of the defueling/decontamination operations, and since any attempt to consider different examination methods for different probable core conditions would generate an unmanageable number of combinations, the following discussion is limited to a single examination schedule for each of two core-condition possibilities--the best case and the worst case. The best case is an estimate of the most favorable core conditions (for which a smaller number of examinations would be needed). The worst case is for a severely damaged core (for which a larger number of examinations would be needed).

The planning and damage-assessment inspections will be conducted during all stages of the decontamination/defueling operation. The principal stages are:

1. Entry into the reactor building
2. Preparations for reactor vessel head removal
3. Removal of the reactor vessel head
4. Removal of the upper internals from the reactor vessel
5. Removal of the fuel from the reactor vessel to the spent fuel pool (this operation will involve several stages, such as fuel encapsulation).

No inspections of the core have yet been made.

Because of the requirement to satisfy ALARA considerations, the work must be conducted in as low a radiation field as is practical. Decontamination of the reactor building should therefore be underway before the first significant inspection activities.

6.2.3 Details of Methods and Facilities

Standard techniques (e.g., gamma scans, visual inspections, dimensional measurements, chemical analysis) will be used for the examination and measurement tasks. The only details considered here are those needed to estimate (for best- and worst-case conditions) occupational radiation doses,* the amounts of contaminated waste that would be generated offsite, environmental effects, and direct economic costs. The estimates given are based on the judgment of persons experienced in reactor design, operations, and maintenance. Examinations required for requalification of reactor coolant system components are excluded.

6.2.3.1 Best-Case Conditions

The bases used in making estimates for best-case circumstances are: (1) the reactor building would be decontaminated and hot spots would be shielded (see Sec. 5.2); (2) only a minimum number of examinations would be required; and (3) the workers entering the reactor building would be required to wear protective clothing and use respirators at least part of the time.

6.2.3.2 Worst-Case Conditions

The bases used in making estimates for worst-case conditions are: (1) during the period after reactor building entry but before opening the primary system, reactor building decontamination would be in progress and not all hot spots would be shielded; (2) for examinations after opening

*Radiation values given are average values for all workers. These values were calculated by evaluating the work to be performed, where the workers would be, how long the work might take, conducting scoping calculations for hot spots, and then using engineering judgment to evaluate the average radiation field.

the reactor coolant system, the reactor building decontamination would be essentially complete and hot spots would be shielded; and (3) the workers entering the reactor building would be required to wear protective clothing and use respirators (full or half-masks) at least part of the time.

6.2.3.3 Logistics, Crew Size, Work Effort, Radiation Levels, Waste Generated, and Associated Facilities

Estimates for the relevant factors are given in Table 6.1 and discussed below for the two cases.

- Crew Size. The average work crew under best-case conditions would consist of three persons--an instrumentation and control technician, a health physics technician, and a specialist/supervisor. The average work crew under worst-case conditions would consist of four people: two instrumentation and control technicians, a health physics technician, and a specialist/supervisor. The crew size would depend on the type of activity and on the location of the effort to be performed. Inspection activities would be performed on an intermittent basis beginning with reactor building entry and continuing through defueling and RCS decontamination activities.

- Work Effort. The probable work effort required for reactor primary system inspection under this alternative is 5200 person-hours under assumed best-case conditions and 41,000 person-hours under worst-case conditions (see Table 6.1). Inspections and examinations associated with activities supporting control-rod drive uncoupling, incore-detector removal or installation, gas sampling, reactor internal and external evaluations, core evaluation, and fuel removal would comprise about 85 percent of the effort under best-case conditions and 80 percent under worst-case conditions.

Table 6.1. Average Dose Rates, Work Times, and Cumulative Occupational Doses for Reactor Coolant System Inspection Operations under Best- and Worst-Case Conditions

Operation Stage	Average Dose Rate (mrem/hr)	Time in Radiation Areas (person-hr)[a]	Cumulative Occupational Dose (person-rem)[a]
Best-Case Conditions			
1. Reactor building entry	10	1,700	17
2. Preparation for vessel head removal	10	800	8
3. Primary system head removal	10	300	3
4. Plenum removal	10	500	5
5. Fuel removal	10	1,900	19
TOTAL		5,200	52
Worst-Case Conditions			
1. Reactor building entry	25	11,000	280
2. Preparation for vessel head removal	10	6,900	69
3. Primary system head removal	10	4,800	48
4. Plenum removal	10	8,500	85
5. Fuel removal	10	9,900	99
TOTAL		41,000	580

[a]Values rounded to two significant figures.

- Radiation Levels. Under best-case conditions, the average dose rate associated with primary system inspection is estimated to be 10 mrem/hr.* It is estimated that the average worker dose rate prior to reactor coolant system breach under worst-case conditions would be 25 mrem/hr because this activity would be accomplished before reactor building decontamination; all examinations made after the primary system breach under worst-case conditions would be conducted after completion of essential reactor building decontamination, so a 10 mrem/hr average dose rate is estimated.*

- Wastes Generated. Wastes that would be generated include 52 incore detector assemblies, contaminated clothing, contaminated inspection equipment, and four out-of-core nuclear detectors under best-case conditions, and eight out-of-core nuclear detectors and 25 contaminated probes under worst-case conditions.

- Special Facility Needs. A closed circuit television system and communications system would be needed, plus various types of probes for examinations (e.g., boroscopes, miniature T.V. cameras, radiation detectors).

6.2.4 Effluents and Releases to the Environment

Virtually no radioactive effluents will be released as a result of inspection activities.

6.2.5 Environmental Impacts

6.2.5.1 Occupational Doses

The expected number of additional cancer mortalities in the work force exposed to the cumulative occupational doses given in Table 6.1 would range from 0.0068 for the best case (52 person-rem) to 0.076 for the worst case (580 person-rem). The expected number of additional genetic effects in the offspring of the work force exposed to these levels would range from 0.014 to 0.151.

Since the inspections will occur intermittently throughout the decontamination and defueling operations, there will not be a separate work force for these inspections per se. The crews of three to four people (Sec. 6.2.3.3) will ideally perform inspection-related tasks along with other core-removal and primary coolant system efforts since inspection and follow up are integrally related, and inspection requires specific engineering skills (Sec. 6.2.2). However, if the inspections were done at the same level of effort during each quarter over a three-year cleanup period, up to 3400 person-hours per quarter would be devoted to inspections under the worst-case conditions. Conversely, if the cleanup takes six years and best-case conditions apply, about 200 person-hours per quarter would be required.

An individual quarterly (three-month) occupational dose of 1 rem has been selected by the staff for calculation of total work force requirements. The reasons the staff selected this value rather than limits as high as 3 rem per quarter given in 10 CFR Part 20 are provided in Appendix L. However, individuals could approach a dose of 3 rem per quarter, if necessary.

With an average radiation field of 10 mR/hr, each person would spend only 100 hours per quarter on inspection efforts so as not to exceed 1 rem/quarter. This means that because of dose considerations, each crew of four would expend only 400 person-hours per quarter; thus, about nine such crews (36 individuals) would be needed to carry out the 3400 person-hours per quarter of inspection work that the staff anticipates would be needed under worst-case conditions during a three-year cleanup period. However, under best-case conditions for inspection, a single crew of four could perform the inspections and perform additional defueling work as well.

6.2.5.2 Offsite Doses

Offsite impacts are indistinguishable compared to defueling (Section 6.4.5.3).

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*See Appendix I for justification.

6.2.5.3 Postulated Accident Effects

The staff has not identified any potential accidents from this activity that would have offsite effects.

6.2.5.4 Psychological-Socioeconomic Effects

Psychological-socioeconomic effects relative to RCS inspection are discussed in Section 6.5.5.4.

6.2.6 Economic Costs

To evaluate the economic costs associated with a particular case, all labor costs, as well as costs for equipment and facilities, need to be included. Estimates of the economic costs for the best-and worst-case conditions are given in Table 6.2. These costs are estimated on a consistent, relative basis, and do not necessarily reflect all costs on an absolute basis. Thus, these costs should be considered only as relative costs for the two scenarios (best case versus worst case). The basis for the costs shown here are given in Appendix K.

Table 6.2 Economic Costs for RCS Inspection (thousands of dollars)

Item	Best Case	Worst Case
Labor	1230	4910
Equipment	160	980
Facilities	0	0
TOTAL	1390	5890

About 85% of the labor costs shown in Table 6.2 are estimated to be incurred in preparation for work inside the reactor building. The majority of this effort consists of training and procedure preparation. Estimated equipment costs consist primarily of special instruments for the necessary evaulations at various stages of plant cleanup. Instrumentation needs will be better defined as detailed planning gets underway and, in some cases, as work is performed. However, the cost estimates are not particularly sensitive to these needs and should not be significantly changed regardless of specific instrument needs. Facility needs are as discussed under Special Facility Needs in Section 6.2.3.3.

6.3 REMOVAL OF THE RPV HEAD AND INTERNALS

Removal of the reactor pressure vessel head (RPVH) and the internal components is necessary to gain access to the reactor core for defueling, and to the bottom of the reactor vessel to remove fuel debris.

Before the RPVH is removed, other decontamination activities directly related to RPVH removal will have been completed: (1) the primary water will have been decontaminated and the chloride concentration reduced to a level sufficient to proceed with RPVH removal, (2) the principal reactor building support equipment and systems needed to proceed with head removal will have been decontaminated and put into satisfactory working condition, and (3) work area decontamination will have been accomplished in accordance with ALARA principles. Also, inspections of the core may be accomplished by insertion of viewing devices through a port in the head, prior to head removal.

6.3.1 Status and Specific Considerations

The reactor pressure vessel head (RPVH) is bolted to the top of the vessel. As shown in Figure 6.1, the control rod drives and the axial power shaping rod (APSR) drives extend through the

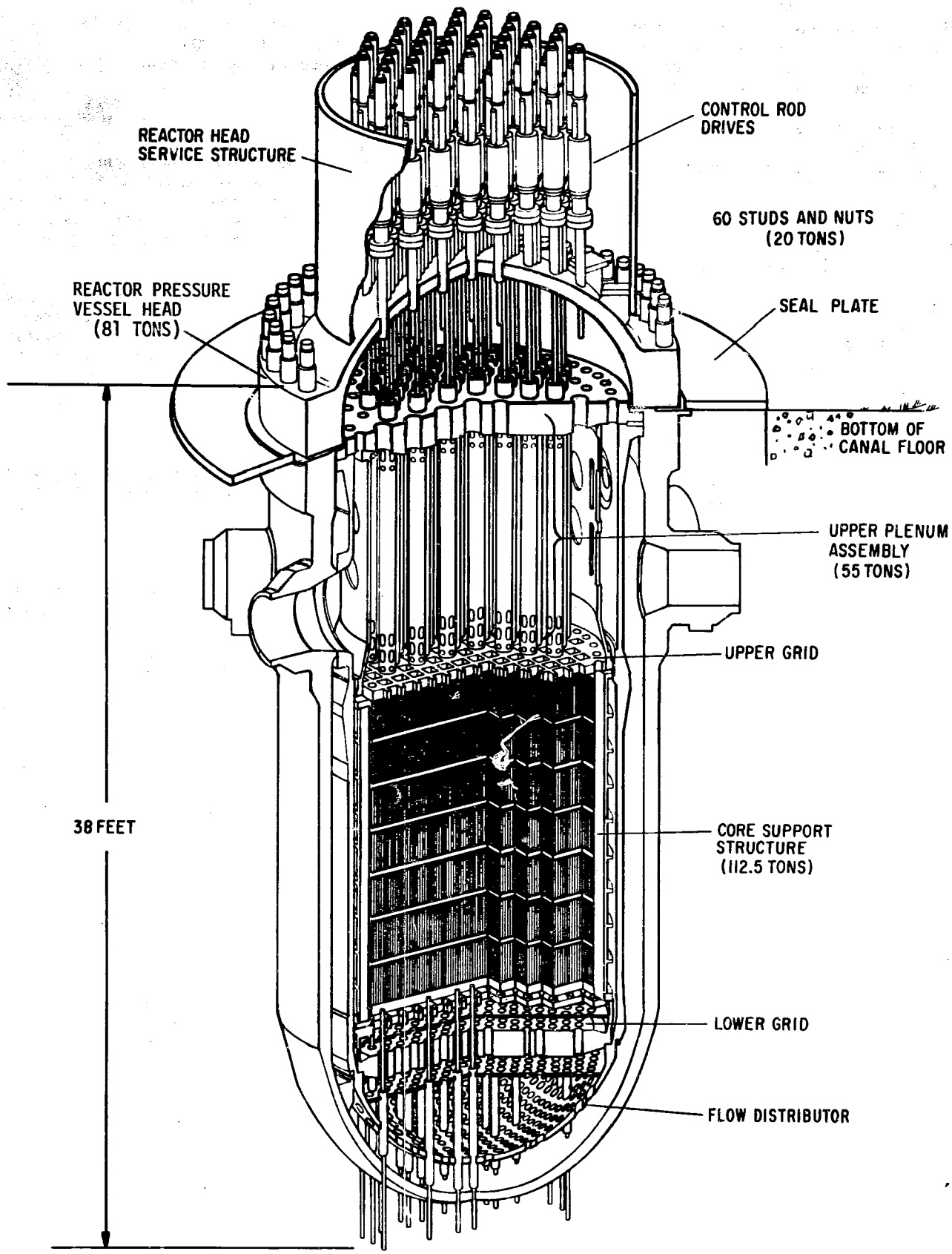

Figure 6.1. Cutaway View of a Typical PWR Similar to the TMI-2 Reactor Pressure Vessel.

RPVH. Before any core damage had occurred during the accident, the control rod drives "scrammed" the control rods to the fully inserted position so as to shut down the reactor. However, the APSRs were maintained at a partially withdrawn position. Because of the accident damage, the control rod and APSR drives may now be inoperable.

There may be fuel and fuel assembly debris and fragments in cavities, crevices, and on surfaces on the upper plenum assembly and the RPVH. There may also be fuel and fuel assembly debris and fragments in the lower grid and flow distributor of the core support assembly and throughout the reactor coolant system. High temperatures and rapidly changing pressures resulting from the accident may have caused: distortion of the RPVH; distortion, warping, and physical dislocation of the upper plenum assembly; binding of the upper plenum assembly to the core support assembly, fuel assemblies, and the control rod assemblies; distortion and warping of the core support assembly and binding of core support assembly to RPV.

The staff has recognized that the actual conditions and extent of damage in the RPV are not known and could cover a broad spectrum. Conditions and damage cannot be accurately determined until the RPVH has been removed. For evaluational purposes, two sets of postulated conditions (worst case and best case, as outlined below) have been developed. The staff is of the opinion that the courses of action developed below for each postulated set of conditions should encompass the range of anticipated conditions.

6.3.2 Alternative Methods Considered

Regardless of the exact conditions found inside the RPV, the RPVH and internals most likely would be removed by the following seven steps (see Fig. 6.1 for identification of components). The core will be constantly kept covered with water throughout the operation.

1. Preparation of Equipment.

2. Installation of the seal plate assembly. (The seal plate assembly is a large flat ring that is fastened between the reactor vessel flange and the canal floor. When this ring is installed it forms a watertight seal allowing the fuel transfer canal (FTC) to be flooded. The fuel transfer canal must be flooded for shielding purposes in order to remove the fuel from the reactor vessel to the spent fuel pool or elsewhere.)

3. Removal of the RPVH closure nuts and studs. (The closure nuts and studs are the bolts that secure the RPVH to the reactor vessel.)

4. Uncoupling of the APSR/CR drive assemblies. (The APSR and CR drives are electromechanical devices for raising and lowering the reactor control elements. Before removal of the reactor head, these drives will be separated from the control element so that the rods remain in the reactor core when the RPVH is lifted.)

5. Removal of the RPVH. (The RPVH is the cover that is removed to gain access to the reactor internals and fuel.)

6. Removal of the upper plenum assembly. (The upper plenum assembly is a large cylinder in the reactor vessel for guiding the control rods and pressing down on the fuel assemblies to maintain proper alignment.)

7. Removal of the core support structure. (This will be done after defueling (see Sec. 6.4). The core support structure is a large basket-like component in the reactor vessel supporting the fuel elements and directing the entering reactor coolant to the bottom of the reactor vessel.)

The best-case (most favorable) conditions are based upon the possibility that damage is largely limited to the core (Ref. 1, Appendix CI, Sec. 8), with minor damage to the other RPV internal components. (The postaccident survival of most of the 52 core thermocouples provides this indication.) The limited damage would not preclude the use of modified existing tooling and normal* or near-normal methods. Core debris attached to component surfaces could be removed manually by workers using long-handled tools. Debris trapped in component cavities could be removed and

*The word "normal" refers to tooling or methods used in the preaccident mode.

handled manually using long-handled pliers, scoops, etc. Removal methods could be characterized as nondestructive and would include any system, tooling, or other means now available to facilitate removal and mechanical disassembly of components when required for removal.

The removed components can be stored in the fuel transfer canal water or on the operating floor behind a temporary shielding wall.

The worst-case (least favorable) conditions are based upon the belief that during the morning of March 28, 1979 (Ref. 1, Appendix CI, Secs. 11 and 15 and Appendix T11, Sec. 2.1 summary). the internal RPV environment at that time may have been drying out, experiencing high temperatures, and being thermally shocked by water quenching. Some of the core temperature measurements (Ref. 1, Fig. CI-12) ranged up to the order of 2000°F to 2500°F. If RPV internal components reached those temperatures, partial local melting and sagging of the components could have resulted. Water quenching at these temperatures also could have caused warping and deformation of the components. Thus, it is assumed that: (1) some components to be removed could have incurred accident damage that precludes the use of existing tooling and normal or near-normal methods; (2) component removal may not be possible until core debris attached to the component surfaces or trapped in component cavities has been removed; and (3) the component removal may be difficult because of deformation or warpage or because of self-welding to adjoining components. Removal methods used could be characterized as destructive and would include engineered equipment, such as: special tooling to cut the component into fragments in order to remove trapped core debris or accomplish removal; special tooling to machine out items, such as studs, that have become fixed; and special tooling to force warped or deformed components into a configuration that could be removed.

Any parts of cut up components would be removed from the RPV and temporarily stored in the fuel transfer canal water until the fragments could be transported out of the canal in containers.

The handling of core debris from the RPV in either the best or worst case requires special control methods and separate temporary storage in the canal water until the debris can be transported out of the canal in containers. The staff has determined that criticality control should be maintained by limiting the container size and shape.* Contamination control can be maintained by designing the container to allow free circulation of water through ports covered by fine screens, if necessary.

6.3.3 Details of Alternative Methods and Facilities

6.3.3.1 Best-Case and Worst-Case Conditions for Steps in the Removal of the RPVH and Internals

The various operations required to remove the RPVH and internal RPV components are listed in Table 6.3. A discussion of each of these operations follows.

1. Preparation of Equipment

The preparations required are essentially the same for the best and worst case conditions. This effort involves installing working platforms, underwater lights; and temporary radiation shielding. Additionally, there will be some preliminary decontamination of the external surfaces of the reactor head service structure, disconnection of electrical cables and cooling water lines, and check out of APSR drive operation.

These operations may generate contaminated waste, such as RPVH insulation, rod drive cabling, and minor amounts of debris. The radiation environment is expected to be variable, ranging from a low of 2 to 3 mR/hr to hot spot levels of perhaps about 2 R/hr (see Appendix I). Work near these hot spots would consist of installation of shielding or possibly operations that would require only a few minutes of worker exposure time. The staff has used an average dose rate of 10 mrem/hr for these operations (see Appendix I).

2. Installation of Seal Plate

The details for this step are common for both the best and worst conditions.

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*Calculations of the minimum critical size for an optimum array of UO_2 fuel in nonborated water give a value of 1.8 ft^3 for a spherical shape. A non-spherical shape with a larger volume would also be safe. These calculations are described in Reference 3.

Table 6.3. Work Effort, Crew Size, and Radiation Levels for Removal of RPVH and Internals

Operation	Normal Work Effort (person-hours)	Estimated Work Effort[a] (person-hours) Best Case	Worst Case	Estimated Crew Size	Radiation Field[b]
1. Preparation of equipment	150	6,000	6,000	15	<10 mR/hr[c]
2. Installation of seal plate	40	400	400	5	<20 mR/hr[d]
3. Removal of RPVH studs & nuts	100	1,200	2,400	5	<100 mR/hr
4. Uncoupling of APSR & CR drives	80	2,000	5,000	4	<20 mR/hr
5. RPVH lift	50	900	1,200	5	10-100 mR/hr
6. Removal of upper plenum	100	2,000	13,000	6	<10 mR/hr
7. Removal of core support structure	100	2,500	17,000	6	<10 mR/hr
Total	600-700	15,000	45,000	(average of 6)	~10 mR/hr

[a]Estimated work effort in a radiation environment.

[b]See Appendix I.

[c]Radiation environment is expected to be variable, ranging from a low of 2 to 3 mR/hr to hot spot levels of perhaps about 2 R/hr. Work near these hot spots would consist of installation of shielding or possibly operations that would require only a few minutes of worker exposure time.

[d]With installation of shielding.

Normally the seal plate is used during short-term refueling operations. For the proposed long-term postaccident defueling operations, the seal should be upgraded. A seal-welded diaphragm or improved sealant would be examples of the type of improvement anticipated. Although initially little waste should be produced by this activity, eventually the seal plate diaphragm and sealing material will require disposal. The worker radiation environment is unknown at this time, but the installation of shielding should reduce the field to 20 mR/hr or less.

3. Removal of RPV Head Closure Nuts and Studs

Under normal conditions this step involves loosening the closure nuts from the studs after cleaning corrosion products from the threads and lubricating the threads by applying a tensioning force to the stud and loosening and removing the nuts. Under worst case conditions, however, loosening may require splitting or stripping jammed nuts and cutting off and machining out some of the difficult studs. As shown in Table 6.3, there is considerable variation in the labor hours for this step to allow for these potential difficulties.

4. Uncoupling of APSR & CR Drives

In order to remove the RPV head it is necessary to uncouple the control rods from the drives, mounted on the head, so the rods will stay in the core when the head is lifted.

Under the best circumstances the control rod drives can be disconnected from the control rods using normal or near normal procedures. Under worst case conditions, however, the control rods may be stuck in the reactor core or plenum grid, thus complicating their disconnection from the control rod drives. This may require cutting the drive housings and/or lead screws that attach to the control rods. As shown by the data in Table 6.3 there is considerable variation allowed in the labor hours for this step to allow for these contingencies.

5. RPVH Lift

Prior to lifting the RPVH, a visual inspection of the core will be made to establish as far as possible the extent of internal core damage, such as hybriding, and physical interference of components. An underwater television camera inserted through a control rod port could possibly be used for this inspection.

Under the best-case conditions, it is assumed that the APSR/CR drives have not been removed and the RPVH with the drives attached can be lifted using the normal head lifting fixture and polar crane. The fuel transfer canal is filled with water and the inside of the head is decontaminated by flushing. Then the RPVH is placed on a storage stand behind a temporary shielding wall on the operating floor.

Under the worst-case conditions the APSR/CR drives have been removed by cutting them off, so dummy control rod followers must be installed to make sure the control rods are not lifted with the RPVH. It is assumed that, to lift the head, a force exceeding a predetermined limit above the weight of the RPVH would be required, so precision jacking equipment is installed to replace the rigging and the head is jacked up until separated from the RPV. The head is then lifted using the same procedure described above.

As shown in Table 6.3, a range of from 900 to 1200 person-hours has been allowed for this operation, which is considerably above the normal requirement of 50 person-hours. Waste produced from this activity for both cases could consist of minor amounts of fuel and clad debris. This material would be collected under water in specially designed filters.

Worker radiation fields would be variable and range from a few mR/hr for underwater operations to perhaps 100 mR/hr.

6. Removal of Upper Plenum Assembly

Under the best-case conditions, the plenum assembly can be lifted using the internals-handling fixture and polar crane. Prior to being lifted, it may have to be jacked up until separated from the RPV guide keys. Before the lift is attempted, a TV camera and a periscope may be installed to visually inspect plenum surfaces and cavities for core debris or damage. Core debris can be removed by using water suction vacuum equipment, grapple, tongs, and water flushing techniques. Dummy control rod followers can also be installed to hold down the APSR/CR assemblies during plenum removal.

Under the worst-case conditions, the upper plenum assembly could be stuck in place, making it necessary to cut it out in pieces, decontaminate the pieces, and transport them out of the containment building in shielded containers. Debris would be placed in specially sized containers that can be handled by the fuel transfer equipment.

As shown in Table 6.3, there is a range of 2000-13,000 person-hours allowed for these contingencies. This compares with about 100 person-hours normally required for this operation. For worst-case conditions, the entire upper plenum assembly, about 110,000 pounds of stainless steel, is considered waste. It would be cut up, perhaps into as many as 150 pieces, and transported out of the reactor building in either shielded containers or through the fuel transfer ports. Worker radiation fields should average less than 10 mR/hr.

7. Removal of Core Support Structure

After the fuel has been removed from the reactor vessel, removal of the core support structure is accomplished the same as outlined above for the removal of the upper plenum assembly, i.e. the core support assembly is lifted using the internals lifting fixture and polar crane in the best case, while it is cut up into pieces for removal under the worst-case conditions.

As shown in Table 6.3, removal of the core support structure could be the most labor-consumptive step in dissassembling the reactor. Because of uncertainties a range of 2500 to 17,000 person-hours has been allowed, compared with 100 required under normal conditions. Under worst-case conditions, the core support structure, consisting of about 225,000 pounds of stainless steel, is considered waste and will be cut into perhaps about 200 pieces under water and transported out of the reactor building in either shielded containers or through the fuel transfer ports.

6.3.3.2 Logistics, Crew Size, Work Effort, Radiation Levels, Waste Generated, and Associated Facilities

Crew size, work effort (person-hours), and radiation fields are summarized in Table 6.3 and discussed below for the two cases.

Crew Size

The crew size will vary depending upon the individual activity and the conditions encountered. Some of the activities, such as preparation, can effectively use large crews--as many as 10 to 15 workers--since many of the preparation tasks can be accomplished simultaneously. Other activities, such as inspections, may require only two to three workers. After assessing the activity and the work that would be accomplished,the staff judged that an average crew size of six workers would be used. The difference between best- and worst-case conditions, therefore, would not be the size of the crew, but the time and work effort required to accomplish a given task.

Work Effort

Person-hours of work effort have been estimated by breaking each case down into major work activities, then estimating the crew size needed to perform the work and the duration of the work (see Table 6.3). Extensive training will have been performed before work in the reactor building actually begins.

The staff assumes that operations associated with the reactor coolant system, i.e., RPVH and internals removal, will be conducted under conditions similar to a normal refueling. Workers would typically be suited up in a double set of coveralls with boots and gloves. Some operations might require respirators for protection against airborne particulates, such as wire-brushing the RPVH stud threads or cutting the control rod drive lead screws. Based upon these considerations, it is estimated by the staff that 15,000 and 45,000 person-hours may be needed for best and worst cases, respectively.

For the best case, about 40 percent of the effort would be for preparation. For worst-case conditions, removal of the upper plenum and core support structure would account for about 80 percent of the total work effort, not including defueling. This compares with the normal effort of 600-700 person-hours for these steps.

Radiation Fields

It has been assumed by the staff that most of the reactor building decontamination will have been completed before any major work is started on the RPVH. Reactor building decontamination criteria, as discussed in Section 5, will reduce background radiation to low levels, and the major radiation source will be from within the reactor coolant system and from the free water surface of the fuel transfer canal. Hot spots also may be encountered in such areas as around the control rod stators and RPVH closure studs. Calculations made to estimate the levels of radiation that may be associated with these hot spots (see Appendix I) indicate that shielding can be provided to reduce the gamma radiation to acceptable levels; thus, installation of such shielding is included in the estimate of work effort.

Two factors will have a major bearing on the level of radiation that will be encountered: (1) the total core fission product inventory is quite low because the reactor had operated at significant power levels for only a few months before the accident, and (2) by the time these removal operations are conducted, perhaps three to five years after the accident, the reactor fission products will have decayed far below the radiation level encountered during a normal refueling operation.

Taking into consideration that reactor building decontamination of the 347-ft level that will have preceded these operations and the fact that shielding can be provided around hot spots, the staff has used an average gamma dose rate of 10 mrem/hr to workers for determining exposure during work associated with removal of the RPVH and internal components (Appendix I).

Waste Generated

The contaminated waste materials produced from the activities are as shown in Table 6.4.

Table 6.4. Wastes Generated during Removal of the RPV Head and Internals

Operation	Waste Generated[a]
1. Preparation	RPVH insulation, rod drive cabling, minor amounts of debris.
2. Installation of seal plate	Seal plate diaphram, sealing material.
3. Removal of RPVH studs and nuts	
Best case	None.
Worst case	Stud and nut fragments.
4. Uncoupling of APSR and CR drives	
Best Case	APSR stators, drive lead screws.
Worst Case	All control drive stators, lead screws, control drive housings, minor amounts of chips and seal material.
5. RPHV lift	Minor amounts of fuel and cladding debris.
6. Removal of upper plenum	
Best case	Fuel and cladding debris.
Worst case	Fuel and cladding debris plus the entire stainless-steel upper plenum assembly weighing 110,000 pounds.
7. Removal of core support	
Best case	Small portion of total core inventory of fuel and cladding.
Worst case	Part of the fuel and cladding plus the stainless steel core support structure.

[a]Some of the wastes may be recoverable after decontamination.

Associated Facilities

Alternatives discussed in Section 5.2 will require support facilities that also can be used for the removal of the RPVH and internals. The only new facility that might be needed would be a water-filled pool for special tooling development, checkout and operator training.

6.3.4 Effluents and Releases to the Environment

6.3.4.1 Normal Releases

Before the RPVH is removed, attachments will be made at various points on the head to remove all pockets of radioactive gas that may be trapped inside the RPV. The vessel will be purged to sweep out these gas pockets, and the gas will be stored or vented to the environment as deemed advisable. It is estimated that a maximum of 100 Ci of Kr-85 gas, essentially the only radioactive gas present, would be released from the RPV by this process. If vented to the environment, the concentration of Kr-85 at the building vent would be 1.77×10^{-3} μCi/mL for about 40 minutes at the building ventilation rate of 50,000 cfm. Such a release is not significantly different from previous low-level Kr-85 releases vented from TMI-2 from time to time.

During certain portions of these operations, the fuel transfer canal and possibly the spent fuel storage pool will be filled. One option is to fill these structures with the decontaminated water from processing of the liquids in the AFHB tanks, the reactor building sump, and the reactor coolant system. If this water is used, it would contain a tritium concentration of about 0.5 μCi/mL. Evaporation from the surface of these storage pools thus would contain tritium which would pass which would pass through the filters in the ventilation systems. For the reactor building, with a ventilation flow of 50,000 cfm, an evaporation rate of 10^{-3} pounds per minute per square foot, and a surface area of 1630 ft^2 (transfer canal), the effluent concentration would be 2.6×10^{-7} μCi/mL. In the fuel handling building the ventilation flow is 36,000 cfm and the storage pool area is 2350 ft^2. Using the same evaporation rate (10^{-3} lb/min-ft^2), the effluent concentration of tritium would be 5.3×10^{-7} μCi/mL. Based on these ventilation rates and effluent concentrations of tritium, the total tritium that would be released to the atmosphere from the reactor building and fuel handling building during one year would be 500 Ci.

Because almost all the operations discussed in this section will be conducted under water, few if any new sources of airborne particulate activity are expected. However, depending on the level of decontamination achieved during cleanup of the reactor building, some airborne particulate activity may be generated simply by people and equipment moving around in the building during removal of the head and internals. The resulting effluents, if any, will be much less than during the original building decontamination effort described in Section 5.

6.3.4.2 Accident Scenarios

The main concerns during this operation are drop accidents which could lead to a loss of water or mechanical damage to the core. Of course, careful monitoring and control of boric acid concentration to ensure that the reactor remains shut down as discussed in Section 4.4 is absolutely essential.

The only fission product that might be released and become airborne as a result of mechanical damage to the core would be Kr-85, which is a gas. Although the Zircaloy cladding on many of the fuel elements is expected to have failed, a few of the elements could be intact and still contain trapped Kr-85. The inventory of Kr-85 in a single fuel element is about 1.5 Ci, of which only a few percent is free gas in the gap that could be released by rupturing the cladding.

The other possibility is a bubble of fission gas trapped in a damaged region of the core that is released by dropping something on top of the core. Since the pressure in such a bubble would be about two atmospheres (i.e., under about 30 ft of water) a bubble one liter in volume would contain about 0.1 moles of gas. The composition of fission gas from low-burnup fuel after one year of decay is 99 percent stable isotopes of Kr and Xe and about 1 percent Kr-85. Thus a one-liter bubble might contain up to 10^{-3} moles, or 35 Ci, of Kr-85 if it were pure fission-product gas. Since the volume of the reactor building is about 2 million ft^3, the release of 35 Ci of Kr-85 would result in a concentration of about 6×10^{-4} μCi/mL inside the building. Assuming a ventilation rate of 50,000 cfm, 90 percent of this Kr-85 would be discharged to the atmosphere in about 100 hours.

At this stage of the operation, a drop accident that caused a failure of the seal plate would result in drainage of the transfer canal to the reactor building basement, but would have no significant radiological consequences since the canal water would contain very little radioactivity (0.01 μCi/mL, exclusive of tritium), and there would be no fuel in the transfer canal.

6.3.5 Environmental Impacts

6.3.5.1 Occupational Doses

Removal of the RPVH and internals includes operations to prepare and modify equipment, to remove the reactor components from the reactor vessel, and to remove components that cannot be reused and other waste products from the reactor building.

It is the staff's judgment, as indicated in Appendix I, that the average radiation exposure rate to worker during an entire shift will be about 10 mR/hr. Therefore, since it has been estimated (Table 6.3) that these operations will require 15,000 to 45,000 person-hours of work effort, the

total cumulative occupational radiation dose will range from 150 to 450 person-rem. The occupational dose includes an estimated 4 to 14 person-rem from inhalation of tritiated water vapor that evaporates from the fuel transfer canal during removal of the PPVH and internals. The fuel canal will be filled with processed water having a tritium concentration of about 0.5 μCi/mL, which is expected to produce an average tritium concentration in the atmosphere of about 2×10^{-6} μCi/mL in areas where workers will spend about 80 percent of their time.

In order to maintain the average atmosphere concentration at 2×10^{-6} μCi/mL or below, it will be necessary to arrange barriers and building ventilation patterns so that the fraction of water vapor at work stations that comes from the fuel transfer canal or any other source of tritiated water vapor does not exceed about 20 percent of the saturation concentration at 75°F.

The expected number of additional cancer mortalities in the work force exposed to this cumulative dose of radiation ranges between 0.02 and 0.06. The expected number of additional genetic effects in the offspring of the work force exposed to this cumulative dose of radiation ranges between 0.04 and 0.12.

Assuming use of three 8-hour shifts a day, and crew sizes as indicated in Table 6.5, the operations associated with removal of the RPVH and internals will require a total calendar time of three to ten months. The removal of the RPVH and internals above the core will require about two to six months, excluding removal of the core support structure. After defueling, an additional one to five months will be spent removing the core support structure and handling the radioactive wastes generated.

6.3.5.2 Offsite Doses

The dose estimates presented here for removal of the RPV head and internals are based on the source terms developed in Section 6.3.4.1. The source terms are for removal of the RPV head (100 Ci of Kr-85) and evaporation of water from the reactor building and fuel handling building (500 Ci of H-3 per year). The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3. The total body dose estimate from the 100-Ci Kr-85 release and the tritium releases are listed in Table 6.6. The total-body population dose received by the human population within 50 miles due to these releases was estimated to be 4 person-rem.

The offsite doses due to the particulate releases mentioned in Section 6.3.4.1 are expected to be less than those described in Section 5.2.5.2. The estimates in Section 5.2.5.2 are significantly less than those due to the 500 Ci/year source term of H-3; hence, the values in Table 6.6 encompass the cumulative dose from these activities.

6.3.5.3 Postulated Accident Effects

The type of accidents for which source terms are developed in Section 6.3.4.2 involve leakage by fuel pin failure of Kr-85 from fuel elements and release of a bubble containing Kr-85 gas trapped in the core. The amount of krypton released in these accident scenarios is so small (1.5 Ci per fuel element accident and 35 Ci from a trapped bubble) that there would be negligible offsite doses. Other fission products that might be released would either be in a particulate or water-soluble form and would be entrained in the RCS water for subsequent water processing.

6.3.5.4 Psychological-Socioeconomic Effects

Psychological-socioeconomic effects relative to removal of the RPV head and internals are discussed in Section 6.5.5.4.

6.3.6 Economic Costs

Estimates of the economic costs for the best- and worst-case alternatives are given in Table 6.7. These costs are estimated on a consistent relative basis, and do not necessarily reflect all costs on an absolute basis. Thus, these costs should be considered only as relative costs for the two scenarios (best case vs. worst case). About 85 percent to 90 percent of the labor costs are expected to be incurred in preparation for work inside the containment building; the majority

Table 6.5. Work Effort, Crew Size, and Calendar Time Required to Remove TMI-2 RPV Head and Internals

Operation	Estimated Work Effort (person-hours)		Estimated Crew Size	Estimated Calendar Time[a] To Complete (months)	
	Best Case	Worst Case		Best Case	Worst Case
1. Preparation	6,000	6,000	15	0.6	0.6
2. Installation of seal plate	400	400	5	0.1	0.1
3. Removal of RPVH studs & nuts	1,200	2,400	5	0.3	0.7
4. Uncoupling of APSR & Cr drives	2,000	5,000	4	0.7	1.7
5. RPVH lift	900	1,200	5	0.3[b]	0.3[c]
6. Removal of upper plenum	2,000	13,000	6	0.5	3.0
7. Removal of core support structure	2,500	17,000	6	0.6	3.9
TOTAL	15,000	45,000	(Average of 6)	3.1	10.3

[a]Calendar time is given to nearest tenth of a month.

[b]One-quarter month.

[c]One-third month.

Table 6.6. Estimates of Offsite Doses to the Maximum Exposed Individual due to Tritium and Krypton Released during Removal of RPV Head and Internals

Location	Pathway	Dose (mrem) Total-Body[a]
Nearest garden[b]	Inhalation	3.4×10^{-2}
	Ground Shine	0
	Vegetable Use	1.2×10^{-1}
	Total	1.5×10^{-1}
Nearest milk goat	Inhalation	1.8×10^{-2}
	Ground Shine	0
	Goat Milk Use	1.4×10^{-1}
	Total	1.6×10^{-1}
Nearest cow milk and garden	Inhalation	3.8×10^{-2}
	Ground Shine	0
	Vegetable Use	1.3×10^{-1}
	Cow Milk Use	5.4×10^{-2}
	Total	2.2×10^{-1}

[a]Doses were calculated for GI-tract, bone, liver, kidney, thyroid, lung, and skin. The dose estimates for all organs except bone were the same as that for total-body. The estimates for bone were zero. Doses were calculated for four age groups: adults, teenagers, children, and infants. The largest of each group are listed in this table. Infant doses are listed for the nearest milk goat location and child doses are listed for the other two.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 mile east-northeast, nearest milk goat = 1.02 mile north, and nearest cow and garden = 1.05 mile east.

Table 6.7. Economic Costs for Removal of RPV Head and Internals (thousands of dollars)

Item	Best Case	Worst Case
Labor	2180	4550
Equipment	1120	1750
Facilities	250	250
TOTAL	3550	6550

of this effort consists of training and procedure preparation. The equipment costs consist of such things as special tooling for seal plate installation, control rod drive removal, and cutting tools for plenum removal. The differences between the best and worst cases are not great since much of this tooling must be developed, checked out, and be available before the actual working conditions have been determined, even though some of the tooling may never be used. The facility needs are not great and consist only of a large water tank with a suitable work platform for tooling mock-up tests and operator training. It has been assumed that suitable onsite indoor locations, such as the turbine hall, exist for this facility.

The methods used to make these cost estimates are described in Appendix K.

6.4 CORE EXAMINATION AND DEFUELING

The objective of core examination and defueling operations is to determine the extent of damage to the reactor core and then remove the fuel assemblies and fuel debris.

6.4.1 Status and Specific Considerations

Information regarding the present status of the core is given in Section 6.2.1.

Because the actual condition of the reactor core is unknown, the staff has developed a framework within which reasonable bounding conditions and alternative courses of action can be developed. As stated previously, the spent fuel pool (SFP) and fuel transfer canal (FTC) will be filled with decontaminated water ($\leq$ 0.01 μCi/mL)* at the start of defueling operations (see Sec. 6.3). A suitable decay-heat-removal system will be in operation as described in Section 4.3. Reasonably clear water is needed to conduct detailed core examination and subsequent defueling operations. Thus, reactor water clarity should be maintained throughout the core examination and defueling phase. Fuel assembly debris may be inadvertently rearranged over the top of the core as a result of removal of the upper plenum. Some trapped fission product gas may be released during defueling; thus, monitoring instruments may be needed to detect any such release. Maintaining the reactor in a shutdown condition is necessary throughout defueling. Additional incore nuclear instrumentation sources may be necessary to monitor the effective neutron multiplication of the core. The conditions of the fuel are as given in Section 6.2.1.

Fuel and fuel debris canning (placement in transfer containers) and transfer to the spent fuel canal (SFP) has been based upon the Allied-General Nuclear Services study[4] of September 1980.

6.4.2 Alternative Methods Considered

The staff's consideration of the environmental impacts of core examination and defueling of the reactor vessel is based upon the core conditions described in Section 6.2.1. The differences in estimates of possible core damage and their resulting impacts on recovery operations have led to the development of two different postulated sets of conditions--a best case based on estimates of the most probable condition of the reactor core and a worst case that reflects the impact of more severe damage conditions (see Sec. 6.2).

The reactor vessel defueling sequence described in this section will involve removing only that fuel material actually within the reactor vessel--not material that may be lodged in other locations within the reactor system, such as coolant piping external to the reactor vessel. Fuel and particulate removal from other portions of the reactor coolant system are discussed in Section 6.5.

The general defueling sequence consists of two series of activities that must be sequenced with activities described in Section 6.3 for removal of the RPV head and internals. The first phase of the reactor defueling will begin with the conclusion of operations 1 to 6 described in Section 6.3 that provide access to the top surface of the core. In this first phase of defueling removal of all the fuel assemblies is accomplished. At that point the defueling activities will be stopped until the lower grid and core support assembly are removed in those activities described in Section 6.3.2, step 7.

The second and final phase of the defueling of the reactor vessel (to scavenge and hydraulically vacuum the remaining fuel and cladding material from the bottom area of the reactor vessel) then will be initiated.

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*Exclusive of tritium.

Television cameras and periscopes (for visual inspections) and underwater lighting and positioning equipment will be needed for defueling operations.

6.4.3 Details of Methods and Facilities

Preparations such as design, construction, and demonstration of special tooling may take about two years. Selection of procedures for some tasks may even have to wait until defueling activities are started. Thus, it may be necessary to plan, design, and fabricate special equipment before a final decision is made on what procedures to use.

Various core-evaluation activities will be necessary to aid in making operational decisions at several points in the defueling process. The removal of the upper plenum assembly from the reactor vessel will permit the use of remotely operated equipment and underwater optical viewing apparatus. A full examination of the core top surface will be required (see Fig. 6.2 for a cross section of core). It would involve an attempt to assess the extent of damage to the central portion of the core, and to the peripheral fuel assemblies. The peripheral fuel assemblies would be checked to determine if they can be removed as entire assemblies with the use of special handling and removal apparatus. As described later, the ability to remove at least one fuel assembly intact from the core periphery will significantly affect the activities.

Examinations of the condition of the internal guide tube openings through the axial length of the fuel assemblies also may be performed to determine whether these passages are sufficiently open to allow for insertion of lifting apparatus into the fuel assemblies. Some of these guide tubes may be structurally deformed or contain debris that cannot be removed. However, it is expected that some guide tube openings will remain clear, especially in peripheral assemblies.

Under the best-case conditions, some fuel cladding and material are expected to be deposited on the top surfaces of the upper flat surfaces of the fuel assemblies (see Fig. 6.3). Also, structural material is expected to be deposited on the upper surfaces of the fuel assemblies as a result of the removal of the reactor vessel upper internals. In the best-case conditions, the particulate matter is assumed to be fuel and cladding material that can be hydraulically vacuumed from the various surfaces. The worst-case conditions would necessitate the removal of additional, larger pieces of structural material and fuel sections, possibly including control and axial power shaping rods as well as parts of fuel assemblies that fused to the bottom of the upper plenum. These larger pieces would have to be knocked or scraped off prior to completing the removal of the plenum structure. These pieces may fall onto the top surface of the fuel assemblies and must be removed before the core examination can be completed. Removal of loose fuel material during the core examination phase is considered an important preparatory requirement proceding the actual defueling.

6.4.3.2 Best-Case Conditions

Under even the best-case conditions anticipated by the staff, significant damage exists within the central portion of the core, with many of the central fuel assemblies bound or fused together at the spacer grid elevations (see Fig. 6.3). Therefore these fuel assemblies cannot be individually removed. However, the staff assumes that under best-case conditions the peripheral fuel assemblies have not been damaged to an extent that prevents extraction of at least one complete fuel assembly using specially designed removal apparatus. The normal fuel handling machines lift the fuel assemblies by the top end fitting. Because successful fuel assembly removal depends upon a structurally sound fuel assembly throughout its length, it is likely that other fuel removal equipment will be needed. The cavity created by removal of one peripheral fuel assembly would permit a sequential extraction of adjacent fuel assemblies radially toward the center of the core by use of the same equipment. At some point on the radial removal path around the core perimeter, the sequential removal activity would reach those fuel assemblies that have sustained the most damage. The removal of these fuel assemblies will require considerably more effort and more complex removal equipment.

Any remaining fuel debris and structural material would be removed after the core support structure was removed using articulated grapple equipment and hydraulic suction apparatus.

6.4.3.3 Worst-Case Conditions

For the worst-case conditions the staff acknowledges that core damage could be so extensive that all of the fuel assemblies have been at least partially damaged and that none of the peripheral

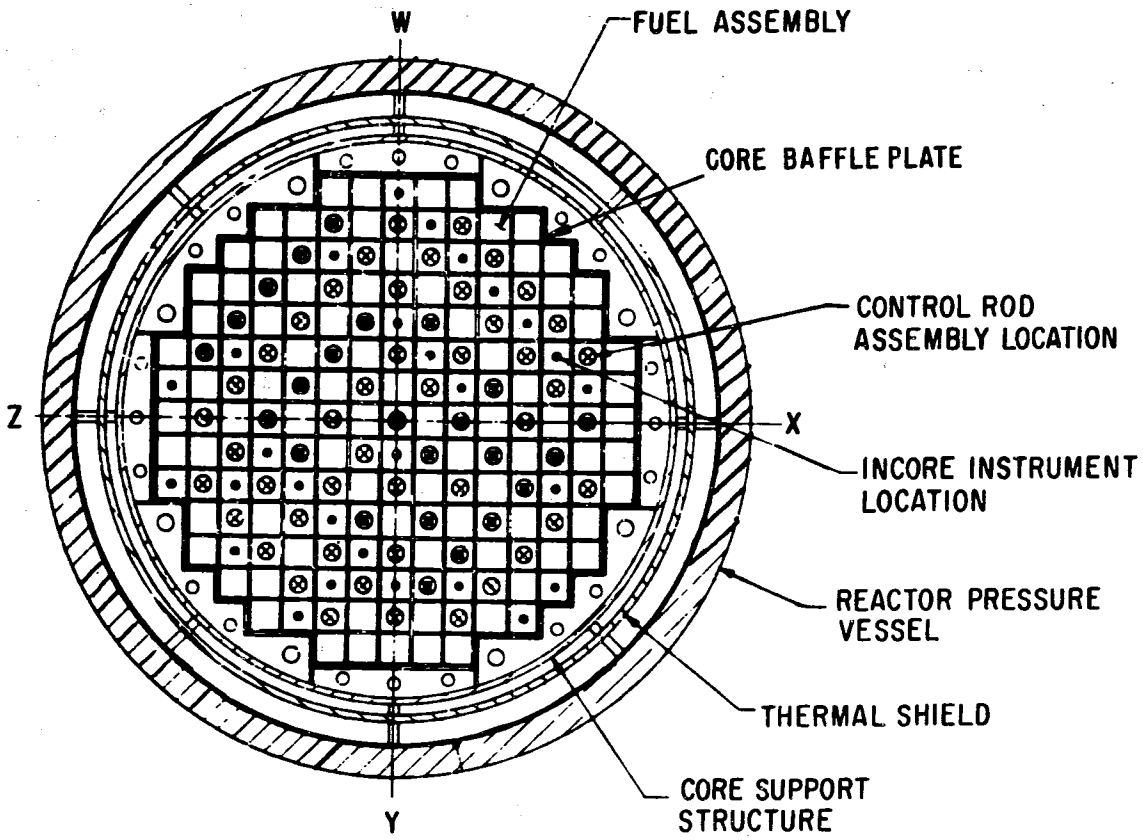

Figure 6.2. Reactor Vessel Internals and Core Cross Section.

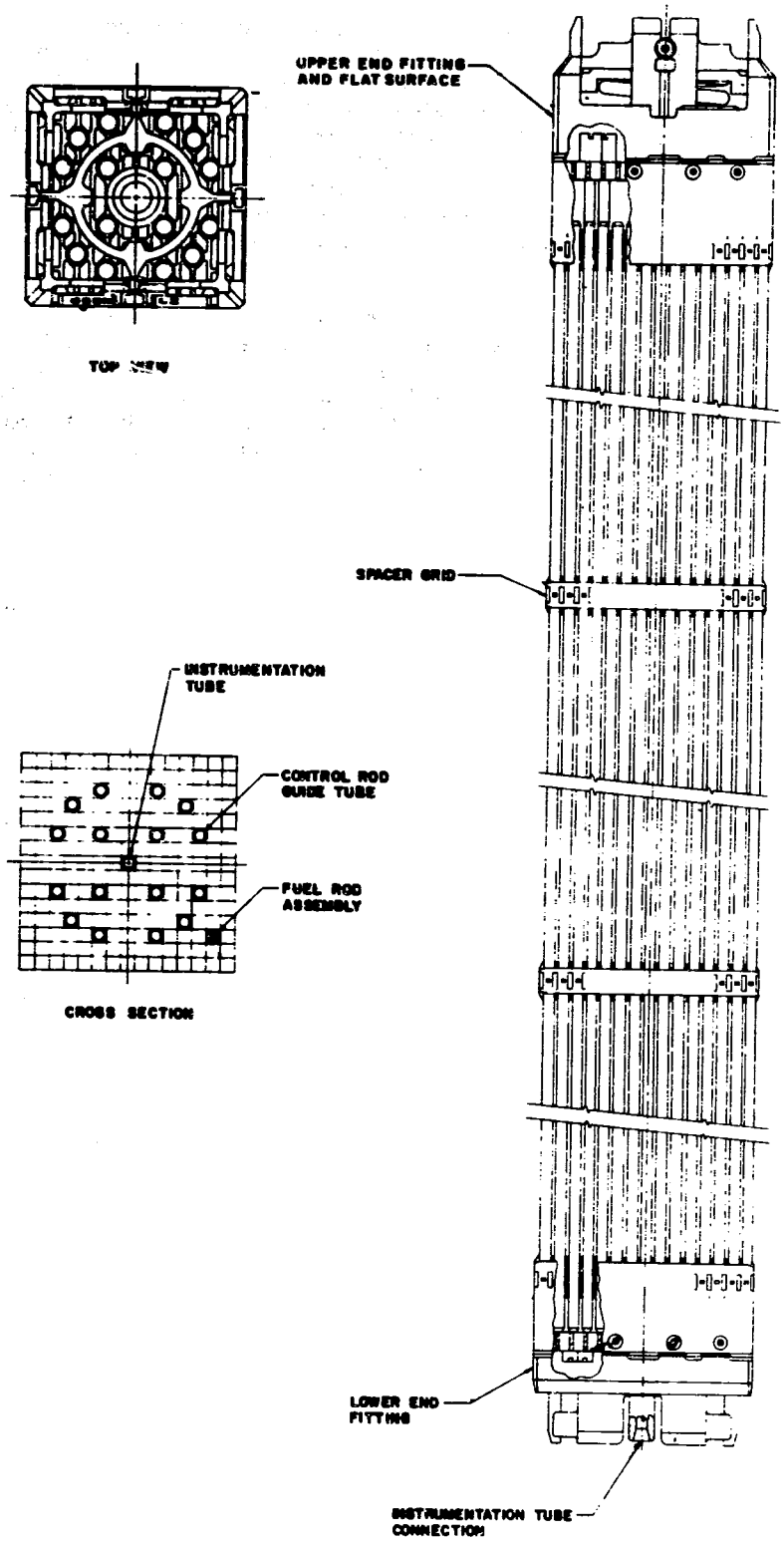

Figure 6.3. TMI Unit 2 Reactor Fuel Assembly.

assemblies can be removed as a complete fuel assembly. As under the best-case conditions, the fuel assemblies in the upper central portion of the core are bound or fused and cannot be removed without separation into smaller segments. Under the worst-case conditions, additional procedures and special equipment would be required to open a full-length cavity on the periphery of the core in order to begin the removal of the first fuel assembly. This initial cavity would be formed by cutting and removing the baffle plates in a segment of the core support structure that provides the most access to the selected peripheral fuel assembly. The removal of adjacent fuel assemblies could then progress by use of the special separation and packaging equipment. Removal of the remaining fuel assemblies would progress radially inward from this location, as under best-case conditions.

Operations under worst-case conditions would require the use of specially adapted equipment. Additional tools, such as hydraulic jacks, may be required to aid in releasing the fuel assemblies from their pockets in the lower grid plate supporting the core assembly. The necessity for use of various items of removal equipment will depend on the actual condition of the core and the procedures chosen. Among the methods reviewed by the staff was that of using cutting tools to cut and shear segments of the core. Since this method would have impacts that lie between the bounding conditions discussed above, this and other techniques are not discussed further.

6.4.3.4 Fuel Debris Handling

The TMI-2 core, as loaded, contained 177 fuel assemblies. It is uncertain what the condition of the fuel will be when removed from the reactor vessel; the staff assumes that the fuel will be in any one or a combination of the following three configurations:

- Intact--intact, but weakened.
- Fused Sections--portions of fuel assemblies fused to each other such that they will have to be physically separated.
- Core Debris--consists of two types: relatively large pieces that can be mechanically handled, and smaller pieces that will have to be hydraulically vacuumed and filtered.

In each of the above configurations, it is likely that the fuel will require canning to provide containment and structural integrity and to prevent spread of contamination during the steps involved in transfer from the reactor building to the spent fuel pool.

Transfer of the fuel from the reactor building to the SFP is via transfer carriage/upender and the fuel transfer tube with the SFP as the end-point.

Interim wet storage in the spent fuel storage pool pending packaging and offsite disposal is anticipated. Dry vault or caisson storage in a new onsite facility is another alternative for interim storage. These options are described in more detail in Section 9.

The procedures and equipment used to handle and package the fuel will be intended to ensure that (1) site personnel exposure is maintained at ALARA levels and (2) the probability of a handling accident is minimized.

6.4.3.5 Final Cleanup

Final cleanup of the reactor vessel would be the same under both sets of conditions and would begin after removal of the core support structure. Any remaining large debris would be removed with remote grappling equipment and placed in transfer containers. All fuel pellets, pieces of cladding, and other small debris would be removed from the bottom surface of the reactor vessel by use of the hydraulic suction equipment and scoop-type pickup apparatus. All material would be placed into transfer containers and moved to a temporary storage location.

6.4.3.6 Special Tools and Equipment

Some of the special remotely operated removal tools and equipment that may be needed will require significant lead times for procurement or for design and fabrication. Design and fabrication effort also will be required to provide mockup equipment for training of those individuals involved

in the fuel removal operations. The equipment development and checkout time may differ significantly between best and worst cases, with the equipment for the worst case being more sophisticated and thus requiring more development and operator-training time. Sufficient lead time must be included to permit fabrication and checkout of this equipment prior to the start of the training.

6.4.3.7 Logistics, Crew Size, Work Effort, Radiation Levels, Waste Generated, and Associated Facilities

Whatever method of defueling is eventually selected, the releases and impacts from that method can be expected to fall within the values given in Sections 6.4.4 and 6.4.5. Crew size, work effort (person-hours), radiation fields, waste generated, and associated facilities are discussed below for the two cases.

Crew Size

Core examination and fuel removal were judged by the staff to require specially skilled personnel for certain tasks. The work crew involved in the best-case activities is expected to average ten people per shift on a six-shift-per-day basis. The worst-case situation may require up to 14 people per shift. Once initiated, each activity of the defueling operation will be a continuous effort until completed.

Work Effort

The total direct person-hours inside the reactor building for the best and worst cases are estimated to be 58,000 and 135,000 person-hours, respectively. Under nonaccident, optimum conditions, an undamaged core can be loaded or unloaded in about 700 person-hours.

Radiation Fields

The staff has estimated an average gamma radiation field of 10 mR/hr for work associated with the defueling activities (see Appendix I). This is based on the assumption that necessary defueling work areas in the reactor building will be decontaminated before defueling begins. Background radiation from building contamination will then be quite low and the major radiation source will be from the water surface of the fuel transfer canal. Hot spots may be encountered during changeout, repair, and modification of equipment. These hot spots can be handled effectively by shielding and by careful design of equipment.

Waste Generated

The waste materials expected to be produced from the two sets of conditions are listed in Section 8.

Associated Facilities

Alternatives previously discussed in Section 5.2 will require supporting facilities that also can be used for defueling. The only new facility that might be needed would be a water-filled pool for development, checkout, and training for any special equipment.

6.4.4 Effluents and Releases to the Environment

The defueling and inspection operations will be conducted through the transfer canal and spent fuel pool water. The only releases expected during normal operations would escape of trapped fission gas bubbles as pieces of the damaged core are pried apart and the evaporation of tritiated water in the transfer canal and spent fuel pools. Soluble and particulate material released by these operations will be picked up by the transfer canal water and removed by the processing systems as discussed in Section 7.

6.4.4.1 Normal Releases

The concentration of Kr-85 in the reactor building which might result from the release of 35 Ci of fission gas was discussed in Section 6.3.4.2. Any airborne particulate activity in the effluent from the reactor building during the defueling and inspection operations is expected to be much less (less than 20 percent) than during the initial decontamination of the reactor building

(see Sec. 5.2). Tritium releases from evaporation of the transfer canal and spent fuel storage pool water are discussed in Section 6.3.

6.4.4.2 Accident Scenarios

During the defueling activities, the main accident concern is related to loss of water from the transfer canal. This could occur if the canal seal plate were damaged (e.g., with the vessel head). Failure of the seal plate would cause a loss of water in the portion of the transfer canal above the reactor vessel and in the adjacent shallow region. Water would not be lost from the deep part next to the transfer tubes, the spent fuel storage pool, or the reactor vessel. An accident that causes failure of the seal plate and the subsequent loss of water from the fuel transfer canal could involve the overheating of a subassembly if one were in the shallow part of the fuel transfer canal at the time of the accident. Although it is likely that a sizable leak in the seal plate would be immediately detected, the rate of water loss could be so rapid that a fuel transfer could not be completed before significant loss of shielding occurred. In this case, an uncooled fuel assembly could possibly be situated in the canal for an extended period of time.

Fuel would not be present in the shallow region except during a transfer or while being placed in a container (can) for a transfer. For the purpose of this accident analysis, a fuel assembly is assumed to be present in the shallow region when water is lost. At the time of defueling, the decay heat generation rate in a single fuel assembly will be on the order of 500 watts or less; hence, not much cooling is needed. However, depending on the time it remained uncovered and on the geometry, the fuel might heat up and release some fraction of the gaseous fission products (Kr-85) trapped in the fuel matrix or in the fuel-clad gap. It is estimated that a maximum of 10 percent of the gaseous activity would be released in this manner. A fuel assembly could contain a maximum of 320 Ci of Kr-85. The resultant concentration in the reactor building would be comparable to the release discussed in Section 6.3.4.2.

A recriticality associated with the movement of fuel or control rods during the defueling cannot occur as long as the boric acid concentration of the water is maintained above 3000 ppm. The consequences of an accident that causes the water to drain from the core are discussed in Section 10.

6.4.5 Environmental Impacts

6.4.5.1 Occupational Doses

The average dose rate to workers in work areas is expected to be about 10 mrem/hr (Appendix I) for both best-case and worst-case conditions.

Under best-case conditions, completion of inspection and fuel removal will require 58,000 person-hours and result in a cumulative occupational dose of 580 person-rem. Under worst-case conditions, completion of fuel inspection and removal will require 135,000 person-hours and result in a cumulative occupational dose of 1350 person-rem.

The occupational dose includes an estimated 20 to 40 person-rem from inhalation of tritiated water vapor that evaporates from the fuel transfer canal and spent fuel pool during core examination and defueling. The fuel canal and pool will be filled with processed water having a tritium concentration of about 0.5 μCi/mL, which is expected to produce an average tritium concentration in the atmosphere of about 2×10^{-6} μCi/mL in areas were workers will spend about 80 percent of their time.

The staff expects defueling to be performed on a six-shift-per-day basis, with ten-member crews for the best-case conditions and 14-member crews for the worst-case conditions. For the best case, each shift could be eight hours long, and crew members would work in the reactor building for four hours and spend four hours performing related activities such as suiting up and desuiting, personal radioactivity monitoring and decontaminaton, and training. Thus, there could be six overlapping 8-hour shifts over a 24-hour period. Each crew member could be limited by the licensee's administrative controls to no more than 1 rem per quarter; thus, assuming an average dose rate of 10 mrem per hour, each crew member could be limited to 100 hours per quarter in the reactor building. The reasons for the staff's selection of this value (for calculation of manpower needs) rather than limits as high as 3 rem per quarter given in 10 CFR Part 20 are provided in Appendix L. However, individuals may receive doses up to 3 rem per quarter if necessary. For

a dose of 1 rem per quarter, 2.5 times as many crew members would be needed as would be the case if staffing requirements were not controlled by the need to limit radiation doses. If radiation exposure were not a consideration, each crew member could work at least 250 hours per quarter. The daily work force requirement would be a ten-member crew for each of six shifts per day, or 60 crew members. For the quarter, therefore, 2.5 times as many workers would be needed, or 150 crew members.

For the worst-case estimate, six 10-hour shifts a day (with four of those hours per shift spent in the reactor building) would be needed. The extra two hours per shift would be for the extra training and planning that would be needed beyond that required for the best case. If 14-member crews were used under worst-case conditions, 210 crew members would be needed. The exact size of the work force may vary, depending on whether unforeseen difficulties develop and on application of the licensee's administrative quarterly dose check point procedures (see Appendix L). Given these variables, the size of the work force could range from 150 to 250 individuals.

For the best-case conditions, six 10-member crews would spend a total of 24 hours (four hours per shift) each day in the reactor building. This would be 240 person-hours per day. For a calendar quarter of 90 days, the work effort in the reactor building would, therefore, be 21,600 person-hours. Since 58,000 person-hours are expected to be needed, about 2.7 quarters, or 8 months, would elapse before defueling would be completed. For the worst-case estimate, using 14-member crews, 30,240 person-hours would be expended per quarter. The work effort of 135,000 person-hours would thus require about 4.5 quarters, or 13 months. The duration of defueling operations thus is estimated to be 8 to 13 months.

Based on the above values of occupational dose and work force, the expected number of additional cancer mortalities in the work force exposed to this cumulative occupational dose would range from 0.08 for the best case to 0.18 for the worst case. The added probability that the average individual worker would die of cancer varies from 1 in 1400 to 1 in 1900. The expected number of additional genetic effects in the offspring of the work force exposed would range from 0.15 to 0.35.

6.4.5.2 Offsite Doses

The offsite dose estimates to the maximum exposed individual resulting from core examination and defueling operations are based on source terms described in Section 6.4.4.1 (35 Ci of Kr-85 and 20 percent of the source term listed in Table 5.10, and assuming no tritium is released) and are listed in Table 6.8. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3. The total-body population dose received by the human population within a 50-mile radius from these activities was estimated to be 2×10^{-3} person-rem.

6.4.5.3 Postulated Accident Effects

The type of accident postulated in Section 6.4.4.2, which would result in offsite doses, is that of uncovering a fuel assembly and subsequently releasing 32 Ci of Kr-85. The doses offsite resulting from such a release are insignificant, and they are negligible in comparison to those from the routine decontamination operations. Consequently, no numeric values for this accident scenario are presented here.

6.4.5.4 Psychological-Socioeconomic Effects

Psychological-socioeconomic effects relative to core examination and fuel removal are discussed in Section 6.5.5.4.

6.4.6 Economic Costs

To evaluate all the economic costs directly associated with a particular alternative, all labor, facility, and equipment costs need to be included. Estimates of the economic costs for the best- and worst-case alternatives are given in Table 6.9. These costs are estimated on a consistent relative basis and do not necessarily reflect all costs on an absolute basis. These costs should be considered only as relative costs for the two scenarios (best case versus worst case).

Table 6.8. Dose Estimates for the Maximum Exposed Individual from Core Examination and Defueling Operations

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	4.8×10^{-7}	6.3×10^{-6}	8.6×10^{-7}
	Ground Shine	1.9×10^{-6}	1.9×10^{-6}	1.9×10^{-6}
	Vegetable Use	3.7×10^{-5}	1.6×10^{-4}	4.2×10^{-5}
	Total	3.9×10^{-5}	1.7×10^{-4}	4.5×10^{-5}
Nearest milk goat	Inhalation	7.3×10^{-7}	2.5×10^{-6}	5.7×10^{-7}
	Ground Shine	1.8×10^{-6}	1.8×10^{-6}	1.8×10^{-6}
	Goat Milk Use	3.1×10^{-5}	2.2×10^{-4}	2.5×10^{-4}
	Total	3.4×10^{-5}	2.2×10^{-4}	2.5×10^{-4}
Nearest cow milk and garden	Inhalation	9.0×10^{-7}	6.9×10^{-6}	9.5×10^{-7}
	Ground Shine	2.8×10^{-6}	2.8×10^{-6}	2.8×10^{-6}
	Vegetable Use	3.6×10^{-5}	2.4×10^{-4}	6.2×10^{-5}
	Cow Milk Use	1.2×10^{-5}	5.5×10^{-5}	4.8×10^{-5}
	Total	5.2×10^{-5}	3.0×10^{-4}	1.1×10^{-4}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children. The dose estimates for the nearest goat location are for adults for total-body, and for infants are for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual loactions are: Nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 6.9. Economic Costs for Defueling (thousands of dollars)

Item	Best Case	Worst Case
Labor	6,620	9,910
Equipment	5,860	5,860
TOTAL	12,480	15,770

Because of uncertainties as to the condition of the core, plus lead time required for tooling, the staff has assumed that regardless of the core condition, costs for preplanning, tooling, and training will be the same for both the best and worst cases. Only the cost of labor required to remove and support removal of the fuel will be different for the best and worst cases.

Compared to the labor and equipment costs, the facility needs are small. In addition to the water-filled tank mentioned in Section 6.3, a fairly large mockup area will be required to set up the special tooling for operator training. The staff has assumed that suitable mockup locations will be available onsite and has made no allowance for facility costs.

6.5 DECONTAMINATION OF PRIMARY SYSTEM COMPONENTS

Postaccident analyses at TMI suggest that fuel debris and perhaps other particulates were scattered throughout the primary system.[1,2] Fission products are thought to have been carried from the exposed fuel and subsequently plated out on all primary system component inner surfaces. The objective of decontamination of the primary system components is to remove the fuel debris and other particulate matter from the system and reduce the fission product plateout on internal surfaces to a level similar to that of operating reactors. For purposes of this discussion, decontamination of the primary system components is considered as being principally two distinct activities--draining and flushing and chemical decontamination.

6.5.1 Status and Specific Considerations

Before decontamination of the primary system components begins, reactor core defueling and removal of the RPVH and reactor internals will have been completed. The components to be decontaminated are the reactor vessel, the steam generators, pressurizer, fuel transfer canal and reactor drain tank, drain tank transfer pumps, reactor coolant pumps and motors, letdown coolers, and associated piping.

A number of special factors must be taken into consideration in the planning and execution of component decontamination. Attention must be given to flushing and chemically decontaminating the dead legs, low points, drain and vent lines, flushing connections, and injection lines. Chemical decontamination solutions should be compatible with component materials. Spot application of chemicals may be necessary. The inventory of contaminated solutions generated by flushing and decontamination should be kept as low as practical. Acquisition of additional equipment or systems needed to flush and chemically decontaminate the pressurizer because of its location in the system, must be planned.

Decontamination of the reactor drain tank may present special problems. The tank (and its associated piping) is partially submerged in highly contaminated water in the reactor building basement and is believed to contain high levels of radioactivity because of the presence of contaminated reactor cooling water and core debris. The exact condition of the tank is not known, but it is believed to be structurally sound and to contain reactor cooling water. The reactor drain tank transfer pumps and coolers also are submerged in water. Repair of some components may be necessary.

The reactor coolant pumps and motors are not in operation and it is not known if they are operable. It is believed that the pumps are mechanically sound. The pumps were shut down during the accident when they were vibrating because of being vapor bound, rather than because of pump malfunction. Two of the pumps were operated (one at a time) for a total of about one month after the accident to provide forced cooling through the primary loop. The condition of the motors at this time is not known. However, pump motor and cabling damage, both mechanical and electrical, could have occurred as a result of reactor building conditions; e.g., the building pressure increase during the accident and the high humidity of the atmosphere in the building. Pump shafts, shaft seals, and impellers may have been damaged by vibration caused by the pump's being vapor bound. A laydown area for pump and motor dismantling and decontamination may be needed.

The letdown piping and coolers also are submerged in the highly contaminated water in the reactor building basement. Both coolers were online during the accident and, therefore, have been exposed to highly contaminated reactor coolant. It also is expected that these coolers will contain a considerable amount of highly radioactive plateout, estimated to range from 10 to 100 $\mu Ci/cm^2$.

6.5.2 Alternative Methods Considered

The decontamination of the primary system components can be considered principally as two distinct activities--removal of loose particulates and removal of more tightly adhering fission-product plateout. It is likely that particulates can be removed by draining, flushing, and filtration,* but that plateout removal will require use of chemical reagents. Reactor system inspections (Sec. 6.2) will indicate the sequence of operations, but the staff anticipates that particulate removal will be carried out first.

The first activity considered in this section is removal of particulates that may have been scattered throughout the primary system and that cannot be readily removed by use of chemical reagents (Sec. 6.5.2.1). Although the exact quantities, locations, and composition of these particulates are unknown, the staff believes that most of the primary system components are contaminated with particulates consisting primarily of irradiated fuel and oxidized cladding (ZrO_2).

The second activity considered is removal of fission product plateout material adhering to various internal surfaces of primary system piping and components (Sec. 6.5.2.2). Although the actual concentration of plateout materials is unknown, higher concentrations of plateout could be expected in areas of heat transfer than in other parts of the system. Plateout removal using chemical solutions is a well-developed and effective decontamination technique.[5]

6.5.2.1 Particulate Removal Using Drain and Flush Techniques

Reactor Coolant System

Two alternative methods have been considered for flushing the reactor coolant system: (1) using the existing reactor coolant pumps and (2) using system pressurization.

Reactor Coolant Drain Tank

Because of the system configuration, the reactor coolant drain tank is discussed separately from the rest of the system.

Two alternative methods have been considered for the draining of the reactor coolant drain tank. The only difference between them is the method of initial draining. Under the first alternative, the tank contents would be drained to the reactor building sump; under the second, the contents would be drained to the reactor coolant bleed holdup tanks. Installation of drain line filters would be required for either alternative.

Drain tank flushing would be the final step for both of the alternatives. Flushing would consist basically of filling the drain tank with water and draining the water to the reactor coolant bleed holdup tank, followed by manual flushing with hoses and water jets. Flushing also would include the 14-inch-diameter pressurizer relief valve header. Small lines of 2 inches or less in diameter would be flushed by other means, removed and capped, or replaced.

6.5.2.2 Plateout Removal Using Chemical Decontamination Techniques

Major considerations involved in chemical cleaning of the reactor coolant system are (1) the chemical reagents to be used and the method by which they are prepared for use, and (2) the procedures used to conduct decontamination--more specifically, the method used to transport the chemical cleaning agents through the coolant system. The reagent chemicals used for decontamination normally would be shipped in concentrated form and then diluted with water for use. To prepare the solutions, large tanks with mixing capability would be needed. The existing reactor bleed tanks or the borated water storage tank could provide this capability. Contamination remaining after drain/flush activities would probably be in the form of activation and fission product plateout in the corrosion product film and some particulate uranium oxide not removed by flushing. Films on stainless steel and other high nickel alloys are not easily removed and require either corrosive treatments or some conditioning step to change the film characteristics. Satisfactory procedures have been developed for removing such films on surfaces of primary coolant

*Design of particulate filters will require consideration of potential criticality hazards.

systems of nuclear reactors and for decontaminating the surfaces.[5] Six alternative processes have been considered for the TMI-2 cooling system: (1) OPG process should fuel particulate dissolution be necessary, (2) the oxalic-citrate-peroxide process, (3) the CAN-DECON process, (4) the alkaline permanganate-citrox (AP-Citrox) process, (5) the alkaline permanganate-ammonium citrate process (APAC), and (6) the Dow Chemical NS-1 process. Details of each are given in Appendix P.

Circulation of Chemical Solutions through Principal Reactor Coolant System Components

Two alternative methods of circulating the decontamination reagents through the principal reactor coolant system components (Fig. 6.4) have been considered: use of the existing reactor coolant pumps or use of new pumping equipment.

Circulation of Chemical Solutions to the Reactor Coolant Drain Tank

Two alternative methods have been considered for circulation of chemical decontamination solution to remove fission product plateout from the internal surfaces of the reactor coolant drain tank. One method would be to feed and bleed (drain) the chemicals through the tank; the other method would be to fill the tank and then recirculate the contents. For both alternatives, decontamination solutions could be provided from a makeup system during the decontamination process.

Chemical Decontamination of Reactor Coolant Pumps and Motors

The procedure for reactor coolant pump decontamination would depend on the condition of the pumps and pump motors, which could range from insignificant contamination of motor internals and no damage to either the motor or the wetted pump components, to a condition of high levels of contamination and severe damage to motor and wetted pump components.

Two alternatives have been considered by the staff for decontamination of the external surfaces of the four reactor coolant pumps and motors. The first alternative would involve inplace decontamination; the second would involve removal and out-of-containment decontamination, and/or disposal. Inplace decontamination would be carried out in the reactor building, and for out-of-containment decontamination, the pumps would be moved out of the building. Inplace decontamination would be carried out in cases when only minor contamination of the motor internals and no significant damage to the motor and other pump components had occurred. Out-of-containment decontamination would be necessary if there was contamination or severe damage to the motor or pump components. These alternatives do not cover those pump components that are normally exposed to the primary coolant. Those components would be decontaminated with the reactor coolant system as discussed previously.

6.5.3 Details of Methods and Facilities

6.5.3.1 Particle Removal Using Drain and Flush Techniques

The draining and flushing alternatives considered differ only in the equipment used and work effort (person-hours) required to perform a given activity. Processed and demineralized water probably would be used for flushing, as practical. The use of "new" water would be minimized. Tanks and piping used to receive and/or store flush solutions most likely would have been already desludged and decontaminated.

Reactor Coolant System

Flushing Using the Reactor Coolant Pumps. This alternative would involve the use of the reactor coolant pumps, a large, specially built core filter installed in the reactor vessel, and inline filters in reactor coolant system drains.

After the reactor defueling is completed and the fuel transfer canal has been drained, the core filter would be installed in the reactor vessel and the RPVH would be reinstalled. This filter would be designed to permit backflow and to allow flow bypass if the pressure drop across it exceeded a prescribed value. Instrumentation would be included to monitor filter performance. Additional filters would be installed on the reactor coolant system drains to trap particulates during the drain and flush activities.

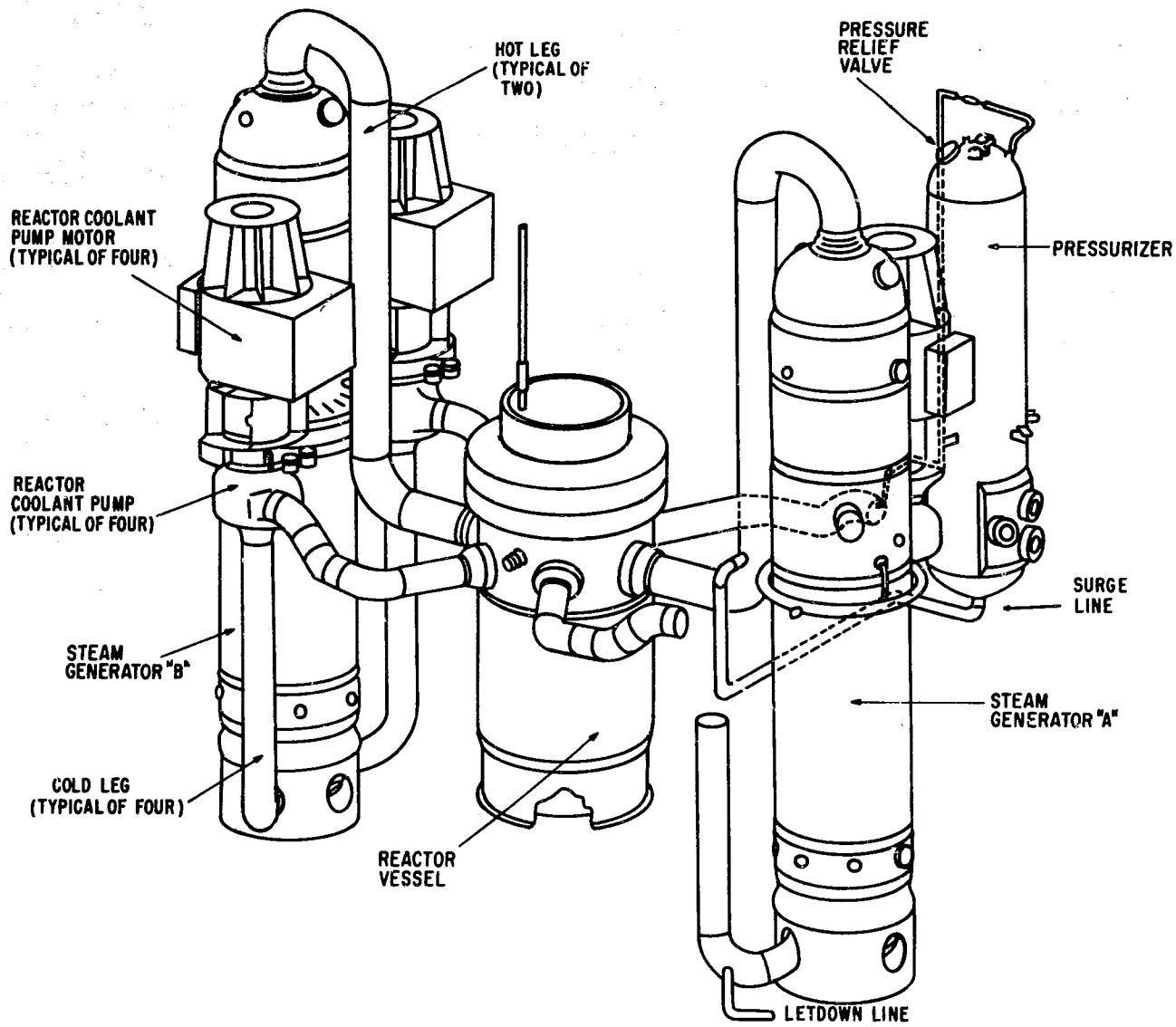

Figure 6.4. Schematic Drawing of Primary System.

To permit use of the four reactor coolant pumps for flushing operations, the reactor system would be filled with water, borated as necessary, and then pressurized so that the coolant pumps could be operated. Water flow through the four pump legs would be used to wash the particulates to the core filter. Drain and sample lines on the various flow loops would be opened and the lines flushed to in-line filters. Particulates collected in the filters would be placed in storage containers.

The water would be circulated throughout the system, with the particulates being captured in a core filter. During circulation, drain valves would be opened to flush particulates to cartridge-type, inline filters.

The advantages of this alternative are that flushing would be largely a remote operation and would not require removal of major components. The disadvantage is that the design, construction, installation, and disposal of the specially built core filter would be costly and time consuming.

For the radiation level estimate for flushing the reactor coolant system, it was assumed that the reactor building and component external surfaces will have been decontaminated to essentially clean conditions and that local hot-spots will have been shielded before the flushing operation. The radiation exposures estimated here apply only to direct effort associated with this flush/drain activity. To take into account the particulates that could exist in the reactor coolant system, the staff has assumed that a core debris volume of 2×10^5 cm^3, equivalent to about two fuel assemblies, is distributed throughout the system, with major collection points at each of the steam generator inlets and outlets and at the pressurizer bottom. These particulates would be collected in several filters during the flush. Particulates captured in the incore filter would be removed and placed in smaller containers for storage. The inline filters containing the accumulated particulates would be stored. Changing of filters would constitute the major source of exposure for the workers.

Flushing Using System Pressure. With this alternative, flushing by use of system pressure would be used. This procedure would not require the use of reactor coolant pumps or support systems. The reactor head (or any other suitable cover) would be reinstalled, and the reactor coolant system then would be filled with water and pressurized with nitrogen in the pressurizer. This would provide the necessary pressure for a high-velocity flush of the drain lines and the sample lines through inline filters to the reactor bleed holdup tanks.

The advantages of this alternative are that the design, construction, installation, and disposal of the special core filter would not be necessary. The disadvantages are that the operation would require more hands-on operations (with greater worker exposure) and, depending on the particulate size and location, flushing might not be as effective as forced circulation using the reactor coolant pumps.

Special equipment might have to be designed for use with either alternative to permit use of a high-pressure water jet to remove remaining particles from the piping. The equipment could be pulled, pushed, or self-propelled through the pipe. This operation would require removal of the primary pumps and component access covers to gain access to certain portions of the piping. Spray water would be used to flush particulates to the nearest low-point drain line available. Any small piping that could not be effectively flushed would be removed and capped.

Reactor Coolant Drain Tank

Flushing and Draining to Reactor Building Sump. The reactor coolant drain tank now contains primary coolant and could contain both particulates and plateout. To implement this alternative, personnel would have to be able to gain access to a drain valve on the bottom of the reactor coolant drain tank that would direct the liquid to the reactor building sump. Preliminary calculations that are based on the soluble fission product concentration in the water and that do not take into account the presence of particulates, plateout, or other nearby sources of radioactivity, indicate that the radiation levels may be low, perhaps on the order of 50 mR/hr at the tank surface. If primary system inspection (Sec. 6.2) indicates that these calculations are accurate, access would not be a problem; however, if particulates and plateout are present, there may be hot spots that could limit personnel access without use of shielding and special tools.

The advantages of this alternative are that the operation would be simple, there would be little impact on schedule, and no refurbishment of equipment would be necessary. The disadvantages are

that local shielding could be required and additional radioactive water would be released to the reactor building sump.

Flushing and Draining to Reactor Coolant Bleed Holdup Tanks. Draining of the reactor coolant drain tank into the reactor coolant bleed holdup tanks would require replacement or repair of the drain tank leakage transfer pumps and possibly of the reactor coolant drain pump. In addition, filtration would be required within the reactor building to remove particulates from the water. Since the pumps are located outside of the drain tank cubicle, radiation exposure from the tank may not be a problem; therefore, pumps could be replaced and the tank drained without special shielding or tooling.

The advantage of this alternative is that no additional radioactive water would be released to the reactor building. The disadvantage is that system modification and equipment replacement or repair would be required.

6.5.3.2 Particulate Removal by Dissolution

It is possible that flushing may not adequately remove fuel debris from the RCS. It is very important to either a rehabilitation program or a decommissioning program that fuel debris be completely removed since each particle is extremely radioactive. Should decommissioning be contemplated, this would adversely effect primary system sectioning and disposal efforts. Should rehabilitation efforts be contemplated, similar problems would occur with required maintenance. Also, a clean (fuel debris free) RCS would be required to prevent recontamination of the RCS at startup. One proven solution for dissolution of fuel element debris is oxalic-peroxide-gluconic (OPG) solution.[6] The corrosion resulting from use of this solution is generally minor for most reactor materials of construction. If an inventory of materials of construction in the primary system revealed potential corrosion problems, appropriate counter measures would be required.

Since the amount of fuel element debris that will be left in the system following defueling and flushing cannot be determined until the RPVH has been removed and the core damage assessed, development of dissolution equipment schemes are counterproductive at this time. If the amount of fuel element debris in the system is large, several system volumes of dissolving solution may be required to assure that the solution does not become too radioactive to handle with existing decontamination solution disposal equipment. Nuclear criticality would be considered in the design of the cleanup equipment. The OPG solution is only partially effective for removal of plateout material and, of course, the RCS film will have aged so no forecast can be made regarding the need for further RCS decontamination until fuel debris removal is complete. In any event, up to this point in the TMI decontamination, there would be no difference in the approach for either decommissioning or rehabilitation.

6.5.3.3 Plateout Removal Using Chemical Decontamination Techniques

The chemical decontamination alternatives considered differ in the type of solutions used, methods of preparation, methods of introducing the chemical solution into the system, and the equipment used. Decontamination process alternatives are detailed in Appendix P.

Reactor Coolant System

Circulation Using Reactor Coolant Pumps. This alternative would involve use of the normal reactor coolant pumps and other inplant equipment to circulate decontamination solutions through the system. The solution would be allowed to flow through the letdown-makeup system and the mini-decay-heat-removal system both for decontamination and cooling. Additional heat may be removed by filling the secondary side of the steam generators with water and maintaining secondary flow. While the decontamination solution was being circulated, drain lines and sample line valves would be opened as necessary and solution allowed to flow through the piping to decontaminate it. The reactor head and internals also could be decontaminated at this time if they were reinstalled in the RPV. At completion of decontamination, the reactor coolant pumps would be stopped and the solution allowed to drain to the reactor bleed tanks. The advantage of this alternative are that no major pieces of new equipment would be needed, and systems for solution mixing and transport already exist. The disadvantage is that existing equipment (pumps) must be operable before decontamination can proceed.

Circulation Using New Pumping Equipment. If new pumping equipment, heat exchangers, and tanks are used to circulate chemical decontamination solution through the reactor coolant system, tanks

mounted on trucks or inplant tanks would be used to mix the chemical solution and then it would be pumped to the reactor coolant system. As an example, the mini-decay-heat-removal system could be used to fill the primary coolant system. The solution could be piped to the intake of the mini-decay-heat-removal system pump. The heat exchanger could be used to heat the solution, or an auxiliary heat exchanger could be installed. The chemical solution would be circulated by sequencing the opening of various drain valves. Piping would be provided from the drain valves back to the intake of the mini-decay-heat-removal system pump. Flow would be established through independent components by opening the drain valves on these components. The spray line to the pressurizer would be connected to the outlet of the pump and then the pressurizer drain valve would be opened to promote circulation. At the completion of circulation, the decontamination solution would be drained from the reactor coolant system into the reactor bleed holdup tanks for further processing.

The advantage of this alternative is operation of reactor coolant pumps and other reactor support systems would not be required. The disadvantages are that procurement and installation of additional equipment would be necessary and that decontamination would require more time because of the lower flow capacity of the new chemical solution pumping system.

Reactor Coolant Drain Tank

Feed and Bleed. For this alternative, decontamination solutions would be introduced into the 14-inch-diameter pressurizer relief valve header near the relief valves to fill the tank and header. Once the tank and header were filled, the solution would be drained to the reactor coolant bleed holdup tank, while at the same time more solution was being fed into the header.

The advantage of this alternative is that fewer system modifications would be required. The disadvantages are that positive circulation would not be ensured, and more liquid wastes would be generated than for the next alternative.

Recirculation. For this alternative, a recirculation line and pump would be installed with a heat exchanger to circulate the decontamination solution from the reactor coolant drain tank to the 14-inch pressurizer relief valve header at a point adjacent to the relief valves and back to the tank via the header. While the solution was circulating, it would be heated to 180-250°F to increase the effectiveness of decontamination. After decontamination was completed, the solution would be drained to the reactor coolant bleed tank for storage and processing.

The advantages of this alternative are ensured circulation and minimum liquid waste. The disadvantage is that additional equipment and modification to existing systems would be needed.

Chemical Decontamination of Reactor Coolant Pumps and Motors

Inplace Decontamination. Inplace decontamination basically would consist of cleaning accessible surfaces within the motor housing; draining and flushing the oil lift system; cleaning, inspecting, and replacing pump seals; and making electrical and mechanical checks. This is the procedure that would be used if the pumps were to be operable (perhaps minor repairs might be needed) and could be used for flushing and decontamination of the reactor coolant system after pump decontamination was completed.

Out-of-Containment Decontamination. Out-of-containment decontamination can be divided into two areas of effort--(1) motor decontamination and repair and (2) pump shaft and impeller decontamination and repair. Since there are four pumps, it may be found that both types of effort or only partial effort is required for a given pump. Under this alternative, the pump motor and pump shaft and impeller assembly would be removed from the reactor building to a laydown area for decontamination and repair.

The pump motors would be decontaminated and repaired or replaced as needed. In cases of severe damage, the motors might have to be shipped offsite for repair (such as for rewinding of the motor). The pump shaft, impeller, seals, bearings, and other wetted components also would be decontaminated, repaired, or replaced as necessary to the extent that the pumps could be used.

6.5.3.4 Logistics, Crew Size, Work Effort, Radiation Levels, Waste Generated, and Associated Facilities

Specific methods for treatment of the various groups of reactor coolant system components will not be selected from among the alternatives until decontamination work is underway. The staff has made estimates of crew sizes, work effort, radiation levels, wastes generated, and special facilities needed for each alternative. These estimates are summarized in Table 6.10 and discussed below.

Crew Size. The work crews required for each activity are shown in Table 6.11.

Work Effort. The estimates of work effort (person-hours) include a nominal estimate, based on good working conditions (low radiation levels, clean atmosphere) and a maximum estimate, based on adverse working conditions (high radiation levels, use of breathing equipment). The estimates for the various activities are shown in Table 6.10. Considerably more effort is required for flushing the reactor coolant system using system pressure than for flushing using the reactor coolant pumps. This is because of the greater number of cartridge-type filter changeouts needed in the former alternative.

The alternative of out-of-containment reactor coolant pump decontamination has several possible subalternatives. The estimated work effort for each subalternative is given in Table 6.12. Equipment movement and handling is a significant portion of the work effort.

Radiation Levels. The average radiation levels estimated for each activity are given in Table 6.10. The activity of draining and flushing reactor coolant drain tanks has somewhat higher radiation levels than for subsequent activities because this work effort could occur during reactor building decontamination. The average radiation levels for draining the reactor coolant drain tanks to the reactor coolant bleed holdup tanks are somewhat lower than draining to the reactor building sump because the pumps for the former alternative are located outside of the drain tank cubicle. Since the chemical decontamination of the reactor coolant system and the reactor coolant drain tanks would be one of the last activities performed, radiation levels would be low.

Waste Generated. The waste expected to be generated by each activity is given in Table 6.13. The flushing water will probably be processed water, minimizing the overall volume of radioactive waste water generated.

Special Facility Needs. A filter backflush facility that would allow particulate removal and subsequent reuse of cartridge-type filters might be needed for the flushing of the reactor coolant system using system pressure alternatives. This would substantially reduce the total number of filters needed. The facility would be a well-shielded room with a high-pressure water supply and suitably shielded equipment for containing the backflushed material. No other special facilities have been identified. The out-of-containment reactor coolant pump decontamination alternative needs those facilities used for reactor building decontamination, such as the containment service building.

6.5.4 Effluents and Releases to the Environment

6.5.4.1 Normal Releases

The decontamination operations will generate solid and liquid wastes. The solid wastes resulting from these operations are described in Section 8. The solutions resulting from decontamination operations are of two types--relatively clear water from simple flushing operations and chemical reagents containing substantial solids. The treatment of these wastes are described and resulting releases discussed in Section 7. The staff expects decontaminating reagents to be made up from processed water; thus the inventories of liquid in the system will not be increased.

Although there will be fission products present in the decontamination solutions, the potential airborne particulates will be negligible; however, tritium will be released by the evaporation of water from the spent fuel pool. These releases will be less than those listed for the defueling operation described in Section 6.3.4.1 (< 300 Ci/year).

Table 6.10 Work Effort, Crew Size, and Radiation Field for Activities in Decontamination of Reactor Primary System

Activity	Work Effort[a] (person-hours)		Crew Size	Radiation Field (mR/hr)	
	Nominal[b]	Maximum[c]		Nominal	Maximum
Flushing Reactor Coolant System					
Using reactor coolant pumps	8,200	13,600	8	10	30
Using system pressure	34,000	40,000	8	10	30
Draining and Flushing Reactor Coolant Drain Tanks					
Drain to reactor building sump	160	460	3	30	100
Drain to reactor coolant bleed holdup tanks	500	1,500	3	20	50
Chemical Decontamination of Reactor Coolant System					
Using reactor coolant pumps	5,000	9,000	10	3	10
Using new pumps	7,000	11,000	10	3	10
Chemical Decontamination of Reactor Coolant Drain Tanks					
By feed and bleed	1,100	2,600	8	3	10
By recirculation	2,700	6,600	8	3	10
Reactor Coolant Pump Decontamination					
In-place	1,200	2,900	3	10	30
Out-of-containment	1,300-4,500[d]	2,600-10,600[d]	6	20	50

[a]Direct work in a radiation environment.

[b]Good working conditions - low radiation levels, clean atmosphere.

[c]Adverse working conditions - high radiation levels, use of breathing equipment.

[d]See Table 6.12.

Table 6.11. Work Crews Required for Activities in Primary System Decontamination

Activity	Crew Required
Flushing reactor coolant system	Five persons for handling and installation of equipment (two technicians, a supervisor, one health physics technician, craftsman), four persons for operation (two technicians, a supervisor, one health physics technician).
Draining and flushing reactor coolant drain tanks	Two technicians or craftsmen, one health physics technician.
Chemical decontamination of reactor coolant system	
Handling and installation of equipment	Two technicians, one supervisor, and one health physics technician, and two craftsmen.
Operation	Two technicians, one supervisor, and one health physics technician.
Chemical decontamination of reactor coolant drain tanks	
Handling and installation of equipment	Three craftsmen, one technician, and one health physics technician.
Operation	Two technicians, and one health physics technician.
Reactor coolant pump decontamination	
Inplace	Two technicians and one health physics technician (multiple crews).
Out of containment	One supervisor three craftsmen, one technician, and one health physics technician.

Table 6.12. Work Effort Estimates for Subalternatives for Out-of-Containment Decontamination of Reactor Coolant Pump Components

	Work Effort (person-hours)	
Subalternative	Nominal	Maximum
Onsite motor repair (motor rewound in containment service building)	4,500	10,600
Offsite motor repair (motor rewound offsite after decontamination)	4,500	9,000
Motor replacement	1,300	2,600
Pump shaft and impeller repair (onsite in containment service building)	2,800	6,000

Table 6.13. Wastes Generated from Activities in Primary System Decontamination

Activity	Waste
Flushing reactor coolant system	
Using reactor coolant pumps	Two large core filters, one core filter support structure, about 20 cartridge-type inline filters. Miscellaneous tools and equipment, contaminated clothing, contaminated cleanup items (rags, brushes, etc.), and about 235,000 gallons of liquid.
Using system pressure	200 cartridge-type, inline filters, miscellaneous tools and equipment, contaminated clothing, contaminated cleanup items (rags, brushes, etc.), and about 235,000 gallons of liquid.
Draining and flushing reactor coolant drain tanks	Six cartridge-type inline filters, miscellaneous tools and equipment, contaminated clothing, contaminated cleanup items, two leakage-transfer pumps, and about 15,000 gallons of liquid.
Chemical decontamination of reactor coolant system	100,000 to 500,000 gallons of liquid (see Appendix P).
Chemical decontamination of reactor coolant drain tanks	Miscellaneous tools, equipment, piping, contaminated clothing; contaminated cleanup items; 30,000 to 45,000 gallons of liquid for feed and bleed (see Appendix P); and 15,000 to 25,000 gallons of liquid for recirculation (see Appendix P).
Reactor coolant pump decontamination	
In-place	Miscellaneous tools and equipment, contaminated clothing, and contaminated cleaning items (see Appendix P). Small quantities of liquids.
Out-of-containment	Miscellaneous tools and equipment, contaminated clothing, contaminated cleaning items (rags, brushes, etc.), pump motors, pump shafts, pump impellers, and pump seals. Small quantities of liquids.

6.5.4.2 Accident Scenarios

To eliminate the possibility of recriticality, it will be necessary to carefully inspect the reactor coolant system to ensure that any remaining fuel particle deposits represent much less than a critical mass before the boric acid concentration is reduced below 3000 ppm. Another potential accident involves the spill of decontamination liquid from the RCS into the reactor building while the primary system pumps are operating. It is assumed that 10 percent (2000 Ci) of the maximum activity content in the untreated liquids (see Appendix G, Table G.2) is spilled before corrective action occurs. Because the solution will be at a temperature of about 200°F, the fraction of activity that would become airborne is somewhat higher thant that at ambient temperature. Accordingly, it is estimated that 0.1 percent (2 Ci) of the activity is transmitted to the building HEPA filters, which release 0.1 percent (0.002 Ci) to the atmosphere in the form of cesium and strontium, as shown in Table 6.14.

Table 6.14. Airborne Releases of Principal Radionuclides due to a Spill of Decontamination Liquids from the RCS into the Reactor Building

Radionuclide	Total Release (Ci)
Cs-137	1.0×10^{-3}
Cs-134	1.6×10^{-4}
Sr-90	7.8×10^{-4}
Sr-89	8.0×10^{-5}

6.5.5 Environmental Impacts

6.5.5.1 Occupational Doses

The cumulative occupational dose, the expected calendar time, and the total work force required for each activity are listed in Table 6.15.

Occupational dose, calendar time, and personnel required are based upon the person-hours and crew size given for each activity in Table 6.10. Occupation dose (person-rem) is calculated as the product of person-hours and radiation field. The calendar time is based upon continuous operation, three shifts per day, with the crew sizes indicated in Table 6.10. In all cases the work force requirements were determined using a dose to individual workers of 1 rem per quarter, rather than by the available work hours in a quarter of a year.

The occupational dose associated with these activities ranges from about 108 to 1740 person-rem. The 108 person-rem value was obtained by adding all the lowest numbers in the "Radiation Dose, Nominal" column in Table 6.15. Similarly, the 1740 person-rem value is based on the highest numbers. The decontamination operations could be accomplished with a work force of from 85 to 500 individuals. Of the five operations given in Table 6.15, only the first two can be done concurrently. Thus, 85 persons would be needed for these two operations performed concurrently. The remaining three operations cannot be done concurrently, so that no additional persons need to be included.

The occupational dose estimates include about 2 to 4 person-rem from inhalation of tritiated water vapor. The tritiated water will be produced by ebaporation from the fuel transfer canal and spent fuel pool, which will be filled with processed water having a tritium concentration of about 0.5 μCi/mL. Evaporation from these open pools of water is expected to produce an average tritium concentration of about 2×10^{-6} μCi/mL in the atmosphere in work areas in the vicinity of

Table 6.15. Occupational Dose, Calendar Time, and Personnel Required for Activities in Decontamination of the Reactor Primary System

Activity	Radiation Dose[a] (person-rem)		Calendar Time (days)		Personnel Required	
	Nominal	Maximum	Nominal	Maximum	Nominal	Maximum
Flushing reactor coolant system						
Using reactor coolant pumps	80	400	40	90	80	400
Using system pressure	300	1000	200	200	100[b]	330[b]
Draining and flushing reactor coolant drain tank						
Drain to containment building sump	5	50	2	6	5	50
Drain to reactor coolant bleed holdup tanks	10	70	7	20	10	70
Chemical decontamination of reactor coolant system						
Using reactor coolant pumps	10	90	20	40	10	90
Using new pumps	20	100	30	50	20	100
Chemical decontamination of reactor coolant drain tank						
By feed and bleed	3	30	6	10	8	30
By recirculation	8	70	10	30	8	70
Reactor coolant pump decontamination						
In-place	10	90	20	40	10	90
Out-of-containment	30	500	20	70	30	500

[a]Rounded to one significant figure.

[b]Work performed over three calendar quarters by same personnel.

these pools (see Sec. 6.3.5.1). The staff estimates that primary system cleanup tasks will require workers to spend 5000 to 10,000 person-hours of work effort in these areas. For the estimate of maximum personnel required, 470 persons may be needed for the first two operations, but 500 could be necessary for the last.

Based on the above values of occupational dose and work force required, the expected number of additional cancer mortalities in the work force would range from 0.014 to 0.23. The added probability that the average individual worker would die of cancer would range from 1 in 2200 to 1 in 6000. The expected number of additional genetic effects in the offspring of the work force exposed would range from 0.028 to 0.45.

6.5.5.2 Offsite Doses

The offsite dose estimates to the maximum exposed individual for the decontamination of primary system components are listed in Table 6.16. These estimates are based on the source term described in Section 6.5.4.1 (300 Ci/year of H-3). The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3. The total-body population dose received by the human population within a 50-mile radius from these activities was estimated to be 3 person-rem.

Table 6.16. Dose Estimates to the Maximum Exposed Individual due to Tritium during Decontamination of Primary System Components

Location	Pathway	Dose (mrem) Total-Body[a]
Nearest garden[b]	Inhalation	2.1×10^{-2}
	Ground Shine	0
	Vegetable Use	7.2×10^{-2}
	Total	9.3×10^{-2}
Nearest milk goat	Inhalation	1.1×10^{-2}
	Ground Shine	0
	Goat Milk Use	8.2×10^{-2}
	Total	9.3×10^{-2}
Nearest cow milk and garden	Inhalation	2.3×10^{-2}
	Ground Shine	0
	Vegetable Use	7.9×10^{-2}
	Cow Milk Use	3.2×10^{-2}
	Total	1.3×10^{-1}

[a]Doses were calculated for GI-tract, bone, liver, kidney, thyroid, lung, and skin. The dose estimates for all organs except bone were the same as that for total-body. The estimates for bone were zero. Doses were calculated for four age groups: adults, teenagers, children, and infants. The largest of each group are listed in this table. Infant doses are listed for the nearest milk goat location and child doses are listed for the other two.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

6.5.5.3 Postulated Accidents

The type of accident for which dose estimates are presented here is the spill of decontamination liquid from the RCS into the reactor building. The accident scenario and source terms are discussed in Section 6.5.4.2. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses is discussed in Section 10.4. Table 6.17 lists the dose estimates for the maximum exposed individual.

6.5.5.4 Psychological-Socioeconomic Effects

The activities considered in this section include reactor cooling system inspection, removal of the reactor pressure vessel head and internal components, and decontamination of primary system components. The staff concludes that although these activities pose negligible offsite health effects, people living in the vicinity of the reactor may view the removal of fuel elements (defueling) and primary system decontamination as threatening. This option is based on the existing low level of anxiety generated by the uncertainties surrounding the impacts of the original accident and subsequent decontamination actions.

The level and proportion of the public's anxiety related to the removal of fuel elements is associated with several factors, including: (1) unforseen delay, danger or controversy surrounding core examination, (2) the condition of the core, (3) availability of technology and skilled personnel to safely defuel the reactor, and (4) the capability of Met-Ed to manage, and NRC to oversee, the decontamination process.

As additional information becomes available from Met-Ed and the news media, and as the certainty and scope of the task is formally defined, stressors should shift from uncertainty to believed threat associated with actual decontamination procedures. Thus, the local public should have less concern with the activities as they safely proceed. However, descriptions of the defueling process are highly technical and open to diverse public interpretation. Differing opinions have in the past lead to controversy and to a belief that hidden danger exists. The degree to which actions remain on schedule and have consistent interpretation will ultimately determine the psychological impact of decontamination procedures. In addition, the factuality and balance of the media's coverage of decontamination will strongly affect psychological consequences.

The hypothesized accidents, although having negligible offsite effects, are expected to aggravate existing public concerns and produce increased levels of anxiety. The level and duration of anxiety would be linked to the length of the accident period, the initiating threat, the credibility of Met-Ed and NRC, the level of controversy or uncertainty generated, and/or the quality of media coverage.

6.5.6 Economic Costs

The work effort (person-hours) estimated for the various alternatives in Section 6.5.3 include only the direct work effort, mostly conducted in the reactor building in a radiation environment, for the described alternatives. To evaluate the economic costs of these alternatives, all associated costs need to be included--labor, equipment, and facilities. Estimates of the economic costs for the best- and worst-case alternatives are given in Table 6.18. These costs are estimated on a consistent relative basis, and do not necessarily reflect all costs on an absolute basis. Thus, these costs should be considered only as relative costs for the two scenarios (best case vs. worst case). The basis for the costs shown here are presented in Appendix K.

About 45 percent of the labor costs shown in the table are estimated to be incurred in preparation for the direct work effort. Most of this effort consists of training and procedure preparation. Estimated equipment costs consist primarily of special filters, pumps, heat exchangers, and system piping modifications needed to accomplish the described cleanup activities. Facility needs in addition to those mentioned in other sections of this document would consist of a filter backflush facility. This facility would permit removal of particulates from and subsequent reuse of cartridge-type filters.

Table 6.17. Dose Estimates for the Maximum Exposed Individual Due to Spilling RCS Water in the Reactor Building

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	4.8×10^{-3}	7.4×10^{-2}	8.8×10^{-4}
	Ground Shine	6.6×10^{-3}	6.6×10^{-3}	6.6×10^{-3}
	Vegetable Use	1.4	5.6	1.5×10^{-2}
	Total	1.4	5.7	2.2×10^{-2}
Nearest milk goat	Inhalation	1.9×10^{-3}	3.0×10^{-2}	6.6×10^{-4}
	Ground Shine	6.6×10^{-3}	6.6×10^{-3}	6.6×10^{-3}
	Goat Milk Use	3.0×10^{-1}	1.6	9.2×10^{-1}
	Total	3.0×10^{-1}	1.6	9.2×10^{-1}
Nearest cow milk and garden	Inhalation	3.2×10^{-3}	5.0×10^{-2}	6.2×10^{-4}
	Ground Shine	6.6×10^{-3}	6.6×10^{-3}	6.6×10^{-3}
	Vegetable Use	1.4	5.6	1.5×10^{-1}
	Cow Milk Use	8.8×10^{-2}	3.8×10^{-1}	1.1×10^{-1}
	Total	1.5	6.0	2.6×10^{-1}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children. The dose estimates for the nearest goat location are for infants.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: Nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 6.18. Economic Costs of Decontamination of Primary System Components (thousands of dollars)

Item	Low	High
Labor	1210	2410
Equipment	1240	4390
Facilities	100	100
TOTAL	2550	6900

References--Section 6

1. "Analysis of Three Mile Island - Unit 2 Accident," Nuclear Safety Analysis Center, NSAC-1, Palo Alto, CA, July 1979.

2. "Three Mile Island, A Report to the Commissioners and to the Public," U.S. Nuclear Regulatory Commission Special Inquiry Group, Mitchell Rogovin, Program Director, January 24, 1980.

3. Memorandum from D.J. Malloy to J.B. Heineman, Argonne National Laboratory, "Some Theoretical Limiting Constraints for TMI-2 Recriticality," April 14, 1980.

4. "Scoping Studies of the Alternative Options for Defueling, Packaging, Shipping, and Disposing of the TMI-2 Spent Fuel Core," Allied-General Nuclear Services, AGNS-35900-1.5-79, September 1980.

5. J.A. Ayres, "Equipment Decontamination with Special Attention to Solid Treatment," Battelle Pacific Northwest Laboratories, BNWL-B-90, Richland, WA, June 1971.

6. J.A. Ayres, "Decontamination of Nuclear Reactors and Equipment," Ronald Press, 1970.

7. WATER TREATMENT PROCESSES AND PROCESSED WATER DISPOSAL

Large volumes of water were contaminated as a result of the accident. Prior to the onset of processing, about 370,000 gallons of this water were present in the Auxiliary and Fuel Handling Building (AFHB) tanks. An additional 700,000 gallons of this water collected and is located in the reactor building (RB) sump and about 96,000 gallons is in the reactor coolant system (RCS). The water in these three locations is referred to as accident generated water, which is specifically defined as:[1]

(a) Water that existed in the TMI-2 auxiliary, fuel handling, and containment buildings including the primary system as of October 16, 1979, with the exception of water which as a result of decontamination operations becomes commingled with nonaccident-generated water such that the commingled water has a tritium content of 0.025 μCi/mL or less before processing.

(b) Water that has a total activity of greater than one μCi/mL prior to processing except where such water is originally nonaccident water and becomes contaminated by use in cleanup.

(c) Water that contains greater than 0.025 μCi/mL of tritium before processing.

Per the February 27, 1980 agreement between the City of Lancaster, Metropolitan Edison Company, and the NRC, accident generated water will not be discharged into the Susquehanna River (barring unforseen emergencies) until the completion of this Programmatic Environmental Impact Statement or January 1, 1982, whichever is earlier. In accordance with the agreement, there have been no discharges of accident generated water from the site. There have, however, been releases of water from the TMI site which contained minute amounts of radioactivity which originated from Unit 2. These releases are made in accordance with the agreement and pertain to the effluents from the Industrial Waste Treatment System (IWTS). This system processes water from normally non-radioactive areas of the TMI Units 1 and 2 such as the Turbine Buildings and Control and Service Buildings. The water from these areas contains trace amounts of contamination. The total releases from IWTS are estimated in excess of ten million gallons per year. During the last half of 1980, water containing less than one thousandth (0.001) of a curie of tritium and one ten thousandth (0.0001) of a curie of other radioisotopes (e.g., cesium) was released from Unit 2 sources. These minor releases are expected to continue throughout the cleanup but have virtually no significance on the impacts discussed in this section.

The water in the AFHB has been decontaminated by processing through an ion exchange system referred to as EPICOR II and the processed water is stored in various tanks onsite. The remaining RB and RCS water must be either removed from its present location or processed for the cleanup to proceed. This section discusses the alternatives considered for disposition of RB and RCS accident water. Where these alternatives lead to liquid effluents or processed water, the alternatives considered for ultimate disposition of this water are also described. Processed water reuse is discussed in Appendix F.

Large volumes of water will be needed for flushing, draining, and decontamination of piping systems, equipment, and interior building surfaces. These decontamination activities also involve mixing this water with strong chemical agents and detergents to enhance decontamination effectiveness leading to chemical-based decontamination solutions during cleanup activities. These water-based and chemical decontamination solutions will become contaminated. The alternatives considered for treatment of these decontamination solutions are also discussed in this section.

7.1 LIQUID WASTE TREATMENT

Liquids involved in the TMI-2 decontamination will require further processing to permit their safe disposal in accordance with the staff's proposed use of the effluent criterion in Appendix I

as discussed in Section 1.6.3.2; these liquids include those directly generated during the March 28, 1979 accident (accident water) as well as liquids contaminated during the cleanup operations. The sources of TMI-2 liquids that will have to be treated and their estimated radioactivity inventories are shown in Table 7.1.

Table 7.1 Summary of Estimated TMI-2 Liquid Waste

Source of Liquid Waste	Volume (gallons)	Curie Inventory In Untreated Liquid	
		Minimum	Maximum
1. AFHB Chemical Decon Solutions	7,000	60	60
2. RB Sump Water	700,000	500,000	500,000
3. RCS Water	96,000	20,000	20,000
4. RCS Flush and Drain	250,000[a]	20,000	100,000
5. RB Decon Solutions			
(a) Water Based	150,000[a]	90	90
(b) Chemical	40,000	10	10
6. RCS Decon Solutions[b]			
(a) Water based	100,000[a]	2,000	20,000
(b) Chemical	500,000[a]	2,000	20,000

[a]Processed water could be used for the cleanup activities resulting in the generation of this liquid waste.

[b]The RCS water-based and chemical decontamination processes are mutually exclusive. Either the water-based or chemical process will be used in the decontamination of the RCS.

To compare alternative treatment systems on a relative basis, the performance of the systems considered was inferred from (1) the performance of the EPICOR II system used to clean up AFHB liquids (2) laboratory-scale tests on zeolite-based ion-exchange systems and (3) the performance of similar systems used to treat liquid wastes containing radionuclides and chemical contaminants of the same or similar species to those present in TMI-2 liquids. The assumptions used to characterize treatment system performance are conservative relative to the decontamination factors that could be achieved. Moreover, the actual performance of any treatment system can be adjusted by varying operating criteria to achieve the decontamination factors desired for a particular source of liquid waste. Therefore, while the performance of the systems will vary, the staff believes that the performance characteristics considered are representative and provide a consistent basis for comparing alternative treatment systems. The performance parameters which characterize each of the systems considered are discussed in detail in Appendix G.

7.1.1 Status and Specific Considerations

7.1.1.1 Efforts to Date

Processing of the accident water in the Auxiliary and Fuel Handling Buildings began in the fall of 1979 using the EPICOR II system and has been essentially completed. In addition, flushing and

draining of the activity in the AFHB piping and tanks and the subsequent processing of the contaminated water has largely been completed. There will be continued inleakage collected in the AFHB sump, however, the activity associated with this water is assumed to derive from primary system leakage to the AFHB. Impacts related to disposition of this activity are included in the discussions that follow regarding treatment of the activity in the RCS water.

One objective of processing AFHB accident water was to make the AFHB tankage available for contingencies such as the storage of reactor building sump water. The initial composition of the water in the AFHB before processing, as well as the initial location of the water, is given in Table 7.2.[2] The volume shown in this table is about 370,000 gallons. Due to in-leakage and flushing operations, the volume actually processed through the EPICOR II system is about 570,000 gallons--501,000 gallons through September 30, 1980 plus about 68,000 gallons thereafter. This processed accident water is stored in tanks (e.g., the Unit 2 Borated Water Storage Tank) which in some cases contained other slightly contaminated water. The commingling of these various liquids has led to a current inventory of about 743,000 gallons. The storage locations of this water and its major constituents are presented in Table 7.3.[3]

About 2200 gallons of chemical decontamination solutions were generated as of September 30, 1980 from AFHB cleanup activities. These liquids were immobilized with vinyl ester styrene (VES) and packaged for offsite disposal (see Section 8).

7.1.1.2 Projected Requirements

Table 7.4 shows the major radionuclides present in the liquid waste sources identified in Table 7.1. The characteristics of each liquid waste source in this table are described below.

Reactor Building Sump Water

About 700,000 gallons of contaminated water (sump water) floods the reactor building basement to a level of about 8 ft above the basement floor. This water is primarily from the reactor coolant drain tank that overflowed when the reactor pressurizer relief valve stuck open early in the accident. Concentrations of dissolved radionuclides and important chemical species in the reactor building sump water are given in Table 7.5. Estimates of the filterable solids in the sump water are given in Table 7.6.[4,5] Oils and grease also are likely components of the reactor building sump water, and the presence of these materials and any colloidal matter must be considered for the various process alternatives.

Reactor Building Decontamination Liquids (Water Based)

The initial decontamination of the reactor building interior and equipment surfaces could involve the use principally of water sprays or steam cleaning methods. The volume of liquid wastes that could be generated in these operations is estimated to be about 150,000 gallons. Processed water could be used for this activity. The concentrations of radioactvity in decontamination liquids that could be generated through the use of remote washing techniques have been estimated to be about 0.2 μCi/mL. These wash liquids would drain down into the reactor building sump and may be added to the existing reactor building sump water for processing. This would depend on the radiation fields at the operating levels in the reactor building and on the schedules developed in the various water processing operations. If necessary, these liquids could also be treated separately.

AFHB and Reactor Building Chemical Decontamination Liquids

Chemical decontamination operations in the AFHB and the reactor building are expected to generate chemical waste solutions that have a full range of detergents and complexing agents. The volume of these chemical decontamination liquids to be processed is expected to be on the order of 40,000 gallons for the reactor building and 7,000 gallons for the AFHB. All the chemical decontamination liquids could be blended, resulting in an average concentration of radioactivity of about 0.4 μCi/mL.

Table 7.2. Composition of AFHB Water Prior to Start of Processing[a,b]

Location	Volume (gallons)	Radionuclide Concentrations (μCi/mL)		
		Cs-137	Cs-134	H-3
Reactor coolant bleed tank A	77,250	28	6.5	0.23
Reactor coolant bleed tank B	77,250	35	7.6	0.27
Reactor coolant bleed tank C	77,250	35	8.7	0.29
Neutralizer tank A	8,780	2.5	0.56	-
Neutralizer tank B	8,780	3.3	0.72	-
Miscellaneous waste holdup tank, auxiliary building sump and sumptank, miscellaneous sump	13,500	10.1	2.4	0.03
Tank farm B	93,000	-	-	-

[a]These values pertain to the situation following the accident but prior to water processing.

[b]Based on memo from W.K. Lehto to J.A. Opelka, Argonne National Laboratory, Subject: Compilation of Information and Data from Visit to TMI-2 on May 6, 1980; May 12, 1980.

Table 7.3 Storage Locations, Volume, and Radionuclide Concentrations of Processed AFHB Accident Water[a]

Storage Tank	Volume[b] (gallons)	Radionuclide Concentrations (μCi/mL)		
		H-3	Cs-137	Cs-134
BWST	422,000	1.1×10^{-1}	3.6×10^{-4}	1.6×10^{-4}
COT-1A	240,000	1.8×10^{-1}	4.6×10^{-6}	7×10^{-7}
CCT-1	26,000	6.6×10^{-2}	6.8×10^{-6}	1×10^{-6}
CCT-2	45,000	6.4×10^{-2}	1.9×10^{-6}	3×10^{-7}
T9B[c]	10,000	6.6×10^{-2}	6.8×10^{-6}	1×10^{-6}
TOTALS	743,000	1.3×10^{-1}	2×10^{-4}	$< 9 \times 10^{-5}$

[a]Based on Letter (with enclosures entitled, 1. Radionuclide Distribution of Auxiliary Building Water, 2. Agenda for Dec. 18, 1980 meeting of Task Group SC-38) from B.J. Snyder, U.S. Nuclear Regulatory Commission, to F.P. Parker, Vanderbuilt University, December 15, 1980.

[b]Rounded.

[c]No measured data--values shown are assumed based on CCT-1 data.

Table 7.4. Radioactivity in TMI-2 Untreated Liquid Waste

Source of Liquid Waste	Average Radionuclide Concentrations (μCi/mL[a])	Isotope Inventory (Ci)[a]					
		H-3	Cs-137	Cs-134	Sr-90	Sr-89	Others[d]
1. AFHB Chemical Decon Solutions	2	-[e]	51	8	1	-	-
2. RB Sump Water	190	2,300	430,000	66,000	7,000	190	75
3. RCS Water	60	30	9,000	1,500	7,800	820	110
4. RCS Flush & Drain[b]	100	-	49,000	7,500	39,000	4,100	550
5. RB Decon Solutions							
(a) Water based	0.2	-	77	12	1		-
(b) Chemical	0.1	-	8	2			-
6. RCS Decon Solutions[b]							
(a) Water Based[c]	50	-	9,900	1,500	7,800	820	110
(b) Chemical[c]	10	-	9,900	1,500	7,800	820	110

[a]Rounded.

[b]The curie content corresponds to the maximum estimated values in Table 7.1.

[c]These two solutions are mutually exclusive. Either the water-based or chemical decon solution will be generated during decontamination of the RCS.

[d]See Table G.8 for detailed distribution of other radionuclides.

[e]"-" denotes less than one curie.

Table 7.5. Estimated Radionuclide Concentrations of Dissolved Constituents of Reactor Building Sump Water as of September 30, 1980[a]

Constituent	Concentration (μCi/mL)	Constituent	Concentration (μg/mL)
H-3	0.95	U	0.028
Cs-134	24	Pu	3.3×10^{-5}
Cs-137	160	Na	1.2×10^{3}
Sr-90	2.6	B	2.0×10^{3}
Sr-89	0.07	Cl	15
Nb-95	2×10^{-5}	Al	3
Zr-95	2×10^{-5}	Ca	10
Ru-106	3×10^{-3}	Cu	10
Sb-125	0.02	Fe	1.8
Te-125m	5×10^{-4}	K	4
Te-127m	5×10^{-4}	Li	1.6
Te-129m	2×10^{-4}	Ni	3
Ce-144	2×10^{-3}	P	0.3
I-129	1.2×10^{-5}	S	9
		Zn	0.5

[a]Based on R.E. Brooksbank and W.J. Armento, "Post Accident Cleanup of Radioactivity at the Three Mile Island Nuclear Power Station," Oak Ridge National Laboratory, ORNL/TM-7091, February 1980; and D.O. Campbell, "Hot Cell Studies," Oak Ridge National Laboratory, presentation to General Public Utilities and U.S. Dept. of Energy staff members, January 31, 1980, but corrected for radioactive decay to September 30, 1980.

Table 7.6. Concentration of Radionuclides in Filterable Solids in Reactor Building Sump Water as of September 30, 1980[a,b]

Nuclide	Concentration (μCi/mL)
Cs-134	0.11
Cs-137	0.71
Nb-95	1.7×10^{-3}
Zr-95	0.037
Ru-106	0.18
Ru-103	1.4×10^{-4}
Ce-141	3.8×10^{-6}
Ce-144	0.083
Co-58	7.7×10^{-4}
Co-60	0.011
Sr-89+Sr-90	8.6
Ag-110m	5.9×10^{-3}
Te-127m	0.27

[a]Total solids = 13×10^3 kg (based on 0.5% solids content).

[b]R.E. Brooksbank and W.J. Armento, "Post Accident cleanup of Radioactivity at the Three Mile Island Nuclear Power Station," Oak Ridge National Laboratory, ORNL/TM-7091, February 1980; and D.O. Campbell, "Hot Cell Studies," Oak Ridge National Laboratory, presentation to General Public Utilities and U.S. Dept. of Energy staff members, January 31, 1980, but corrected for radioactive decay to September 30, 1980.

Reactor Coolant System Water (Primary Water)

The reactor coolant system (RCS) contains about 96,000 gallons of water contaminated with fission products and reactor core debris from the accident. The processing of this water is subject to somewhat different constraints than processing AFHB or reactor building sump water. Although removal of contamination from the primary water is necessary to permit safe access to the reactor vessel, the decontamination processes and subsequent makeup must still maintain adequate boron to provide neutron-absorption capability during subsequent reactor defueling. Estimated concentrations of important constituents in the primary water are given in Table 7.7.[6]

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Ce-141	3.8×10^{-6}
Ce-144	0.083
Co-58	7.7×10^{-4}
Co-60	0.011
Sr-89+Sr-90	8.6
Ag-110m	5.9×10^{-3}
Te-127m	0.27

[a]Total solids = 13×10^3 kg (based on 0.5% solids content).

[b]R.E. Brooksbank and W.J. Armento, "Post Accident cleanup of Radioactivity at the Three Mile Island Nuclear Power Station," Oak Ridge National Laboratory, ORNL/TM-7091, February 1980; and D.O. Campbell, "Hot Cell Studies," Oak Ridge National Laboratory, presentation to General Public Utilities and U.S. Dept. of Energy staff members, January 31, 1980, but corrected for radioactive decay to September 30, 1980.

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Table 7.7. Estimated Concentrations of the Principal Constituents of the Reactor Coolant System Water as of September 30, 1980[a]

Constituent	Concentration (μCi/mL)	Constituent	Concentration (μg/mL)
H-3	0.08	Na	1.1×10^3
Cs-134	4.5	B	3.7×10^3
Cs-137	29		
Sr-89	2.4		
Sr-90	23		
Nb-95	8×10^{-4}		
Zr-95	5×10^{-4}		
Ru-106	0.1		
Sb-125	4×10^{-3}		
Te-125m	1.6×10^{-3}		
Te-127m	0.03		
Te-129m	6×10^{-5}		
Ce-144	0.03		
Co-58	2×10^{-4}		

[a]Based on "TMI-Data Base," compiled by Argonne National Laboratory - W.K. Lehto, November 21, 1980.

Reactor Coolant System Flush and Drain Water

The flushing and draining of the RCS will commence following defueling of the core. About 250,000 gallons of water will be generated from the draining and flushing of the RCS and will require processing. Processed water could be used for this activity. The concentrations of contaminants expected in the drain and flush water will depend on at least two variables--(1) the amount of radioactivity released to the water during the flush and drain operation and (2) the amount of water that will be required to flush and drain the system. It is estimated that between 20,000 Ci and 100,000 Ci could be present in this water with resulting concentrations ranging from about 20 μCi/mL to 100 μCi/mL.[7] The choice of processing alternatives depends on the actual concentration of radioactivity in the water and the volume of contaminated water that must be treated.

Reactor Coolant System Chemical Decontamination Liquids

Chemical decontamination of the reactor coolant system following drain and flush operations may involve use of relatively mild reagents such as in the CAN DECON proprietary process or more chemically aggressive chemicals such as alkaline permanganate. These decontamination techniques are described in detail in Appendix P. It is estimated that from 2,000 to 20,000 Ci of radioactivity could be contained in the reactor coolant system decontamination liquids, depending on the extent of contamination within the coolant system surfaces being treated. If the CAN DECON process is employed, about 100,000 gallons of decontamination liquids would result. If a more aggressive chemical decontamination is pursued, as much as 500,000 gallons of liquid might require eventual processing. This combination of radioactivity and volumes leads to liquid waste concentrations between 1 μCi/mL and 50 μCi/mL. Processed water could be used for these activities.

Some of these liquid wastes in the form of accident water and AFHB chemical decontamination solutions already have been generated. Other liquids will be generated during the project. The

relative time periods over which these liquids could be generated and be available for treatment are shown in Figure 7.1.

7.1.2 Alternatives Considered

The accident generated water in the reactor building sump and the primary system cannot be left in its present condition and location if the cleanup effort is to proceed. Some of the alternatives considered for disposition of this water involve its cleanup through the use of filtration, ion exchange, evaporation and bitumenization techniques. Others include transfer from its present location to onsite storage facilities or processing the water for transport and disposal at a low-level radioactive waste disposal facility. As decontamination solutions are generated, they too must be either cleaned up, stored, or processed and shipped offsite. The alternatives considered for accident-generated water and decontamination solutions are discussed below.

7.1.2.1 Long-Term On-Site Storage

The onsite storage alternative involves transfer of the accident water from its present locations in the reactor building and primary system to storage tanks. This water could be transferred to tanks within the plant, if available, or to newly constructed exterior tanks. In either case, the storage tanks would have to be heavily shielded to reduce radiation levels in areas near the tanks.

7.1.2.2 Direct Immobilization

This alternative involves direct immobilization of the accident water using a binder material such as cement or VES for either temporary onsite storage or offsite shipment to a commercial shallow land burial facility. The processes that use these binder materials to immobilize liquid radioactive waste to make it acceptable for transport and disposal are described in Appendix H. Implementation of this alternative would involve construction of a waste immobilization and handling facility. The activity level of the accident water would require that this facility be remotely operated.

7.1.2.3 Treatment Processes and Systems

The treatment alternatives considered involve processing the liquid through treatment systems which remove the contaminants. One or more of the following processes can be used to achieve the desired cleanup objective; (1) filtration, (2) ion exchange, (3) evaporation, and (4) bitumenization. The principles involved in each process and their applicability to cleanup of TMI-2 liquid wastes are discussed below.

<u>Filtration</u>

Filtration is a physical process whereby particles suspended in a liquid are separated from it by forcing the liquid through a porous medium. As the liquid passes through the porous medium, the suspended particles are trapped on the surface of the medium or within the body of the medium itself. The mechanical device which contains or supports the filter medium is referred to as a filter cartridge. Loading of the filter medium to its capacity requires removal of the particulates from the medium (e.g. backwashable filter) or replacement of the medium. Replaceable filters will be used in processing most TMI-2 liquid wastes. Filtration is applicable to TMI-2 liquid wastes as an initial step in a process. It is not an appropriate treatment process in and by itself, because much of the radioactivity is in solution and thus is not removed by filtering.

<u>Ion Exchange</u>

Ion exchange involves the removal of ionic species from an aqueous phase. The earliest applications of ion exchange were "water softening"--the substitution of sodium ions for calcium and magnesium ions in water to reduce its hardness. These applications were initiated in the late 1800's and early 1900's using natural and synthetic zeolites (aluminosilicate minerals). Synthetic ion exchange resins were discovered in the late 1930's and were developed rapidly. At present, the synthetic organic resins are used in most ion exchange applications.

In practice, the liquid containing the ionic species is placed in contact with the ion exchange media selected to preferentially remove specific ions. To provide this contact, the ion exchange media is placed in demineralizer vessels or lines through which the liquid flows. Vessels containing the ion exchange media for treatment of TMI-2 liquid wastes are referred to as (1) prefilter

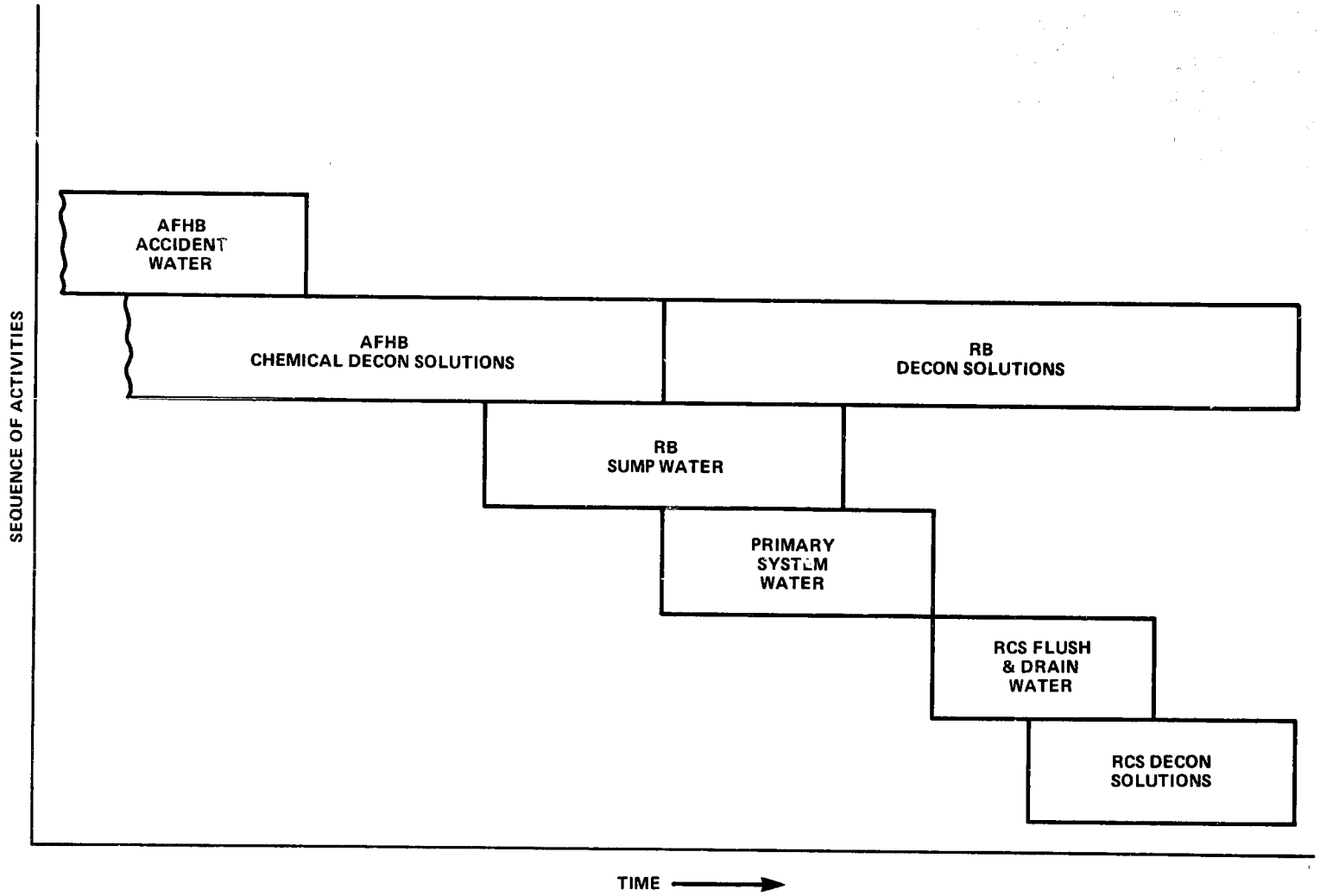

Figure 7.1 Typical Sequence for Liquid Waste Processing

and zeolite liners, (2) cation liners, and (3) mixed bed liners (liners containing anion and cation media). These designations refer to the type of ion exchange media in the vessel. Once a system is designed, the performance of the media in each vessel can be adjusted to suit a particular liquid waste treatment need, as specific as individual batches of water to be processed.

Ion exchange is appropriate for accident water and water based decontamination solutions. It is not appropriate for treatment of chemical decontamination solutions because the chemical nature of these liquids would lead to rapid breakdown and plugging of ion exchange media.

Evaporation

Evaporation is the removal of a volatile solvent from a nonvolatile solution by boiling the solution. Evaporation is an ancient art. Open, direct-fired pans containing salt water were used for making salt in the Middle Ages. Solar energy has long been used for concentrating brine in open ponds.

The solution to be evaporated at TMI-2 is the liquid waste and the volatile solvent to be vaporized is water, essentially free of contaminants. The contaminants are retained in the concentrated solution (or bottoms) while the relatively clean volatiles are vaporized and then condensed as processed water or distillates. The major parameters which describe evaporator performance are (1) the decontamination factor, which is defined as the ratio of the contaminant concentrations in the feed solution and the concentrations in the distillate, and (2) the volume reduction factor which is the ratio of feed solution volume to the concentrated "bottoms volume." Evaporation is only appropriate for treatment of TMI-2 liquid wastes with low to moderate concentrations of dissolved solids.

Bitumenization

This is a process which combines evaporation and immobilization in one step. The evaporator bottoms are immobilized in an asphalt-like material (bitumen) and the vaporized water removed from the liquid waste is condensed for further treatment as processed water. Bitumenization is only appropriate for TMI-2 liquids of low to moderate radioactivity concentration with at least 5 weight percent solids content.

Treatment Systems

The four basic treatment processes can be combined to form treatment systems for TMI-2 liquid waste. Figure 7.2 shows a simplified diagram for these systems. The specific treatment systems considered for TMI-2 liquid waste include the following:

1. Submerged Demineralizer System (SDS) - This system combines filtration with three stages (zeolite, cation, and mixed bed media) of ion exchange.

2. Modified Submerged Demineralizer System - This system is similar to the SDS above with the exception that it employs a larger cation ion exchanger.

3. SDS/EPICOR II System - This system combines the SDS (alternative number 1) with polishing (or final cleanup) in EPICOR II (alternative number 6).

4. Zeolite/Evaporator System - This system combines filtration with ion exchange (zeolite media), evaporation, and ion exchange (mixed bed media).

5. Zeolite/EPICOR II System - This system combines filtration with ion exchange (zeolite media) and polishing by the EPICOR II system (alternative number 6).

6. EPICOR II System - This system combines three stages of ion exchange (layered organic and inorganic, cation, and mixed bed media).

7. Modified EPICOR II - This system is similar to EPICOR II (alternative number 6) with the exception that only inorganic (zeolite or zeolite/titanate) media is used in the first stage ion exchanger. The titanate is inorganic material which has strong removal specificity for strontium.

	ALTERNATIVE SYSTEM	FILTERING	STAGE 1 GROSS CS/SR REMOVAL	STAGE 2 CS/SR POLISHING	STAGE 3 FINAL POLISHING	EFFLUENTS
1.	SUBMERGED DEMINERALIZER SYSTEM (SDS)	FILTER	ZEOLITE BED	CATION BED	MIXED BED	PROCESSED WATER, EXPENDED FILTERS, INORGANIC AND ORGANIC MEDIA
2.	MODIFIED SDS SYSTEM	FILTER	ZEOLITE BED	LARGE CATION BED	MIXED BED	PROCESSED WATER, EXPENDED FILTERS, INORGANIC AND ORGANIC MEDIA
3.	SDS/EPICOR II SYSTEM	FILTER	ZEOLITE BED PREFILTER	CATION BED CATION BED	MIXED BED MIXED BED	PROCESSED WATER, EXPENDED FILTERS, PREFILTERS, INORGANIC, AND ORGANIC MEDIA
4.	ZEOLITE EVAPORATOR SYSTEM	FILTER	ZEOLITE BED	EVAPORATOR	MIXED BED	PROCESSED WATER, EXPENDED FILTERS, INORGANIC AND ORGANIC MEDIA, EVAPORATOR BOTTOMS
5.	ZEOLITE EPICOR II SYSTEM	FILTER	ZEOLITE BED PREFILTER	CATION BED	MIXED BED	PROCESSED WATER, EXPENDED FILTERS, PREFILTERS, INORGANIC AND ORGANIC MEDIA
6.	EPICOR II SYSTEM		PREFILTER	CATION BED	MIXED BED	PROCESSED WATER, EXPENDED PREFILTERS AND ORGANIC MEDIA
7.	MODIFIED EPICOR II SYSTEM		ZEOLITE OR ZEOLITE AND TITANATE BED	CATION BED	MIXED BED	PROCESSED WATER, EXPENDED INORGANIC AND ORGANIC MEDIA
8.	EVAPORATOR RESIN SYSTEM	FILTER	EVAPORATOR	CATION BED	MIXED BED	PROCESSED WATER, EXPENDED FILTERS, AND ORGANIC MEDIA, EVAPORATOR BOTTOMS
9.	BITUMEN RESIN SYSTEM		EXTRUDER EVAPORATOR	CATION BED	MIXED BED	PROCESSED WATER, EXPENDED ORGANIC MEDIA, BITUMENIZED SOLID WASTE

Figure 7.2 Alternative Liquid Waste Treatment Systems

8. Evaporator/Resin System - This system combines filtration with evaporation and two stages (cation and mixed bed media) of ion exchange.

9. Bitumen/Resin System - This system combines evaporation and immobilization with two stages (cation and mixed bed media) of ion exchange.

The functions of each stage of the systems shown in Figure 7.2 and their applicability to TMI-2 liquid wastes are as follows:

Filtration Units

Filters are required upstream of ion exchange vessels to prevent solids from clogging the ion exchange media.

First Stage Processing Units

The function of the first stage is to remove large quantities of cesium and strontium from the liquid. In ion exchange vessels, zeolites and mixtures of zeolites and titanates are most effective for the higher concentration liquids. Organic resins can also be used but the radioactivity loading on these media is limited by the stability of the organic resin media. Thus, first stage organic resin ion exchange media cannot be used for RB sump water and may not be suitable for RCS (primary system) water and RCS flush and drain water.

The use of evaporators for first stage cleanup is also activity level limited. Bitumenization has the same limitations as an evaporator for gross removal of cesium and strontrium from high concentration activity liquids.

Second Stage Processing Units

Once gross removal of cesium and strontium is accomplished, the influent to the second stage is relatively low in concentration. Therefore, ion exchange units containing organic resins or an evaporator are both suited for further removal of cesium and strontium and other ionic contaminants. The vapors produced from first stage evaporation and bitumenization are condensed before processing through ion exchange vessels.

Third Stage Processing Units

The function of this stage is to polish the liquid through removal of trace contaminants. The mixed bed designation indicates that both anion and cation organic resins are present in the demineralizer vessel. The quality of the effluent from the second stage unit could be such that a mixed bed demineralizer is not necessary.

The details of the alternatives considered for the handling and treatment of TMI-2 radioactive liquid waste are presented in the following section (7.1.3). Appendix G presents detailed performance characteristics for the treatment systems.

7.1.3 Details of Liquid Waste Treatment Alternatives

Each alternative considered above was evaluated in detail to assess its suitability for treatment of TMI-2 liquid waste sources. This section presents the results of these evaluations, provides the basis for eliminating alternatives from further consideration, and compares the alternatives judged suitable for treatment of each waste type. Some details of system hardware and facilities are also presented.

7.1.3.1 Long-Term Storage

The accident water in the RB sump contains about 500,000 curies of radioactivity or about 0.7 curies per gallon. If this water were placed, unprocessed, in 55-gallon drums, each drum would contain about 35 curies and exhibit radiation levels of about 30 R/hr on its surface. About three feet of concrete shielding would have to be placed around this drum to reduce radiation levels to acceptable levels. Comparable shielding would have to be placed around storage tanks that would be needed to store this unprocessed liquid on site.

Tankage could be constructed and appropriate shielding could be installed. However, the major contaminants in this liquid have relatively long-half lives and any substantial reduction in the

activity level in the water to render it innocuous would take over 300 years. Removal of the water from the storage tanks any sooner would require that it be treated prior to disposition.

Several years would be required to license, design, and construct long-term high activity liquid waste storage facilities at the site. Even when constructed, these facilities could not, with adequate assurance, protect the public health and safety by providing isolation of the radioactive water from the environment for at least 300 years. This facility would require maintenance, and monitoring continuously for 300 years. The staff does not consider this option to be feasible, considering the length of time required for decay and the inability to ensure adequate institutional controls throughout this period. Furthermore, it is anticipated that the cleanup will not be considered complete without processing or removal of the accident water.

7.1.3.2 Direct Immobilization of Accident Water

Under this alternative, accident water (RB sump and RCS) would be immobilized in cement or VES and either stored onsite or transported offsite for disposal at a low-level radioactive waste disposal facility.

Table 7.8 summarizes the impact of implementing this alternative based on the approximately 800,000 gallons of accident water in the RB sump and RCS. The highlights of this summary include:

- Cumulative public radiation dose will be about 75 person-rem
- Cumulative occupational radiation dose will be about 825 person-rem
- Implementation will take 5 years
- Work in the Reactor Building would be deferred for about 4.5 years
- Total cost in 1980 dollars will be about $40 million
- 3900 cubic yards of cement would be required.

Based on these impacts, this alternative is not considered practicable for this high activity water, principally because of the high radiation doses to workers and the long delay in completing the cleanup.

Immobilization of AFHB chemical decontamintion solutions is currently being carried out using the VES technique described in Appendix H. Thus, direct immobilization without evaporation to reduce volume is still being considered for relatively low volume chemical decontamination solutions.

A separate discussion of untreated chemical decontamination solutions is presented in Section 8.2.

7.1.3.3 Zeolite-Based Treatment Systems

This section provides a detailed discussion of the zeolite-based treatment systems which include the Submerged Demineralizer System (SDS), the modified SDS, the SDS/EPICOR-II system, the zeolite/evaporator system, and the zeolite/EPICOR-II system. The licensee has designed and is presently constructing an ion exchange system for cleanup of high specific activity liquids. This system is designed to operate under water in Spent Fuel Pool B and is referred to as the Submerged Demineralizer System or SDS.[8]

A process flow diagram for the SDS system is presented in Figure 7.3. Filtration is performed by two filters, one rough and one fine. The filters are disposable cartridges about 1.7 ft in diameter and 4.5 ft long, with a volume of about 10 ft^3. Following filtration, the liquid is transferred to a batching tank. The next step involves processing through two parallel trains of three ion exchange vessels each. These vessels are also 1.7 ft ft in diameter and 4.5 ft long and constructed of stainless steel. The internal volume of each vessel is about 10 ft^3 and contains 8 ft^3 of zeolite (or zeolite/titanate) ion exchange media. This step removes over 99 percent of the cesium and strontium in the water. Further removal of cesium and strontium, plus removal of other ions, results from processing through two downstream parallel cation vessels, each containing 8 ft^3 of organic resins. These vessels are essentially the same size as the zeolite vessels but are constructed of carbon steel. Final polishing is performed by processing through a mixed bed ion exchange vessel about 6 ft in diameter by 6½ ft high. This vessel contains about 135 ft^3 of

Table 7.8. Impacts of Immobilization[a] and Offsite Disposal of Accident Water

A. Waste Characteristics
   1. Liquid Volume - 796,000 gal
   2. Average Radionuclide Concentration - 170 μCi/mL
   3. Liquid Content per Drum - 30 gallons
   4. Radioactivity per Drum - 19 Ci
   5. Drum Radiation Level - 15 R/hr

B. Solidification Operations
   1. Facility Design and Construction - 2 years
   2. Number of Drums to be Processed - 26,300
   3. Average Facility Throughput - 29 drums/day
   4. Time Required to Process all Drums - 907 days
   5. Total Time to Implement - 5 years

C. Packaging and Transportation
   1. Drums per Cask Shipment - 14
   2. Number of Cask Shipments - 1,800
   3. Cask Miles Travelled (Round Trip) - 9.7 million

D. Resource Use
   1. 55-Gallon Drums - 26,300
   2. Cement - 5,300 tons, or 3900 cubic yards
   3. Gasoline/Diesel Fuel - 1.64 million gallons
   4. Burial Site Land Use - 395,000 ft^3

E. Cumulative Radiation Dose
   1. Occupational (handling) - 79 person-rem
   2. Occupational (transport) - 617 person-rem
   3. Occupational (solidification) - 130 person-rem
   4. Public - 75 person-rem due to passing on route, 24 person-rem due to onlookers near truck

F. Costs
   1. Design and Construction - $3.5 million
   2. Operation - $3.2 million
   3. Transport and Disposal - $33 million

[a]Based on cement immobilization.

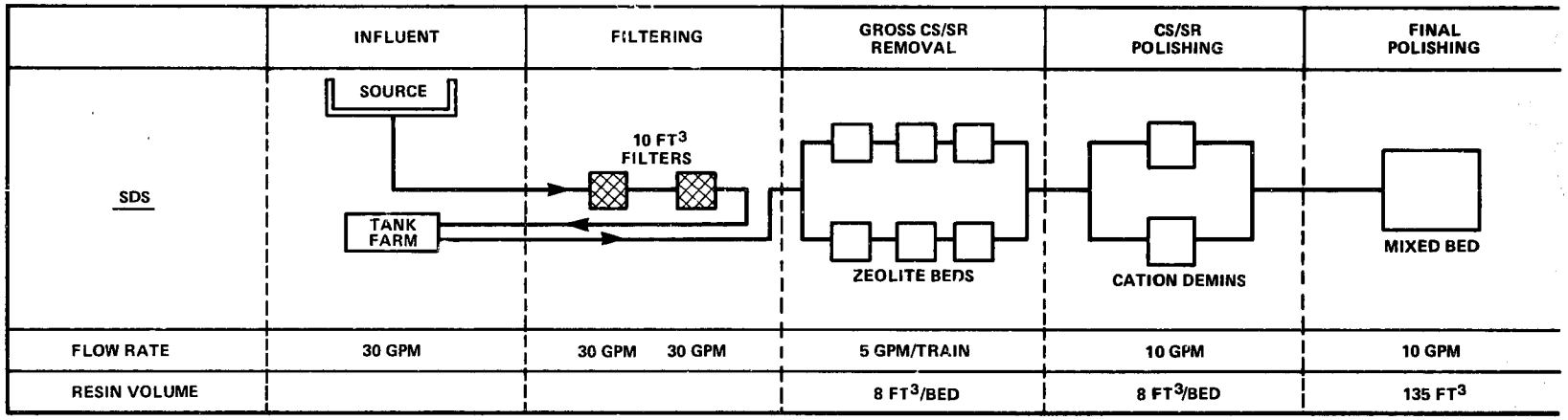

Figure 7.3 SDS Process Diagram

anion and cation organic resin. As filters or ion exchange media are expended, the filter cartridge or the vessel containing the media are removed and replaced. These expended filter cartridges and ion exchange vessels are then handled as solid waste.

The design flow rates shown on Figure 7.3 show a throughput of 10 gpm. Based on EPICOR II system experience, the average (including downtime) expected from an SDS type system is 2-3 gpm.[9]

RB Sump Liquid

Due to the high concentration of radionuclides in RB sump water, about 130 μCi/ml, systems which use zeolites (or zeolites and titanates) are best suited for cleanup of this liquid waste because of the radiation stability of inorganic ion exchange media.[10] Figure 7.4 shows the configurations of the zeolite systems considered and evaluated.

The performance of the SDS has been independently evaluated using small samples of RB sump water and laboratory scale ion-exchange columns.[11] These evaluations indicate that the current design may not adequately remove colloidal Cs and Sr, Ru, and Sb from the sump water. These limited evaluations also indicate that process changes may enhance system performance.

First, the size of the cation ion-exchange beds, two at 8 ft^3 each, is not operationally adequate to remove sodium ions in the RB sump water and the effective decontamination factor across these ion-exchange vessels could be one. The excess sodium ions in the cation bed effluents also hamper the performance of the mixed bed ion-exchange vessel reducing the decontamination factors for the above radionuclides that would otherwise be achieved. The effect of these factors is to reduce overall performance of the system to the point where effluents need further polishing if the processed water is to be disposed of by discharge to the Susquehanna River (see Section 7.2). The SDS cation and mixed bed vessels do provide the benefit of removing sodium, making the task of subsequent polishing easier. Second, colloidal forms of Cs and Sr that are not easily removed by once-through ion-exchange treatment appear to be present. Processing modifications such as addition of heat and holdup time may improve the removal of this colloidal Cs and Sr.

In order to evaluate the performance of SDS relative to alternative treatment systems, it was conservatively assumed that the worst case conditions described above for Cs, Sr, Ru and Sb removal occur in the full scale system. Actual system performance could yield improved decontamination factors in excess of those predicted by the laboratory experiments for these radionuclides. Additionally, processed water could be recycled through SDS or another polishing system to improve water quality.

The alternative referred to as the modified SDS system includes the first suggested change replacing the cation demineralizers with larger resin beds. Two parallel cation beds containing 40 ft^3 of organic resin each will enhance sodium removal and lead to substantial improvements in system performance. the second modification noted above has not been considered in detail in this section or Appendix G; however, it might add to the efficacy of the system without increased waste generation and is considered to be covered by this alternative. In addition, this alternative is conceived to cover any other small modifications that might be incorporated in the SDS process as a result of knowledge gained from initial operations, etc. that do not significantly change the types or amounts of generated wastes.

Other alternatives involve use of the SDS type filters and zeolite beds in series with other treatment systems. One alternative would process SDS zeolite bed effluent through an evaporator and the other would polish this effluent by processing through a second system, EPICOR II. Another variation would merely take SDS effluent after the mixed bed ion exchanger and process it through the EPICOR II system. The SDS followed by EPICOR II polishing, as well as the other zeolite-based systems, are considered adequate for treatment of RB sump water.

Primary System Water

The radionuclide concentration in this water is about a factor 3 lower than that of RB sump water, 60 μCi/mL, because of dilution by makeup water since the accident. The contaminants include almost equal quantities of cesium and strontium so particular care must be taken regarding strontium removal. Moreover, since this water provides cooling for the damaged fuel, a portion of it must be processed on a feed-and-bleed basis and kept adequately borated

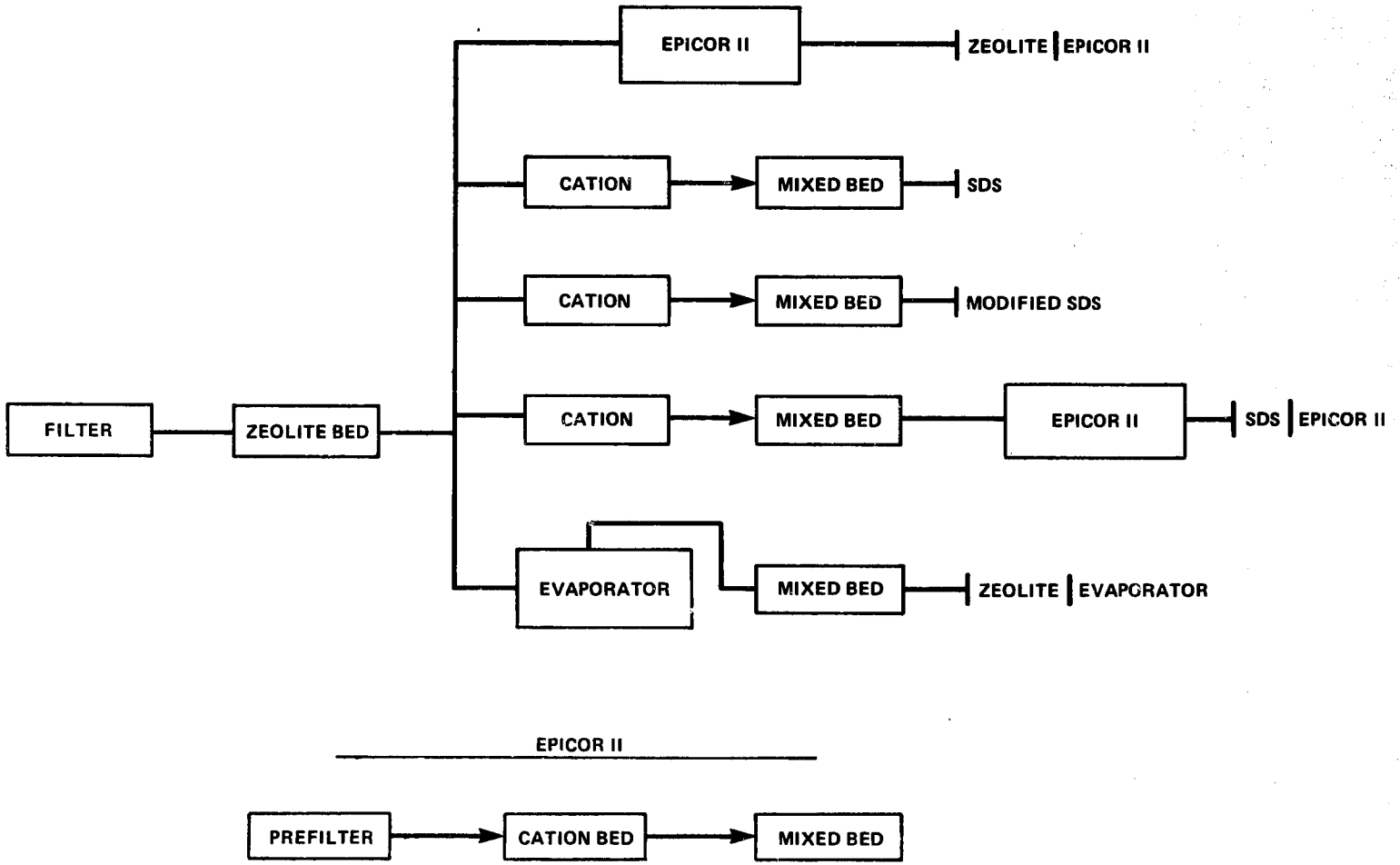

Figure 7.4 Alternative Treatment System Configuration for RB Sump Water.

This water contains sodium in concentrations comparable to those in RB sump water. Therefore SDS treatment of primary system water could be impeded for the same reasons as those described above under RB sump water. However, the modified SDS and other zeolite based systems are considered adequate for treatment of this water.

RCS Flush and Drain Water

This water is expected to have contaminants similar to those present in primary system water; however, the estimated radionuclide concentrations could range from 20 μCi/mL to 100 μCi/mL. Since the need to treat this water will not arise until after defueling when there is no longer a need to maintain boron levels, the system used to treat RB sump water or primary system water could also be used for treatment of these liquids.

RCS Decon Solutions (Water Based)

Use of the CAN-DECON Technique (see Appendix D) to decontaminate the reactor coolant system could generate about 100,000 gallons of liquid waste. This liquid is expected to contain between 2000 Ci and 20,000 Ci which results in radionuclide concentrations between 1 μCi/mL and 50 μCi/mL. A zeolite bed could be used to strip radioactive ions from the decontamination solution during the decontaminating phase. During the liquid cleanup phase, a zeolite bed in series with a mixed resin bed could be used to treat these liquids.

RCS and AFHB/RB Chemical Decontamination Solutions

These liquid wastes will contain relatively high concentrations of chemicals and detergents. Treatment of these liquids using ion exchange media will lead to chemical breakdown or plugging of the ion exchange media. Therefore, none of the ion-exchange treatment systems described above can be used to treat this waste. This waste is normally treated by an evaporative process or immobilized directly in a solidification facility.

Processing Rates

All the above zeolite-based treatment systems have comparable throughputs, about 100,000 gallons per month. The time required to process this water will, however, depend on how the selected system is operated. For once-through treatment of the various liquids, the RB sump water could be processed in 7 months, the RCS flush and drain water in about 2½ months, and the RCS water based decon solutions in about 1 month. For the primary system water, only a portion of the 96,000-gallon inventory can be processed on a once-through basis because the reactor vessel and cooling circuits must remain filled and borated. Two alternatives are available for treatment of the primary system water. The minimum elapsed time results when about 70,000 gallons are drained from the primary system and processed on a once-through basis and then the remaining 26,000 gallons used for fuel cooling and boration are processed on a feed-and-bleed basis. This approach leads to processing about 300,000 gallons over a period of about 3 months. If the entire inventory is processed on a feed-and-bleed basis, about 800,000 gallons would have to be processed over a period of about 8 months.

Facilities

The equipment that comprises the zeolite-based system used for cleanup of RB sump water and other high activity water that can be processed prior to defueling would be located in Spent Fuel Pool B and operated under water for shielding. Figure 7.5 shows a schematic of the SDS system when installed in the spent fuel pool. The layout shown is equally applicable to the other alternatives. As shown, filters, zeolite vessels, and cation vessels are submerged while the relatively large mixed bed vessel is operated within a shield cask above the spent fuel pool water surface.

Both spent fuel pools will have to be emptied of equipment prior to the defueling. If a zeolite-based system is selected for cleanup of RB sump water and is also used for cleanup of other water such as RCS flush and drain or RCS water-based decon solutions, the equipment will have to be relocated to existing onsite space or a new building.

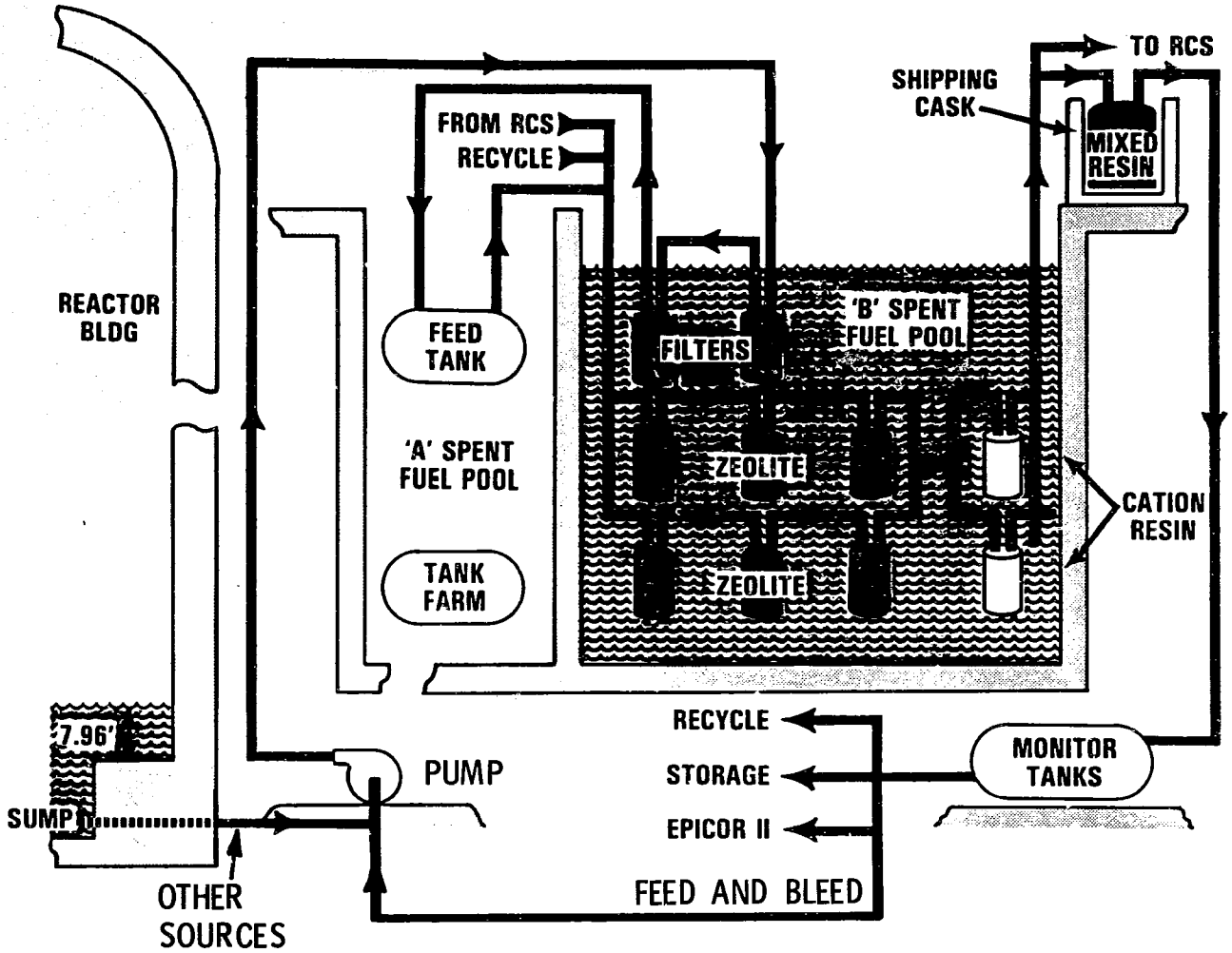

Figure 7.5. Schematic Diagram of TMI-2 SDS Radwaste Treatment System.

7.1.3.4 EPICOR-II and Modified EPICOR II Treatment Systems

This section provides a detailed discussion of the EPICOR II and modified EPICOR II treatment systems. The licensee has successfully operated the EPICOR II ion exchange system for cleanup of AFHB accident water. This system was designed to process liquid wastes with specific activities below 100 μCi/mL.

A process flow diagram for the EPICOR II system is presented in Figure 7.6. As shown, the influent is first transferred to a batch tank. For gross cesium and strontium removal, the waste is processed through an ion exchange vessel referred to as a prefilter liner. This vessel is about 4 ft in diameter and 4 ft high and contains about 30 ft^3 of inorganic and organic resin in layers designed to best clean up a particular batch of influent. Experience to date with processing of AFHB water shows that over 95 percent of the cesium and strontium in the influent is removed in the prefilter ion exchange vessel. The liquids are then processed through a cation exchange vessel for further removal of cesium, strontium, and other ions. This vessel is the same size as the prefilter and contains about 30 ft^3 of cation organic resin. Final polishing is accomplished by processing through a relatively large mixed bed ion exchange vessel. This vessel is 6 ft diameter by 6 ft high and contains about 115 ft^3 of anion and cation organic resins. When the ion exchange media are expended, each of the three vessels are removed from the system and replaced.

Performance of the EPICOR II system during cleanup of the AFHB accident water has exceeded design expectations. System performance for a typical batch of AFHB water is shown on Table G.11 in Appendix G. It is reasonable to expect comparable performance if the system is used to process

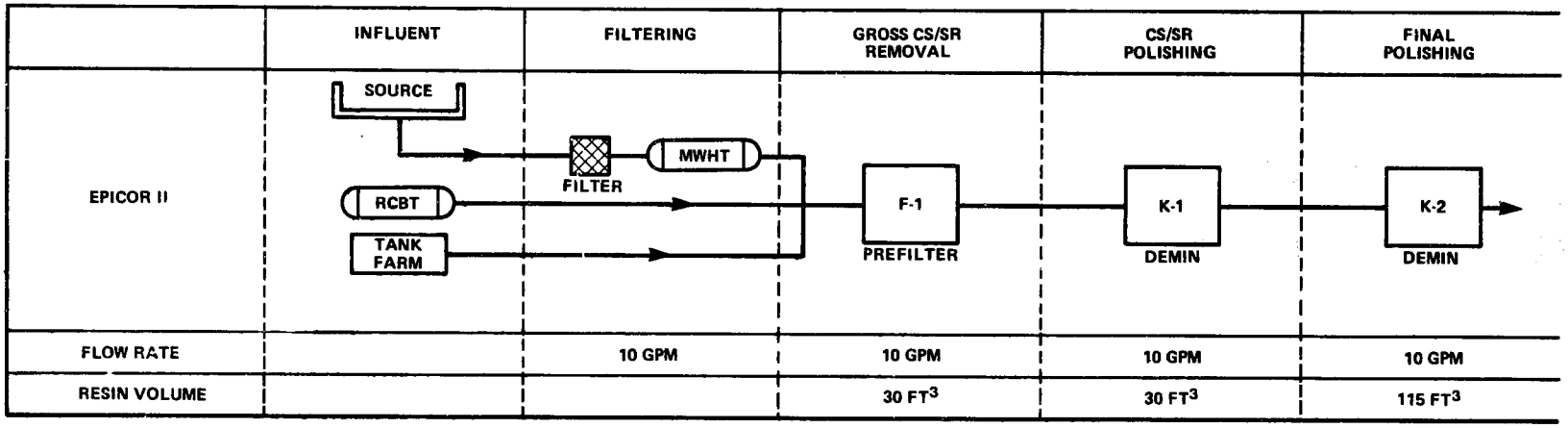

Figure 7.6. EPICOR II Process Diagram.

other liquids. The average throughput of the system has been about 2 gpm over the 16 months it has operated.

A modified version of EPICOR II is also being considered.[12,13] The modification would consist of replacing the first stage prefilter with a zeolite ion exchange vessel which would be loaded to higher radioactivity levels than the prefilters, resulting in corresponding reductions in solid waste generated. This vessel would have the same water processing characteristics as the first stage vessels used in the SDS. Since zeolites would be loaded to higher radioactivity levels than the expended prefilters, the zeolite bed would be operated within its own shield. The modified version of EPICOR II may bypass the mixed bed ion exchanger.

RB Sump Water

The radionuclide concentrations in this water are a factor of 2 greater than the design basis for treatment with EPICOR II or the modified EPICOR II systems. Therefore, neither system is suitable for treatment of this water.

RCS Water

Primary system water, RCS flush and drain water, and RCS water-based decon water have, or are expected to have, specific activities below the 100 μCi/mL design limit for EPICOR II. Therefore, EPICOR II and the modified EPICOR II systems are suitable for treatment of these liquids. The EPICOR II system generates relatively large volumes of organic resins compared to the other systems which use higher loaded zeolites for gross removal of contaminants in the first stage. Moreover, these first stage organic resins would be highly loaded which could lead to potential solid waste storage and disposition problems as discussed in Section 8.1. The modified EPICOR II system eliminates the first stage organic resins, reduces the volume of generated solid waste, and alleviates the concerns associated with the extended storage of higher specific activity organic resins (i.e., greater than 10 Ci/ft^3).

RCS and AFHB/RB Chemical Decontamination Solutions

The chemicals and detergents in these liquids would lead to chemical decomposition of the ion exchange media. Therefore, neither EPICOR II nor the modified EPICOR II system are considered suitable for treatment of these wastes. These wastes are normally treated with an evaporative process or immobilized directly in a solidification facility.

Processing Rates

The EPICOR II system was designed for a throughput of 10 gpm but the time averaged throughput (including outages) has been about 2 gpm. Similar performance can be expected for the modified EPICOR II system. This throughput corresponds to a processing rate of about 80,000 gallons per month. Once-through treatment of RCS flush and drain water will take about 3 months. Treatment of the RCS water-based decon water will require an additional 1½ months. As indicated in Section 7.1.3.3 above, the entire primary system inventory cannot be processed on a once-through basis. The minimum time required to process this water would be about 4 months. Feed-and-bleed processing of the entire inventory would take about 10 months.

Facilities

If either the EPICOR II system or a modifed version of it is selected for treatment of RCS water, the equipment will remain in its present location. A schematic of the system in its present locale is presented on Figure 7.7. As shown, each ion exchange vessel is operated within a shield and, when expended, is removed using a transfer shield. If the prefilter is replaced by a zeolite ion exchange vessel, a second shield will be placed in the existing prefilter shield to maintain radiation levels at their current levels during operation of a modified EPICOR II system. Additional shielding will also be required for the transfer shield.

7.1.3.5 Evaporator/Resin System

The licensee initiated the process of designing an evaporator/solidification facility to treat liquid wastes and immobilize evaporator bottoms. This activity has been deferred but is still considered an alternative.

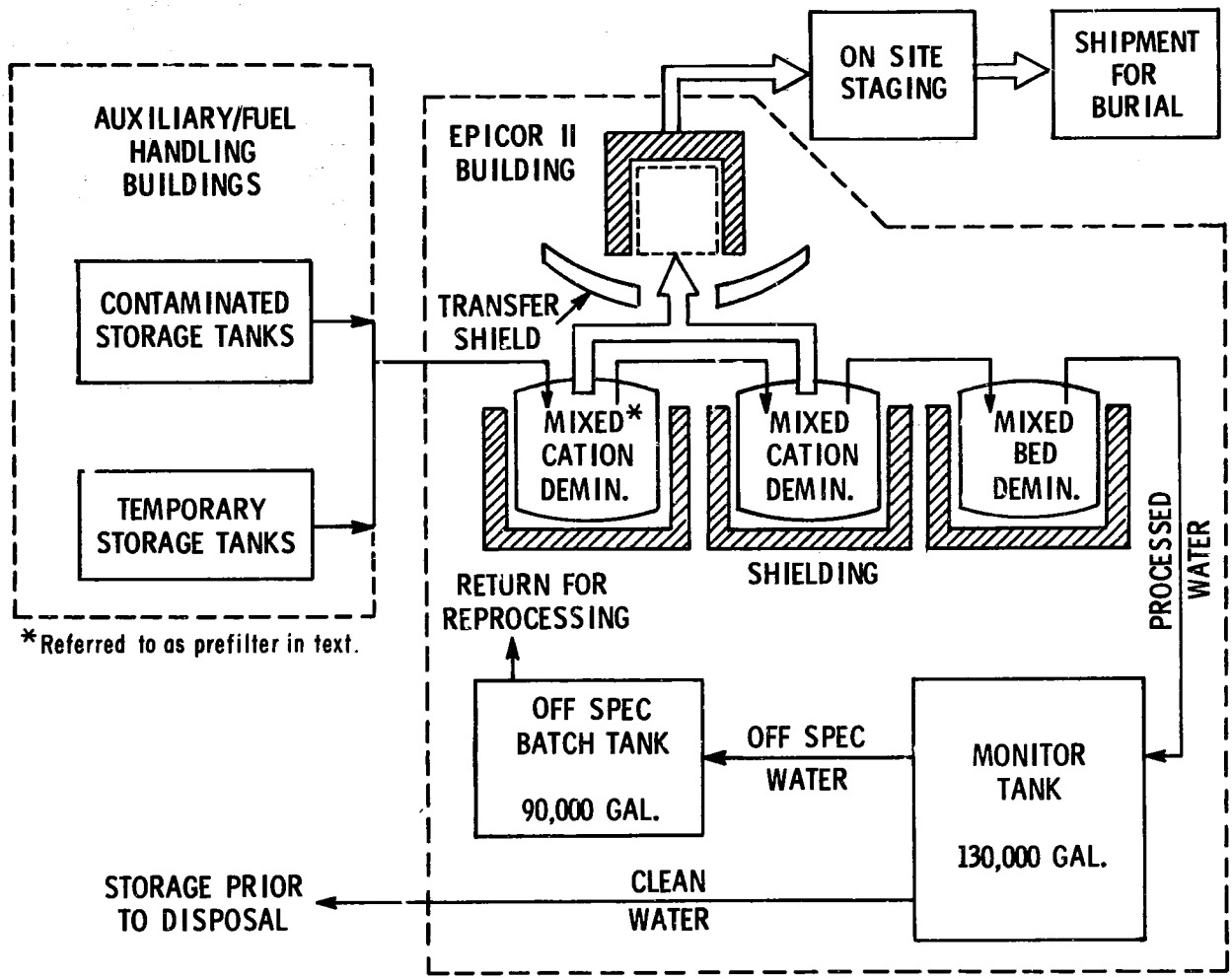

Figure 7.7. TMI-2 EPICOR II Radwaste System.

A process flow diagram for a typical evaporator/resin system is shown on Figure 7.8. As shown, influent is transferred to a batch tank for pretreatment (e.g., pH adjustment). Following this conditioning step, it is transferred to the evaporator. The feed rate can vary from 10 to 30 gpm for the applications considered for TMI liquids. The contaminants are concentrated in the evaporator bottoms which are transferred from the evaporator for immobilization and disposed of as solid waste. The Volume Reduction Factor (VRF) can vary between 10 and 100, dependent on the characteristics of the influent. A VRF of 100 is representative of water based liquids while the relatively strong chemicals that could be generated from decontamination activities would have a VRF of 10 to 30. Thus, the radioactivity concentration in the bottoms could be 10 to 100 times greater than that in the influent.

The vaporized water referred to as a distillate is condensed. The condensed water is then processed through demineralizers for removal of soluble contaminants. A cation demineralizer vessel and a mixed bed demineralizer may be required. Use of a 30-gpm evaporator/resin system for treatment of TMI-2 liquids is discussed below.

RB Sump Water

This water has radionuclide concentrations of about 190 μCi/mL which when processed will lead to evaporator bottoms with concentrations in the 2,000 μCi/mL to 20,000 μCi/mL range. These high concentrations will lead to extremely high radiation fields in and around the evaporator facility. Industry experience with processing even relatively low concentration liquid waste, 0.1 to 10 μCi/mL, in similar evaporators has indicated potential problems.[14] Most systems have not been reliable and outage maintenance has been excessive relative to other types of treatment systems (e.g., ion exchange).[15] Some nuclear facilities have started up their evaporators, encountered

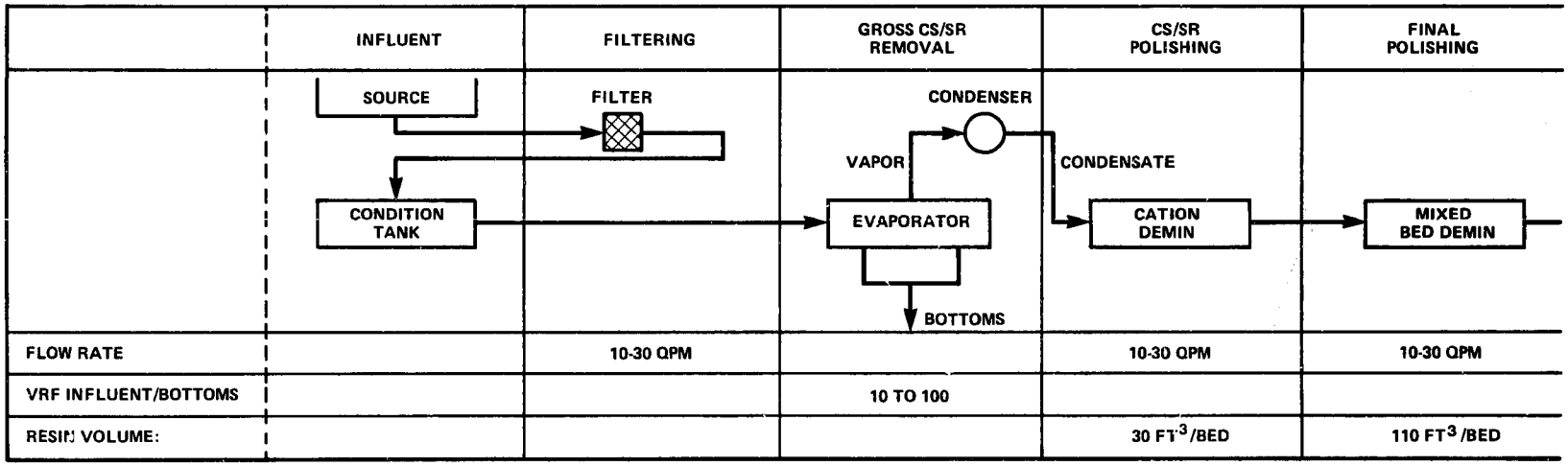

Figure 7.8. Evaporator/Resin Process Diagram.

problems with their operation and maintenance, and then shut them down permanently.[16] When evaporator experience to date is considered in conjunction with the highly concentrated evaporator bottoms that would arise from processing RB sump water, personnel exposure during outage maintenance would be excessive.

Additionally, evaporator bottoms with radionuclide concentrations in the 2,000 μCi/mL to 20,000 μCi/mL range when immobilized in a solidification facility, are well above the levels considered suitable for shallow land burial disposal. Thus, these materials would have to be handled as high specific activity wastes. The presence of borates and other precipitates in the bottoms and their liquid form also reduce the options available for their disposition as high specific activity waste. Neither government nor commercial facilities have produced or treated high specific activity wastes, from evaporators, with similar characteristics.

Therefore, for all of the above reasons, an evaporator is not considered suitable for first stage treatment of RB sump water.

RCS Water

Primary system water (∿ 60 μCi/mL), RCS flush and drain water (20 μCi/mL to 100 μCi/mL) and RCS water-based decon solutions (5 μCi/mL to 50 μCi/mL) all have relatively high radionuclide concentrations compared to the liquids that have been treated to date using evaporators. Thus the above discussion for RB sump water is applicable to this water and evaporators are not considered suitable for treatment of this waste.

Chemical Decontamination Solutions

The chemical solutions that could be generated from AFHB/RB cleanup and RCS decontamination are expected to have relatively low radionuclide concentrations, less than 1 μCi/mL to 10 μCi/mL. Wastes with similar concentrations and characteristics have been treated with evaporators. Therefore, evaporation is considered a suitable alternative for treatment of these wastes.

Processing Rates

A 30-gpm capacity unit is expected to have a time averaged throughput (which includes outages) of 3 to 6 gpm. The design to average throughput ratio (6 to 10) is higher than that considered for ion exchange systems (3 to 5) because of the more frequent maintenance outages expected with the evaporator. The anticipated volume reduction factor for the solutions that could be processed would range from about 10 to 30. Thus the radionuclide concentrations in the evaporator bottoms could range from 10 μCi/mL to 300 μCi/mL.

The time required to process 500,000 gallons of decon solutions at 3 gpm would be about four months. The higher throughput of 6 gpm would reduce this to about 2 months.

7.1.3.6 Bitumen/Resin System

The bitumen/resin system described in this section is based on a design which employs one step evaporation/solidification to process TMI-2 radiactive liquid waste.[17,18] The basic component of the system is a twin screw extruder/evaporator. The screws rotate inside heavy metal, steam jacketed barrels. The system combines a volume reduction and solidification process by simultaneously feeding measured quantities of liquid waste and hot asphalt to the extruder/evaporator which evaporates the free water, mixes the remaining radioactive solids with asphalt, and dumps the mixture into a waiting 55 gallon drum on a turntable. The mixture typically consists of 50 wt% radioactive solids in asphalt. The extruder/evaporator is equipped with steam domes with integral condensers and the distillate is directed to a downstream distillate collection and oil filtration system prior to polishing in cation and mixed bed demineralizer vessels. A process flow diagram of the bitumen/resin system is shown in Figure 7.9.

The bitumen/resin system is a relatively low capacity system, capable of processing liquid waste at approximately 0.5 gpm. This evaluation assumes that two systems would be installed to provide a combined flow capacity of 1 gpm. Additionally, the radiation stability of the bitumen/radwaste product is adequate to only about 10^9 rads. Thus, administrative controls would have to be imposed on waste influent to the system, taking into account the concentration of the waste and anticipated onsite storage to ensure the integrity of the product and its container prior to disposal.

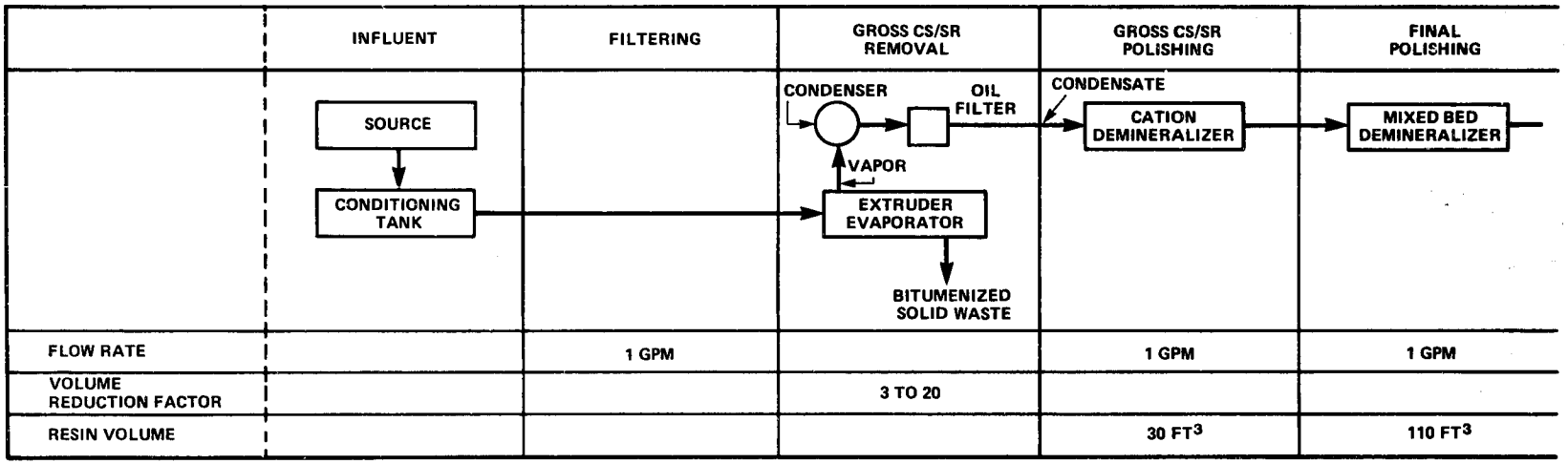

Figure 7.9. Bitumen/Resin Process Diagram.

The staff is well aware that given the low capacity of the system, the application of a bitumen/ resin system in a commercial nuclear power plant is usually associated with the use of a preconcentrator (e.g., crystallizer evaporator). However, given the higher than normal activity concentrations and/or anticipated solids content of the TMI-2 liquid waste (see Table 7.1), the use of a preconcentrator was judged to be neither necessary nor desirable. The use of a redundant (i.e., 1 gpm capacity) bitumen/resin system for treatment of TMI-2 liquid wastes is discussed below.

RB Sump Water

This water has a concentration of approximately 190 μCi/mL (5.3 Ci/ft^3) and an estimated solids content considerably less than 10%. Volume reduction of this water in the bitumen/ resin system would result in a waste product with an activity concentration considerably in excess of that which is considered suitable for organic material (i.e., up to 10 Ci/ft^3). Therefore, this system is considered unsuitable for RB sump water given the uncertainties associated with ultimate disposition and near-term (on the order of several years) stability of the waste product. Radiation damage to bitumen wastes occurs at approximately 10^9 rads exposure and is characterized by product swelling as a result of the generation of gases such as hydrogen, methane, carbon dioxide and carbon monoxide.

RCS Water

Primary system water has a concentration of approximately 60 μCi/mL (1.8 Ci/ft^3) and a solids content considerably less than 10%. Volume reduction of this water in the bitumen/ resin system would be unsuitable for the same reasons as for the RB sump water.

AFHB/RB Chemical Decontamination Solutions

The chemical solutions that are expected to be generated from decontamination activities in the AFHB and RB are expected to have relatively low activity concentrations, less than 1 μCi/mL to 10 μCi/mL, and relatively high solids content, well in excess of 10%. This waste is considered particularly well suited for processing in a bitumen/resin system. Additionally, the total estimated waste volume (47,000 gallons) is not an unreasonable quantity for a 1 gpm system. The bitumen/resin system is a high reliability system and it should not take more than several months to process this waste. Given the high solids content of the waste, the staff estimates that the volume reduction factor for this waste would be in the range of 3 to 20.

RB Water Based Decontamination Solutions

The water-based decontamination solutions that are expected to be generated from activities in the reactor building are expected to be mixed with RB sump water prior to processing. Thus, for the reasons cited above, regarding RB sump water processing, the bitumen/resin system is considered unsuitable for processing this water. In the event that these decontamination solutions are not mixed with sump water but are collected separately, the bitumen/resin system would be adequate for processing this waste.

RCS Flush and Drain Water

The radionuclide concentrations in the RCS flush and drain water is expected to range from 20 μCi/mL to 100 μCi/mL. This is comparable to the RCS water concentration. Therefore, the bitumen/resin system is not considered suitable for treatment of this water.

RCS Decon Solutions

If the CAN DECON technique is used, the radionuclide concentrations are expected to be in the 5 μCi/mL to 50 μCi/mL range. At 50 μCi/mL, these concentrations are too high for treatment of these liquids with a bitumen/resin system. If one of the other chemical techniques is used the expected concentrations are about a factor of 5 lower i.e., 1 μCi/mL to 10 μCi/mL. Chemical solutions with concentrations in this lower range are suitable for treatment with a bitumen/resin system.

7.1.3.7 Summary of Liquid Waste Treatment Alternatives

Table 7.9 shows the alternatives considered for treatment of liquid wastes. A "*" on the table indicates the alternative is considered suitable for treatment of the liquid waste shown. The

Table 7.9. Treatment Alternatives Suitable for TMI Liquids[a]

Treatment System Alternative	(A) AFHB Chemical Decontamination Solutions	(B) Reactor Building Sump Water	(C) Reactor Building Decontamination Water	(D) Reactor Building Decontamination Chemicals[c]	(E) Reactor Coolant System: Cooling Water[d]	(F) Reactor Coolant System: Flush and Drain Water[d]	(G) Reactor Coolant System: Decontamination Water[e]	(H) Reactor Coolant System: Decontamination Chemicals[c]
	Type of Liquid to be Processed							
1. Zeolite Alternatives								
(a) Zeolite/resin (SDS)		*	*		*	*	*	
(b) Zeolite modified resin (Mod SDS)		*	*		*	*	*	
(c) Zeolite/evaporator		*	*		*	*	*	
(d) Zeolite/EPICOR II		*	*		*	*	*	
(e) SDS/EPICOR II		*	*		*	*	*	
(f) Modified EPICOR II					*	*	*	
2. EPICOR II			(f)		*	*		
3. Evaporator/resin	*			*				*
4. Bitumen/resin	*			*				*

[a]Note: * indicates system to be discussed as an alternative method.

[b]Alternative in the event this water is not processed along with reactor building sump water.

[c]Chemical properties of these liquids are not compatible with purely ion-exchange alternatives.

[d]Ion-exchange processes provide adequate alternatives.

[e]Processing of CAN DECON decontamination solutions will probably require system alignment changes.

[f]Only if reactor building decon water collected separately from the reactor building sump water.

alternatives indicated by a "*" have advantages and disadvantages when compared to each other on a relative basis. The major parameters considered to compare alternatives are listed in Table 7.10 and discussed below.

Time to Install a System

The time required to place a system into operation depends on many factors. Some systems require the design, construction and checkout of equipment and facilities (including new buildings), while others do not. The relative time spans among alternatives are indicative of when cleanup of accident water could begin.

Time to Implement

Once the facilities and equipment needed to treat liquid wastes are operational, the performance characteristics of the process determines how long it will take to treat the liquid waste. The processing time is also an indication of the level of manpower that would be required to process the water.

Liquid Effluents Produced

Some systems produce a liquid effluent. The inventory of radionuclides in these effluents is a measure of relative treatment system performance. After processing, these effluents must be stored onsite for reuse or ultimate disposition. Whether liquid effluents satisfy the criteria for outside storage is another measure of relative treatment system performance.

Solid Wastes Produced

Each system also produces solid wastes, some of which require further processing and handling before disposal. The characteristics of these solids determine the extent of additional processing and handling needed and in some cases limit the options available for storage and ultimate disposal. Thus, the characteristics of the resultant solid wastes and their ability to be handled and readily disposed of is another measure of relative system performance.

7.1.4 Effluents and Releases to the Environment

7.1.4.1 Normal Operations

Effluents and releases during treatment of the liquid wastes described above for the processes considered are presented in this section. Releases attributable to airborne material (aerosol) resulting from liquid processing are shown in Table 7.11. Releases are shown in two ways: (1) the total release for an operation, expressed in curies, and (2) the concentration of the release, expressed as microcuries per milliliter (μCi/mL), of gaseous effluent. The bases for the calculation of releases are given below.

As a result of various processing operations involving liquid materials, small quantities of the liquids become airborne and enter the vapor spaces above the liquid. The composition of such aerosols is identical (at the time of formation) to that of the liquid from which it was derived. The liquid in the aerosol evaporates, leaving suspended residual solids. The airborne material is carried by subsequent movement of air in process vent systems, then ultimately combined with building ventilation air, treated by the air cleaning system for the building, monitored by radiation detectors, and finally released to the environment by controlled discharge. If necessary, releases to the environment can be stopped by shutting down the ventilation system. While in transit larger particles settle out rapidly, while very small particles remain suspended and tend to agglomerate with each other to form larger particles. The air cleaning system includes one or more stages of HEPA filters which remove the bulk of the solid airborne material. Volatile materials, such as tritiated water vapor are not affected by the air cleaning system.

The gaseous effluents and releases from these operations attributable to the formation of aerosols are dependent on four principal factors: (1) the concentration of the radionuclides in the liquid being treated, (2) the processing rate, (3) the fraction of the material processed that becomes aerosolized, and (4) the fraction of such materials that passes through the air cleaning system uncontrolled.

To calculate effluents and releases, the staff used a throughput of 10 gallons per minute, the design basis. The time averaged throughput, i.e., gallons processed over several months, will be

Table 7.10. Comparison of Treatment Systems

Alternative	Parameter							
	Time		Liquid Effluents			High-Specific-Activity Waste		
	To Install System	To Process 100,000 Gallons	Produced	Suitable for Outside Storage	Relative Quality Based on Curie Content	Produced	Stable During Extended Storage	Suitable for Disposable at Routine Shallow Land Burial Facilities
Direct immobilization[a]	3 yrs	3 mos	No	NA	NA	No	NA	NA
Long term storage	3 yrs	1 wk	NA	NA	NA	NA	NA	NA
Evaporator/resin system[a]	3 yrs	1 mo	Yes	Yes	High	No	NA	NA
Bitumen/resin system[a]	3 yrs	4 mo	Yes	Yes	High	No	NA	NA
Zeolite based systems[b]								
SDS	3 mos	1 mo	Yes	Yes[e]	Low	Yes	Yes	No
Modified SDS	6 mos	1 mo	Yes	Yes	Low	Yes	Yes	No
Zeolite/Evaporator	3 yrs	1 mo	Yes	Yes	High	Yes	Yes	No
Zeolite/EPICOR II	6 mos	1.5 mos	Yes	Yes	High	Yes	Yes	No
SDS/EPICOR II	3 mos	1.5 mos	Yes	Yes	High	Yes	Yes	No
Modified EPICOR II[c]	3 mos	1.5 mos	Yes	Yes	High	Yes	Yes	No
EPICOR II System[c]	0	1.5 mos	Yes	Yes	High	Yes	No	No[d]

[a]Considered suitable only for treatment of chemical decontamination solutions. Table entries apply only to this waste type.

[b]Considered suitable only for treatment of waste water, not chemical decontamination solutions. Table entries apply only to these waste types.

[c]Considered suitable for all sources of waste water except reactor building sump water. Table entries apply only to these waste types.

[d]Refer to discussions in Sections 8 and 9.

[e]Dependent upon radionuclide distribution, see Section 7.2.4.2.

Table 7.11. Gaseous Effluents and Releases Attributable to Aerosol Formation from Processing of Various Liquids (Normal Operations)

Nuclide	Identification of Process Liquid						
	RB Sump	RB Decon Water	Decon Chemicals[a]	RCS Primary Water	RCS Flush and Drain	RCS Decon Water	RCS Decon Chemicals[a]
Normal Releases, Total Curies							
H-3	2.5×10^{-1}	b	*	2.7×10^{-3}	*	*	*
Cs-137	4.1×10^{-2}	6.4×10^{-6}	1.2×10^{-5}	9.9×10^{-4}	4.9×10^{-3}	9.9×10^{-4}	9.9×10^{-4}
Cs-134	6.4×10^{-3}	9.8×10^{-7}	1.8×10^{-6}	1.5×10^{-4}	7.7×10^{-4}	1.5×10^{-4}	1.5×10^{-4}
Sr-90	6.9×10^{-4}	7.6×10^{-8}	1.4×10^{-7}	7.8×10^{-4}	3.9×10^{-3}	7.8×10^{-4}	7.8×10^{-4}
Sr-89	1.9×10^{-5}	3.0×10^{-9}	5.6×10^{-9}	8.2×10^{-5}	4.1×10^{-4}	8.2×10^{-5}	8.2×10^{-5}
Normal Release, Concentration, μCi/mL							
H-3	1.1×10^{-8}	*	*	1.1×10^{-9}	*	*	*
Cs-137	1.9×10^{-9}	2.2×10^{-12}	1.7×10^{-11}	3.9×10^{-10}	7.0×10^{-10}	3.5×10^{-10}	7.0×10^{-11}
Cs-134	2.8×10^{-10}	3.4×10^{-13}	2.7×10^{-12}	6.0×10^{-11}	1.1×10^{-10}	5.4×10^{-11}	1.0×10^{-11}
Sr-90	3.0×10^{-11}	2.7×10^{-14}	2.0×10^{-13}	3.1×10^{-10}	5.5×10^{-10}	2.8×10^{-10}	5.5×10^{-11}
Sr-89	8.2×10^{-13}	1.0×10^{-15}	8.3×10^{-15}	3.2×10^{-11}	5.8×10^{-11}	2.9×10^{-11}	5.8×10^{-12}

[a]Chemical solutions are processed through evaporator/resin or bitumen/resin system.

[b]Note: * indicates that if processed water is recycled for these applications, tritium concentrations will approximate that in processed water used.

about one-fourth of this value. This rate is typical of that expected to be used for the operation considered and is based on operating experience to date with EPICOR II.[19] A conservative value of 1×10^{-4} has been used as the fraction of material processed that is aerosolized and ultimately reaches the air cleaning system. For the processing operations considered, the actual value is expected to be much less. These statements are based on the operating experience with EPICOR-II and the experience accumulated in the processing of irradiated nuclear fuels. The processing of nuclear fuels consists of a large number of complex operations that include fuel dissolution, solvent extraction, and evaporation. The systems considered for treatment of TMI-2 liquids are, by contrast, less complex and involve only a few process steps that are simple in nature and thus should result in considerably less aerosolized material.

The efficiency of the air cleaning system for removing solid airborne materials (aerosols) is assumed to be 99.9% (stated alternatively, this corresponds to a penetration of aerosols through the system of 0.1%, or 0.001). Actual penetration values are expected to be much lower based on the theory of operation of air cleaning systems and based on operating experience. The key element in the cleaning system for removal of airborne materials is the HEPA filter. At TMI, two stages of HEPAs in series are present in the cleaning system. The principal characteristic of the HEPA filter is that it is very efficient for collecting very small and very large particles. The specification for the removal efficiency of each installed HEPA filter for 0.3 μm particle sizes is 99.97% (corresponding to a penetration fraction of 0.0003). This efficiency is demonstrated to be within specification by suitable in-place testing. A second HEPA filter in series with the first will remove 99.97% of the material passing through the first HEPA. Thus, it is evident that the assumed 99.9% efficiency for the system is conservative and readily achievable.[20]

To calculate the release concentrations, expressed as μCi/mL of gaseous effluent, the ventilation flow rate passing through the air cleaning system must be known. Because the locations of all processing operations considered in this section are not known at this time, the specific appropriate ventilation rates for these calculations also are not known. Therefore, all calculations have been normalized to a ventilation flow rate of 10,000 cfm. If, for example, actual ventilation rates are doubled (i.e., 20,000 cfm), the calculated concentrations would be one-half of those shown in Table 7.11. This assumption has no effect on the total releases, expressed as curies.

7.1.4.2 Accident Conditions

The failure of a HEPA filter is considered a credible accident event (i.e., an accident that could occur). The operation of the air cleaning system is constantly monitored, and a HEPA filter failure would be detectable in a matter of minutes. However, it is assumed here that the accident conditions would exist for a period of 15 minutes before remedial action is taken.

Experience has shown that when a HEPA filter fails, a fraction of its inventory of previously filtered materials would be released. A representative value of 0.001 has been used for this fraction. The releases calculated to result from such an accident are shown in Table 7.12.

7.1.5 Environmental Impacts

7.1.5.1 Occupational Doses

AFHB Water Processing (EPICOR II) Experience[9]

The EPICOR II system is designed for use in decontaminating radioactive water with minimal personnel radiation exposure. The EPICOR II operations which cause occupational radiation exposure are: operation and sampling (about 20 percent of the occupational dose), liner changeout (about 40 percent), and maintenance and outages (about 40 percent). Liners have been placed in an interim onsite storage facility in a dewatered condition; none of the liner contents have been immobilized to date.

In general, there has been a decreasing trend in the occupational doses incurred during the processing of AFHB water through the EPICOR II system. For the first batches handled, only about 14,000 gallons of water were processed per person-rem of occupational dose. As of May 5, 1980, 225,000 gallons had been processed through EPICOR II, with a cumulative occupational dose of 8.4 person-rem, equivalent to about 30,000 gallons of water processed per person-rem of occupational dose.[7] As of June 8, 1980, about 331,000 gallons had been processed with a cumulative occupational dose of 11.4 person-rem,[8] corresponding to 29,000 gallons processed per person-rem, mated unit maintenance dose of about 0.003 person-mrem per gallon processed. On this basis, the total maintenance cumulative dose is about 2 person-rem.

Table 7.12. Gaseous Effluents and Releases from Failure of a HEPA Filter during Processing of Various Liquids

Nuclide	Identification of Process Liquid						
	RB Sump	RB Decon Water	Decon Chemicals	RCS Primary Water	RCS Flush and Drain	RCS Decon Water	RCS Decon Chemicals
Accident Releases, Total Curies							
Cs-137	4.3×10^{-2}	6.4×10^{-6}	1.2×10^{-5}	9.9×10^{-4}	4.9×10^{-3}	9.9×10^{-4}	9.9×10^{-4}
Cs-134	6.4×10^{-3}	9.8×10^{-7}	1.8×10^{-6}	1.5×10^{-4}	7.7×10^{-4}	1.5×10^{-4}	1.5×10^{-4}
Sr-90	6.9×10^{-4}	7.6×10^{-8}	1.4×10^{-7}	7.8×10^{-4}	3.9×10^{-3}	8.0×10^{-4}	7.8×10^{-4}
Sr-89	1.9×10^{-5}	3.0×10^{-9}	5.6×10^{-9}	8.2×10^{-5}	4.1×10^{-4}	8.4×10^{-5}	8.2×10^{-5}
Accident Release, Concentration, μCi/mL							
Cs-137	1.0×10^{-5}	1.5×10^{-9}	2.8×10^{-9}	2.3×10^{-7}	1.2×10^{-6}	2.3×10^{-7}	2.3×10^{-7}
Cs-134	1.5×10^{-6}	2.3×10^{-10}	4.3×10^{-10}	3.6×10^{-8}	1.8×10^{-7}	3.6×10^{-8}	3.5×10^{-8}
Sr-90	1.6×10^{-7}	1.8×10^{-11}	3.2×10^{-11}	1.8×10^{-7}	9.2×10^{-7}	1.9×10^{-7}	1.8×10^{-7}
Sr-89	4.4×10^{-9}	7.1×10^{-13}	1.3×10^{-12}	1.9×10^{-8}	9.6×10^{-8}	2.0×10^{-8}	1.9×10^{-8}

still a marked improvement over the dose incurred for initial batch processed. As of August 1980, the bulk of the AFHB water had been processed. About 55,000 Ci had been removed from about 500,000 gallons of water. The cumulative occupational dose incurred was a little less than 15 person-rem.

The expected number of additional cancer mortalities in the work force exposed to a cumulative dose of 15 person-rem would be 0.002. The expected number of additional genetic effects in the offspring of the work force exposed to this cumulative dose of radiation would be 0.004.

This EPICOR II experience is relevant to the exposure that could be received by personnel operating similar systems. Table 7.13 summarizes this experience in terms of various unit exposures.

Table 7.13. Summary of EPICOR II System Occupational Exposure Experience

Operational Authority	Total Cumulative Dose Received (person-rem)	Unit Dose
1. Operation and Sampling	3	6.0×10^{-6} person-rem per gallon processed.
2. Liner Changeout	6	8.5×10^{-2} person-rem per liner removed.
3. Maintenance and Outages	6	1.2×10^{-5} person-rem per gallon processed.
	15[a]	

[a]Overall average performance shows 3×10^{-5} person-rem per gallon processed and about 2.7×10^{-4} person rem per Ci removed.

Reactor Building Sump Water

This water is assumed treated with ion exchange systems which may be submerged in the spent fuel pool. These systems include SDS, modified SDS, zeolite/evaporator, zeolite/EPICOR II, and SDS/EPICOR II (see Section 7.1.3.1). The system will be operated from a remote control console similar to the EPICOR II arrangement. Thus, the occupational exposure incident to sampling and operation will be comparable to EPICOR II. The average dose for these activities was about 0.006 person-mrem per gallon. Thus, cleanup of RB sump water could lead to cumulative doses of about 4.2 person-rem.

The dose due to liner changeout for an SDS type system should be below that experienced with EPICOR II since liners will be changed out underwater. The number of liners to be handled is about twice the number handled for EPICOR II because zeolite liners have to be sequenced. Based on fields a factor of 10 below those during EPICOR II liner changeout and twice the handling, the liner changeout cumulative dose should be a factor of 5 below the EPICOR II experience or about 1.2 person-rem.

Maintenance on the submerged systems will also be performed through the water shielding. However, the time required to perform this maintenance remotely will increase. Combining these two factors leads to an estimated unit maintenance dose of about 0.003 person-mrem per gallon processed. On this basis, the total maintenance cumulative dose is about 2 person-rem.

Based on the above assumptions, the total estimated cumulated dose for treatment of 700,000 gallons of RB sump water using a submerged ion exchange treatment system is about 7.4 person-rem.

If this water is further treated by polishing with a second system such as EPICOR II, the estimated additional dose would be about 3 person-rem for operation, about 0.5 person-rem for liner

changeout and about 0.5 person-rem for outages and maintenance for a total of 4 person-rem. These lower doses are due to the much lower activity water that would be processed through a second polishing system.

The sump water is expected to be processed over four quarters of the year. The cumulative dose range of 7.4 to 11.4 person-rem incurred over four quarters is equivalent to 1.9 to 2.9 person-rem per quarter. Based on an individual dose of 1 rem per quarter, a work force of 2 to 3 persons would be needed.

The expected number of additional cancer mortalities in the work force exposed to this cumulative radiation dose is less than 0.001. This means that the added probability that the average individual worker would die of cancer is about one in 1400. The expected number of additional genetic effects in the offspring of the work force exposed to this cumulative dose is about 0.002.

Reactor Coolant System and Flush/Drain Water

Since these two sources of liquid waste will probably be treated by the same system, they have been combined in this section. The total volume of liquid to be cleaned up is about 350,000 gallons containing between 40,000 Ci and 120,000 Ci. However, feed and bleed processing of the coolant system water will result in about 530,000 gallons being processed through the selected treatment system.

EPICOR II as well as a modified version of EPICOR II is being considered for treatment of this water. The most recent experience with EPICOR II indicates an average cumulative dose of 1 person-rem per 30,000 gallons processed. Based on this experience, treatment of the 530,000 gallons could result in about 18 person-rem. At the higher curie content, about three times the number of liners will have to be changed out, leading to about 14 person-rem of additional dose. Thus, use of these systems could lead to a cumulative dose between 18 and 32 person-rem.

If an SDS type ion exchange system is used for cleanup of this water, the system could be operated submerged for cleanup of coolant water but would have to be moved to a different locale for cleanup of flush and drain water.

Using assumptions similar to those described above for RB sump water, the estimated cumulative dose for processing 280,000 gallons of water containing 20,000 Ci is 3 person-rem. Estimates for subsequent cleanup of 250,000 gallons of RCS flush and drain water can be based on the EPICOR II summary in Table 7.13. To remove 20,000 Ci, the cumulative dose estimate is about 5 person-rem. Removal of 100,000 Ci will increase this to about 7 person-rem due to the increased number of zeolite liners to be changed out.

Therefore, the cumulative dose can range from a minimum of about 8 person-rem (40,000 Ci removed by an SDS type system) to a maximum of 32 person-rem (120,000 Ci removed by an EPICOR II system).

The staff assumes that the crew size for the processing of this water will be the same as that used for processing the water in the AFHB. At a rate of 23,000 gallons per week (based upon EPICOR II experience), processing of 530,000 gallons will take about 23 weeks.

The expected number of additional cancer mortalities in the work force of 16 persons receiving this cumulative dose of radiation (8 to 32 person-rem) ranges between 0.001 and 0.004. This means that the added probability that the average individual worker would die of cancer ranges from one in 4000 to one in 7600. The expected number of additional genetic effects in the offspring of the work force receiving this cumulative radiation dose ranges between 0.002 and 0.008.

Chemical Decontamination Liquids

The relatively low volume of AFHB/RB chemical decontamination solutions could be directly immobilized. The dose estimates for immobilizing these liquids are presented in Section 8.2.

The relatively large volume of chemical solutions that could be generated from RCS decontamination with one of the chemical techniques could be treated with either an evaporator/ resin or a bitumen/ resin system. This 500,000 gallons of chemical solution could contain between 2,000 Ci and 20,000 Ci and the estimated doses for removing 20,000 Ci are presented below.

Evaporator Resin Alternative. The average throughput of an evaporator that could be used would range from 3 gpm to 5 gpm. The unit would be operated from a remote control console so that the EPICOR II experience is relevant. This experience indicated that 0.006 person-rem was received per gallon. Thus, for the lower throughput, cumulative personnel dose for these operations would be about 2.5 person-rem. For the higher throughput of 5 gpm, this would be reduced to about 1.5 person-rem. The dose arising from resin vessel changeout will be lower than that experienced with EPICOR II since the ion exchange media is only used for polishing. For EPICOR II, about 0.085 person-rem was received per liner changed-out. Processing of 500,000 gallons would require changeout of about 5 resin liners. Conservatively using the EPICOR II experience leads to a cumulative dose of about 0.5 person-rem. On the other hand, maintenance on an evaporator will result in higher dose than EPICOR II experience. For EPICOR II, this cumulative dose was about 1.2×10^{-5} person-rem per gallon. For the evaporator, maintenance exposure is estimated to be a factor of 5 to 10 higher than this, or 6.0×10^{-5} to 1.2×15^{-4} person-rem per gallon. At 500,000 gallons, the maintenance dose could range from 30 person-rem to 60 person-rem. Thus, the total cumulative dose for processing 500,000 gallons could range from a minimum of about 32 person-rem to a maximum of about 63 person-rem. If the activity of the water is 1 μCi/mL instead of 10 μCi/mL, this dose could be reduced by as much as 50 percent.

Bitumen/Resin Alternative. The bitumen/resin system is a high reliability unit and the time averaged throughput will approximate the design throughput of 1 gpm. The unit would be operated from a remote control console so the EPICOR II experience of 0.006 person-rem per gallon is relevant. For a throughput of 1 gpm, the average cumulative dose would be twice the EPICOR II figure or 0.012 per gallon--6 person-rem for 500,000 gallons. Since this unit uses ion exchange vessels to polish distillate, the number of vessels to be removed will be comparable to those required with the evaporator resin unit described above. Conservatively using the EPICOR II experience figure of 0.085 person-rem per liner, the dose incident to changeout will be about 0.5 person-rem.

The extruder-evaporator units require a minimum amount of maintenance compared to an evaporator/resin system. The lower throughput of two units, 1 gpm, is compensated for by less outage so the maintenance requirements for these units would be comparable of those experienced by the 2-gpm EPICOR II system, 1.2×10^{-5} person-rem per gallon. At this dose rate, processing approximately 500,000 gallons will result in a maintenance outage dose of about 6 person-rem. Thus, the total estimated cumulative dose to process 500,000 gallons is about 12.5 person-rem.

Summary. If RCS chemical decontamination solutions are treated with an evaporator/resin system, the estimated occupational cumulative dose could range from 32 to 63 person-rem. Processing the same liquid waste through two bitumen/resin systems could result in an occupational cumulative dose of about 12.5 person-rem. Thus, the estimated cumulative dose incident to processing RCS chemical decontamination solutions could vary from 12.5 person-rem over about 11 months to a maximum of 63 person-rem over about 4 months.

The estimated cumulative dose for cleanup of RCS chemical solutions, if they are generated, ranges from a minimum of 13 person-rem to a maximum of 63 person-rem. For the evaporator/resin, 64 person-rem will be received over about 5 months. For the bitumen/resin system, 13 person-rem will be received over about 25 months.

The expected number of additional cancer mortalities in the work force receiving to this cumulative radiation dose would range from 0.002 to 0.008. The expected number of additional genetic effects in the offspring of the work force receiving this cumulative dose would range from 0.0034 to 0.016.

7.1.5.2 Offsite Doses

The dose estimates presented here for processing of accident and decontamination water are based on the source terms developed in Section 7.1.4.1 and listed in Table 7.11. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3. The dose estimates to the maximum exposed individual for processing the water are listed in Table 7.14. The 50-mile total body population dose associated with the processing was estimated to be about 2 person-rems.

Table 7.14. Dose Estimates for the Maximum Exposed Individual due to Processing of Accident and Decontamination Water

Location	Pathway	Dose (mrem)[a] Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	2.9×10^{-3}	4.0×10^{-2}	2.5×10^{-3}
	Ground Shine	5.5×10^{-3}	5.5×10^{-3}	5.5×10^{-3}
	Vegetable Use	2.3×10^{-1}	9.6×10^{-1}	1.2×10^{-1}
	Total	2.4×10^{-1}	1.0	1.3×10^{-1}
Nearest milk goat	Inhalation	3.3×10^{-3}	1.5×10^{-2}	1.7×10^{-3}
	Ground Shine	5.3×10^{-3}	5.3×10^{-3}	5.3×10^{-3}
	Goat Milk Use	9.9×10^{-2}	7.2×10^{-1}	7.3×10^{-1}
	Total	1.1×10^{-1}	7.4×10^{-1}	7.4×10^{-1}
Nearest cow and garden	Inhalation	3.1×10^{-3}	4.5×10^{-2}	2.8×10^{-3}
	Ground Shine	8.1×10^{-3}	8.1×10^{-3}	8.1×10^{-3}
	Vegetable Use	3.4×10^{-1}	1.4	1.8×10^{-1}
	Cow Milk Use	3.8×10^{-2}	2.0×10^{-1}	1.4×10^{-1}
	Total	3.9×10^{-1}	1.7	3.3×10^{-1}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children and for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

7.1.5.3 Postulated Accident Effects

The type of accident for which dose estimates are presented is the rupture of a HEPA filter while processing accident or decontamination water. The accident scenarios are described in Section 7.1.4.3 along with their source terms. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses is discussed in Section 10.4. The estimated doses to the maximum exposed individual for each accident scenario are listed in the following tables:

Scenario	Table Number
Liquid Processing:	
RB Sump Accident Water	7.15
RB Decon. Water	7.16
AFHB & RB Chemical Decon. Water	7.17
RCS Accident Water	7.18
RCS Flush and Drain Water	7.19
RCS Decon. Water and RCS Chemical Decon. Water	7.20

Table 7.15. Dose Estimates for the Maximum Exposed Individual Caused by Failure of a HEPA Filter during Processing of Reactor Building Sump Accident Water

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	1.1×10^{-2}	1.0×10^{-1}	3.9×10^{-2}
	Ground Shine	2.8×10^{-1}	2.8×10^{-1}	2.8×10^{-1}
	Vegetable Use	2.2	11	6.5
	Total	2.5	11	6.8
Nearest milk goat	Inhalation	2.5×10^{-2}	5.0×10^{-2}	2.8×10^{-2}
	Ground Shine	2.8×10^{-1}	2.8×10^{-1}	2.8×10^{-1}
	Goat Milk Use	4.7	33	40
	Total	5.0	33	40
Nearest cow and garden	Inhalation	7.2×10^{-3}	7.2×10^{-2}	2.7×10^{-2}
	Ground Shine	2.8×10^{-1}	2.8×10^{-1}	2.8×10^{-1}
	Vegetable Use	2.2	11	6.5
	Cow Milk Use	8.4×10^{-1}	5.1	4.9
	Total	3.3	16	12

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children and for the nearest goat location are for adults for total body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 7.16. Dose Estimates for the Maximum Exposed Individual Caused by Failure of a HEPA Filter during Processing of Reactor Building Decontamination Water

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	3.6×10^{-6}	6.4×10^{-6}	4.2×10^{-6}
	Ground Shine	4.3×10^{-5}	4.3×10^{-5}	4.3×10^{-5}
	Vegetable Use	3.0×10^{-4}	1.5×10^{-3}	9.7×10^{-4}
	Total	3.5×10^{-4}	1.5×10^{-3}	1.0×10^{-3}
Nearest milk goat	Inhalation	3.6×10^{-6}	6.4×10^{-6}	4.2×10^{-6}
	Ground Shine	4.3×10^{-5}	4.3×10^{-5}	4.3×10^{-5}
	Goat Milk Use	7.1×10^{-4}	4.9×10^{-3}	5.9×10^{-3}
	Total	7.6×10^{-4}	4.9×10^{-3}	5.9×10^{-3}
Nearest cow and garden	Inhalation	2.5×10^{-6}	9.0×10^{-6}	4.0×10^{-6}
	Ground Shine	4.3×10^{-5}	4.3×10^{-5}	4.3×10^{-5}
	Vegetable Use	3.0×10^{-4}	1.5×10^{-3}	9.7×10^{-4}
	Cow Milk Use	1.7×10^{-4}	7.6×10^{-4}	7.4×10^{-4}
	Total	5.2×10^{-4}	2.3×10^{-3}	1.8×10^{-3}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children and for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 7.17. Dose Estimates for the Maximum Exposed Individual Caused by Failure of a HEPA Filter during Processing of Reactor Building Decontamination Water

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	6.7×10^{-6}	2.4×10^{-5}	1.1×10^{-5}
	Ground Shine	7.9×10^{-5}	7.9×10^{-5}	7.9×10^{-5}
	Vegetable Use	5.7×10^{-4}	2.7×10^{-3}	1.7×10^{-3}
	Total	6.6×10^{-4}	2.8×10^{-3}	1.8×10^{-3}
Nearest milk goat	Inhalation	6.7×10^{-6}	1.2×10^{-5}	7.9×10^{-6}
	Ground Shine	7.9×10^{-5}	7.9×10^{-5}	7.9×10^{-5}
	Goat Milk Use	1.3×10^{-3}	9.2×10^{-3}	1.1×10^{-2}
	Total	1.4×10^{-3}	9.3×10^{-3}	1.1×10^{-2}
Nearest cow and garden	Inhalation	4.6×10^{-6}	1.7×10^{-5}	7.4×10^{-6}
	Ground Shine	7.9×10^{-5}	7.9×10^{-5}	7.9×10^{-5}
	Vegetable Use	5.7×10^{-4}	2.7×10^{-3}	1.7×10^{-3}
	Cow Milk Use	3.2×10^{-4}	1.4×10^{-3}	1.4×10^{-3}
	Total	9.7×10^{-4}	4.2×10^{-3}	3.2×10^{-3}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children and for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 7.18. Dose Estimates for the Maximum Exposed Individual Caused by Failure of a HEPA Filter during Processing of RCS Accident Water

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	4.8×10^{-3}	7.3×10^{-2}	8.9×10^{-4}
	Ground Shine	6.5×10^{-3}	6.5×10^{-3}	6.5×10^{-3}
	Vegetable Use	1.4	5.5	1.5×10^{-1}
	Total	1.4	5.6	1.6×10^{-1}
Nearest milk goat	Inhalation	1.9×10^{-3}	3.0×10^{-2}	6.5×10^{-4}
	Ground Shine	6.5×10^{-3}	6.5×10^{-3}	6.5×10^{-3}
	Goat Milk Use	2.8×10^{-1}	1.6	9.2×10^{-1}
	Total	2.9×10^{-1}	1.6	9.3×10^{-1}
Nearest cow and garden	Inhalation	3.3×10^{-3}	5.1×10^{-2}	6.1×10^{-4}
	Ground Shine	6.5×10^{-3}	6.3×10^{-3}	6.5×10^{-3}
	Vegetable Use	1.4	5.5	1.5×10^{-1}
	Cow Milk Use	8.9×10^{-2}	4.0×10^{-1}	1.2×10^{-1}
	Total	1.5	6.0	2.8×10^{-1}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children and for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 7.19. Dose Estimates for the Maximum Exposed Individual Caused by Failure of a HEPA Filter during Processing of RCS Flush and Drain Water

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	2.4×10^{-2}	3.7×10^{-1}	4.4×10^{-3}
	Ground Shine	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}
	Vegetable Use	7.0	28	7.4×10^{-1}
	Total	7.1	28	7.8×10^{-1}
Nearest milk goat	Inhalation	9.6×10^{-3}	1.5×10^{-1}	3.3×10^{-3}
	Ground Shine	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}
	Goat Milk Use	1.5	8.1	4.6
	Total	1.5	8.3	4.6
Nearest cow and garden	Inhalation	1.6×10^{-2}	2.5×10^{-1}	3.1×10^{-3}
	Ground Shine	3.3×10^{-2}	3.3×10^{-2}	3.3×10^{-2}
	Vegetable Use	7.0	28	7.4×10^{-1}
	Cow Milk Use	4.4×10^{-1}	1.9	5.7×10^{-1}
	Total	7.5	30	1.3

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children and for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 7.20. Dose Estimates for the Maximum Exposed Individual Caused by Failure of a HEPA Filter during Processing of RCS Decontamination Water or RCS Chemical Decontamination Water

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	4.9×10^{-3}	7.5×10^{-2}	8.9×10^{-4}
	Ground Shine	6.5×10^{-3}	6.5×10^{-3}	6.5×10^{-3}
	Vegetable Use	1.4	5.7	1.5×10^{-1}
	Total	1.4	5.8	1.6×10^{-1}
Nearest milk goat	Inhalation	2.0×10^{-3}	3.1×10^{-2}	6.5×10^{-4}
	Ground Shine	6.5×10^{-3}	6.5×10^{-3}	6.5×10^{-3}
	Goat Milk Use	3.0×10^{-1}	1.6	9.2×10^{-1}
	Total	3.1×10^{-1}	1.6	9.3×10^{-1}
Nearest cow and garden	Inhalation	3.4×10^{-3}	5.2×10^{-2}	6.1×10^{-4}
	Ground Shine	6.5×10^{-3}	6.5×10^{-3}	6.5×10^{-3}
	Vegetable Use	1.4	5.7	1.5×10^{-1}
	Cow Milk Use	9.0×10^{-2}	4.0×10^{-1}	1.2×10^{-1}
	Total	1.5	6.2	2.8×10^{-1}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children and for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

7.1.6 Liquid Processing Costs

The alternative water processing facilities and associated operating costs are detailed in Appendix K. Based upon the details contained in Appendix K, the staff has bounded the costs for the various water processing alternatives as a function of the contaminated waters to be processed. The staff believes that the costs for water processing and facility construction will lie somewhere between $21 million and $29 million regardless of the combination of alternatives selected. The staff has made no attempt to quantify the costs of licensing or qualification testing of the various alternatives, therefore these costs should only be used as relative comparison.

7.1.7 Comparison of Liquid Waste Treatment Alternatives

The alternatives discussed above were compared on a relative basis considering time to initiate processing, time to complete processing, worker exposure, quality of liquid effluent (if any) and costs. Table 7.21 summarizes these parameters as a function of alternative. The conclusions that can be drawn from this table are as follows:

- Time

  The times to initiate processing vary significantly due to facility and equipment availability. Those alternatives which use systems and equipment currently installed or being installed can be initiated in 0 to 6 months, while those alternatives requiring new facilities or hardware take a minimum of 3 years.

  The implementation times to process 100,000 gallons vary from 1 week to 4 months with most alternatives in the 1- to 3-month range. The ion exchange techniques require the least time for implementation.

- Worker Exposure

  For comparison purposes, it was assumed that worker exposure was based on processing the RB sump water. As shown, direct immobilization leads to maximum exposure, 16 person-rem per 100,000 gallons of water processed (see Table 7.21).

  The estimated exposures for the other alternatives are lower than the exposure for direct solidification and roughly comparable to one another.

- Number of MPCs in Effluent

  The allowable Maximum Permissible Concentrations (MPC) in 10 CFR Part 20 are based on the relative toxicity of radionuclides released to either air or water and provide a measure of the processing performance for a particular alternative. Thus, the number of MPC's in liquid effluents are a measure of relative toxicity among the various alternatives. As shown, systems which process RB sump and primary system water on a once through basis and do not recycle or polish the effluent with a second system or technique, have MPC values as much as two orders of magnitude higher than other treatment alternatives. The highest MPC values correspond to those alternatives which exhibit the poorest relative processing performance. As shown in Table 7.21, the estimated processing performance of EPICOR II for primary system water compares favorably with the most effective zeolite-based treatment systems (i.e., zeolite/EPICOR II and SDS/EPICOR II) and exhibits better performance than the remaining zeolite-based treatment systems, including the modified EPICOR II. However, it should be noted that the predicted performance of EPICOR II is based on the operating performance of the system to date. The predicted performance of the zeolite-based systems is based, in part, on conservatively estimated decontamination factors for those system components for which there is no site operating experience.

- Costs

  Treatment costs were summarized in Section 7.1.6 and waste disposal costs are summarized in Section 9, with the basis for all estimates presented in Appendix K.

7.2 DISPOSAL OF PROCESSED WATER

Processed water is the liquid effluent arising from treatment of AFHB accident water, RB sump water, and RCS primary system water. It is distinguished from other contaminated water by its tritium content. About 743,000 gallons of this water are in storage, and another 796,000 gallons will arise from future treatment of RB sump and primary system water. This section discusses the alternatives considered for disposition of processed water and their impacts. In general, where disposal alternatives result in liquid and gaseous effluents to the environment, the radionuclide distributions provided in the source term tables in this section account for greater than 95% of the calculated offsite doses.

Table 7.21. Parameters for Comparison of Liquid Waste Treatment Alternatives

Alternative	Parameter									
	Estimated Implementation Date[a]	Time to Implement per 100,000 gallons	Estimated Worker Dose person-rem per 100,000 gallons	MPC's in Liquid[b] Effluent		Other Wastes[c] Generated				
				RB Sump Water	Primary System Water	Immobilized Solids	Concentrated Liquids	Low-Level Resins	High-Specific Activity Resins	High-Specific Activity Zeolites
Direct immobilization[d]	1/84	3 mos	16	NA	NA	+				
Long-Term Storage	1/84	1 wk	NA	NA	NA					
Evaporator/ Resin System[d]	1/84	1 mo	NA	NA	NA		+	+		
Bitumen/Resin System	1/84	4 mo	NA	NA	NA	+		+		
Zeolite Based Systems										
- SDS	2/81	1 mo	1.1	6.2×10^3	5.4×10^4			+		+
- Modified SDS	3/81	1 mo	1.1	2.2×10^3	2.2×10^4			+		+
- Zeolite/ Evaporator	1/84	1 mo	2.0	3×10^2	2.7×10^2		+	+		+
- Zeolite/ EPICOR II	3/81	1.5 mo	1.6	3×10^2	27			+		+
- SDS/ EPICOR II	2/81	1.5 mo	1.6	3×10^2	27			+		+
- Modified EPICOR II	2/81	1.5 mo	NA	NA	2.2×10^4			+		+
EPICOR II System	1/81	1.5 mo	NA	NA	35			+	+	

[a]Based on time zero of 1st quarter 1981. Table shows quarter/year.

[b]A measure of the relative processing performance of the alternatives suitable for processing RB sump water and primary system water.

[c]A + indicates waste form is generated.

[d]Alternative considered suitable for treatment of chemical decontamination solutions.

7.2.1 Current Status and Projected Water Disposal Requirements

The current status of processed water and the amounts that may be generated for disposal are described below.

7.2.1.1 Current Status

As described in Section 7.1.1, 743,000 gallons of processed AFHB accident water are in storage on the site. This water contains about 360 Ci of tritium (H-3) plus about 0.8 Ci of cesium; characteristics of the water were given in Table 7.3.[3]

No decision has been made as to which treatment system(s) will be used to process the accident water in the reactor building sump or the water in the reactor coolant system (RCS). However, the tritium concentration will not be changed by whatever water treatment alternative is selected unless clean water is added. The current tritium inventory in the plant is as follows:

Processed Water in Storage (mostly AFHB). About 360 Ci of tritium in 743,000 gallons of water with an average tritium concentration of 0.13 μCi/mL.

Untreated Reactor Building Sump Water. About 2520 Ci of tritium in 700,000 gallons of accident water with an average tritium concentration of 0.95 μCi/mL.

Untreated RCS Water. About 30 Ci of tritium in 96,000 gallons of water with an average tritium concentration of 0.08 μCi/mL.

The processed water available from these three sources is about 1,540,000 gallons containing about 2,910 curies of tritium plus trace quantities of other radionuclides. This amount of tritium is the maximum that will be available in processed water. Unavoidable in-leakage and cross contamination of water sources might increase the volume of water available for disposal and thereby reduce concentrations of tritium by dilution.

7.2.1.2 Projected Water Disposal Requirements

As described above, a minimum of 1,540,000 gallons of processed water will be available for disposal. The characteristics of the processed AFHB water are known since this water has already been processed through EPICOR-II. The characteristics of the processed water arising from treatment of RB sump water and RCS primary system water are not known since they will depend on the treatment system(s) used. However, the tritium content of this untreated water will not change during treatment, so the current tritium inventory of 2,910 curies represents a maximum. Reuse of processed water and other activities could reduce this tritium inventory due to evaporation losses. Table 7.22 summarizes these potential losses and the activities that lead to them. However, for the purposes of evaluating alternative methods of processed water disposition and their impacts, the maximum tritium inventory of 2,910 curies was used. If the losses shown in Table 7.22 are experienced, then the impacts due to tritium releases will be proportionately lower than those presented in succeeding sections. Reuse of processed water is discussed in detail in Appendix F.

The volume of 1,540,000 gallons of processed water is a minimum and could increase by 10 to 20 percent. This increase could be due to in-leakage into the reactor building, addition of clean water used for decontamination to the reactor building sump and commingling of clean water with processed water. NRC has directed the licensee to minimize the volume of accident related water.[21] Any such volume increases would merely dilute the concentrations of tritium and other contaminants in the AFHB water and the water to be processed. To ensure that release impacts are assessed under worst-case conditions, the discussions that follow are based on this minimum volume of 1,540,000 gallons.

The characteristics of treated reactor building sump water and RCS primary system water will depend on the treatment system(s) selected to process this water. The alternatives being considered were discussed in Section 7.1.3. Staff estimates of the quality of processed water that

Table 7.22. Tritium Evaporation Losses from Potential Reuse Applications[a] and Other Facilities

Application	Water Volume	Tritium Initial Loss Rate[c]
1. Shielding of ion exchange system in spent fuel pool	230,000 gallons of processed AFHB	50 Ci/yr
2. Use in SFPs and FTC during defueling	1,040,000 gallons of processed water	500 Ci/yr
3. Use to shield spent fuel and waste in SFPs after defueling	690,000 gallons of processed water	300 Ci/yr
4. Loss from reactor building sump prior to processed	700,000 gallons of unprocessed reactor building water	500 Ci/yr[b]

[a]See Appendix F for discussion of reuse applications.

[b]Conservative estimate see Section 5.2.4.1.

[c]Rate will decrease exponentially with time.

would arise from implementation of various alternatives for treatment of RB sump water are presented in Table G.19 of Appendix G and those for RCS (primary system) water are presented in Table G.20 of Appendix G.

To bound the concentrations of radionuclides in the processed water to be disposed of, the best- and worst-case treatment systems, were identified from these tables and used to characterize processed reactor building and RCS water. The system which produced effluent with the lowest concentrations of contaminants without significantly delaying cleanup, was the SDS/EPICOR II alternative. The other system which produced effluent with the highest concentrations of contaminants, was the SDS alternative. The characteristics of the effluents arising from the use of these treatment systems were combined with those of the stored AFHB water to provide bounding cases for processed water. Table 7.23 shows the best-case conditions with SDS/EPICOR II system treatment. The worst-case conditions, based on the SDS alternative, are shown in Table 7.24.

For purposes of this analysis, it has been conservatively assumed that, for these two conditions, all processed water will eventually be disposed of and will not be reused regardless of the future disposition of the facility. In the event water is retained in the spent fuel pools after defueling and this water is left in place, the volumes would be reduced by about 690,000 gallons.

7.2.2 Alternative Methods Considered

Disposal alternatives were evaluated on the basis of the bounding conditions for radionuclide concentrations given in Section 7.2.1. Ten major alternatives have been considered for disposition of processed water. As shown in Figure 7.10, these alternatives can be grouped as follows:

- Long-Term, Onsite Storage
  - In liquid storage tanks
  - Solidified as cement slabs
- Onsite Disposal
  - As solidified low-level waste in shallow land burial trenches
  - As bulk liquid using an underground injection well.

Table 7.23. Radionuclide Inventory - Processed Water
Best Case - SDS/EPICOR II Treatment
(Ci)

Radionuclide	AFHB[a] Water (743,000 gallons)	Reactor Building Sump[b] Water (700,000 gallons)	Primary[c] System Water (96,000 gallons)	Total[d] (1,540,000 gallons)
H-3	3.6×10^{2}	2.5×10^{3}	2.7×10^{1}	2.9×10^{3}
Cs-137	5.7×10^{-1}	4.7×10^{-7}	9.9×10^{-9}	5.7×10^{-1}
Cs-134	2.5×10^{-1}	6.6×10^{-8}	1.5×10^{-9}	2.5×10^{-1}
Sr-90	ND[e]	4.1×10^{-5}	4.6×10^{-5}	8.7×10^{-5}
Sr-89	ND	1.1×10^{-6}	4.8×10^{-6}	5.9×10^{-6}
Ru-106	$<2.1 \times 10^{-2}$	3.9×10^{-3}	1.7×10^{-2}	4.2×10^{-2}
Sb-125	$<1.7 \times 10^{-2}$	5.3×10^{-2}	1.4×10^{-3}	7.1×10^{-2}
Te-127m	ND	1.3×10^{-3}	1×10^{-1}	1×10^{-1}
Ce-144	$<1.8 \times 10^{-2}$	2.5×10^{-4}	5×10^{-3}	2.3×10^{-2}

[a]From Table F.1.
[b]From Table G.19.
[c]From Table G.20.
[d]Rounded to two significant figures.
[e]"ND" means not detectable.

Table 7.24. Radionuclide Inventory - Processed Water
Worst Case - SDS Treatment
(Ci)

Radionuclide	AFHB[a] Water (743,000 gallons)	Reactor Building Sump[b] Water (700,000 gallons)	Primary[c] System Water (96,000 gallons)	Total[d] (1,540,000 gallons)
H-3	3.6×10^{2}	2.5×10^{3}	2.7×10^{1}	2.9×10^{3}
Cs-137	5.7×10^{-1}	4.3	9.9×10^{-2}	5.0
Cs-134	2.5×10^{-1}	6.6×10^{-1}	1.5×10^{-2}	9.3×10^{-1}
Sr-90	ND[e]	4.1	4.6	8.7
Sr-89	ND	1.1×10^{-1}	0.48	0.6
Ru-106	$<2.1 \times 10^{-2}$	3.9	17	21
Sb-125	$<1.7 \times 10^{-2}$	53	1.4	54
Te-127m	ND	0.7	50	51
Ce-144	$<1.8 \times 10^{-2}$	0.25	5.0	5.3

[a]From Table F.1.
[b]From Table G.19.
[c]From Table G.20.
[d]Rounded to two significant figures.
[e]"ND" means not detectable.

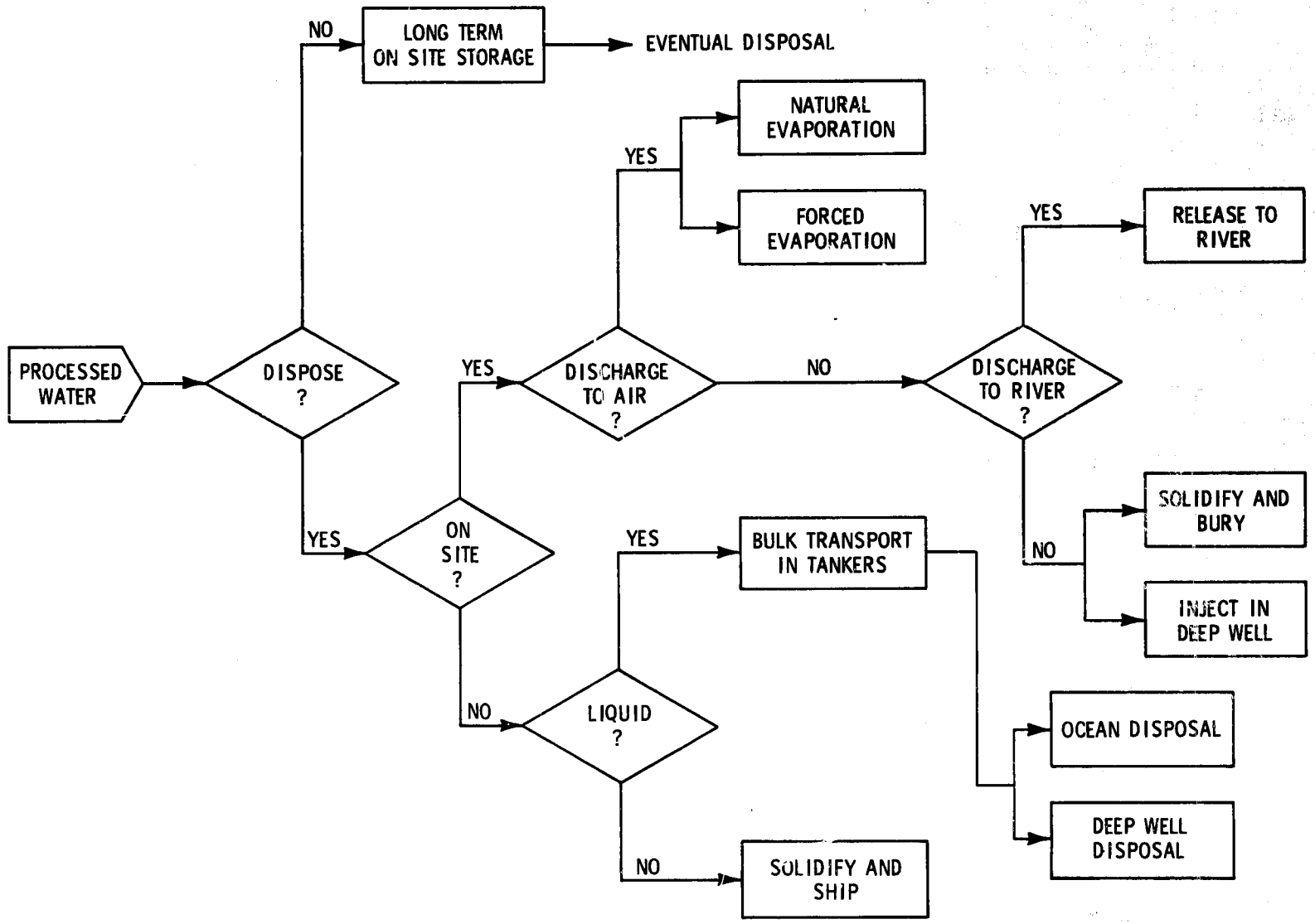

Figure 7.10. Decision Tree for Disposition of Processed Water.

- Shipment Offsite for Disposal
  - As bulk liquid for deep-well-injection disposal
  - As bulk liquid for ocean disposal
  - Solidified and packaged for disposal at a low-level radioactive waste diposal facility.
- Discharge to the Environment
  - Controlled release to the Susquehanna River
  - Controlled release to air via natural evaporation
  - Controlled release to air via forced evaporation.

Evaluations for these alternatives are presented below.

7.2.2.1 Long-Term, Onsite Storage

Under this alternative, processed water would be retained at the plant site for periods in excess of the plant's projected operating lifetime of 25 or more years. The water could be stored in liquid form in large storage tanks or it could be solidified with cement and stored in solid form. As shown in Tables 7.23 and 7.24, the principal contaminant is tritium, a beta emitter. Since beta radiation does not penetrate the tanks or the cement, direct radiation exposure is not a problem during storage in either liquid or solid form. However, immobilization of the processed water in cement would not prevent the loss of tritium to the environment by diffusion.

Storage as Bulk Liquid

Long-term storage (at least 10 half lives) to reduce concentrations of tritium either to innocuous levels or to primary drinking water standards is not practicable because of tritium's 12.8-year half-life. Given a maximum tritium concentration of 1.0 μCi/mL in storage tanks, attainment of EPA primary drinking water standard concentrations (2×10^{-5} μCi/mL) would require about 200 years. Retention for these periods of time is not considered practicable because of the inability to guarantee institutional controls for periods in excess of 100 years. In effect, the TMI site would become a low-level waste disposal site for radioactivity in a mobile form. The staff does not consider this a satisfactory alternative.

Two outside processed water storage tanks with a capacity of 500,000 gallons each have been constructed.

A criterion for storage of processed accident water in these tanks is that the content of radioactivity stored in each outside processed water storage tank should be limited such that a tank failure would not result in greater than 10 CFR Part 20 (Table II, Col. 2) concentrations at the nearest drinking water intake for combined radionuclides as a function of actual tank volume. In order to assure conformance with this criterion, several assumptions and calculations are required. Details of the assumptions used by the NRC staff for this storage criterion are discussed in Section 7.2.4.2.

In equation form this criterion requires that the storage of radioactivity in an outside storage tank be limited such that in the event of a rupture:

$$\Sigma \frac{Ci}{MPCi} \leq 1$$

where:

Ci = concentration (μCi/mL) of the ith nuclide at the nearest downstream drinking water intake (Brunner Island)

MPCi = maximum permissible concentration (10 CFR Part 20, Table II, Col. 2 - μCi/mL)

The concentration at the nearest drinking water intake (Ci, μCi/mL) is determined by:

$$Ci = \frac{(0.5)\ Ai}{Q\ \Delta\ t} = \frac{Ai}{6.4 \times 10^6}$$

where:

Q = minimum river flow rate to overtop Red Hill Dam (16,000 ft^3/sec)

Δt = release period (2 hours)

Ai = tank activity prior to rupture (Curies)

0.5 = fraction of the tank volume that discharges to the river

6.4×10^6 is a constant (Ci/μCi/mL)

Combining these two equations yields the limiting storage requirement expressed in the above criterion:

$$\Sigma \quad \frac{Ai}{MPCi} \leq 6.4 \times 10^6$$

Thus, the quantity (Curies) of each radionuclide in the storage tank will be considered in assuring that a tank failure would not result in greater than 10 CFR Part 20 concentrations at the nearest drinking water supply intake (see Section 7.2.4.2).

At present, 743,000 gallons of AFHB water are stored in tanks within the plant. When available, the two PWSTs will provide an additional 1 million gallons of storage capacity for a total minimum onsite storage capacity of about 1.75 million gallons. This capacity exceeds the minimum volume of about 1.54 million gallons. This excludes storage of some of this water in the spent fuel pools. If processed AFHB water is used to shield an SDS type system in the spent fuel pool, the storage capacity will increase to about 2 million gallons.

These storage tanks can be used for relatively long-term storage, 15 to 25 years or until a decision on disposal of the processed water is made. Following any such interim storage, the processed water will have to be permanently disposed of, and some of the alternatives discussed in this section will have to be considered. Therefore, while bulk liquid storage for periods up to about 25 years is practicable, it merely defers the ultimate disposition of the water and thus, is not considered further as a disposal option.

Storage As Cement Blocks

Under this alternative, the processed water would be mixed with cement to form relatively large blocks. To characterize this alternative, it was assumed that the blocks would be rectangles (6' × 6' × 10') containing 1600 gallons of processed water and weighing 40,000 pounds. About 960 blocks would be required to immobilize 1,540,000 gallons of water. Each block would require about 190 ft^2 of surface area for storage with inspection aisles between blocks. Thus, these blocks would occupy a maximum of about 4 acres of land if stored one layer only.

These cement blocks would immobilize the water, and when coated with asphalt or other weather-resistant material, they could be stored outside for relatively long periods. Following such storage, they could be shipped offsite to a radioactive waste disposal facility.

A portion of the tritium would be released to the atmosphere during immobilization. Additionally, the use of cement does not permanently immobilize the tritium. Recent work by Monsanto indicates that with similar mixtures of cement and tritiates, diffusion through the concrete leads to loss of tritium in relatively short time periods.

Storage of processed accident water for relatively long periods is a practicable alternative which would permit deferral of a decision on ultimate disposition of the water. Since immobilization of the processed water into cement blocks would minimize flexibility relative to ultimate disposition, interim storage as bulk liquid seems preferable.

7.2.2.2 Onsite Disposal

Under this alternative, the site would be used as a permanent disposal facility for processed water in either liquid or solid form.

For disposal in solid form, shallow land burial trenches would be constructed to dispose of either cement blocks (see Section 7.2.2.1) or processed water immobilized in cement and packaged in drums. Onsite disposal in liquid form would involve construction of a Class I deep injection well and injection of the processed water into the well as a means of permanent disposal. There are regulatory obstacles to implementation of both these alternatives. The deep injection well would require an EPA permit and may also be subjected to NRC licensing. An onsite shallow land burial facility would also be subject to NRC licensing. Moreover, implementation of these alternatives would convert the site into a permanent repository for low-level radioactive waste--an unacceptable condition since the staff has consistently committed to not converting the site to a permanent repository. Therefore, these alternatives are not considered further.

7.2.2.3 Shipment Offsite for Disposal

Under this alternative, processed water would be packaged in solid or bulk liquid form and transported off the plant site. If transported offsite in solid form, the material could be disposed of at a licensed, low-level radioactive waste disposal facility. If transported offsite as a bulk liquid, the water could be processed at another facility, and then disposed of in a deep injection well or in the ocean. Any of the other alternatives described below could also be implemented offsite.

7.2.2.4 Discharges to the Environment.

Three of the alternatives considered would discharge processed water to the environs under controlled conditions: (1) release of liquids to the river, (2) release of vapor to the atmosphere via natural evaporation, and (3) release of vapor to the atmosphere via forced evaporation combined with a partial release of liquids to the river.

Discharge to the Susquehannna River

Under this alternative, processed water would be sampled, analyzed, and discharged under controlled conditions to the Susquehanna River in accordance with the plant Technical Specifications which implement 10 CFR Part 20 and Appendix I criteria.

Natural Evaporation Pond

Under this alternative, processed water would be placed in a lined pond. Local climatological conditions will evaporate the water, and vapor containing tritium will be released to the atmosphere reducing the tritium concentration in the pond. Water losses due to evaporation are assumed to be offset by rainfall so there will be no net loss of pond water volume.

Forced Evaporation

The natural pond evaporation technique described above could be improved by heating the water entering the ponds. A spray system would also improve evaporation rates from the ponds. These procedures may result in a fog over the pond and river channels adjacent to the island. They would also have to be controlled to prevent worker exposures in areas contiguous to the ponds above 10 CFR Part 20 guidelines. Thus, this alternative was not evaluated in depth.

Another method of forced evaporation involves the use of the Unit 2 service water cooling tower. The mechanical draft service water cooling tower is a three-cell unit with a recirculation rate of about 19,000 gpm. Its use would provide a controlled method of simultaneously releasing processed water to the air via the cooling tower vapor and to the river via cooling tower blowdown. Use of this approach would produce fog and a small amount of drift, both containing tritium. This could be minimized by operating only during favorable meteorological conditions.

7.2.3 Details of Alternative Methods

Details of the alternatives considered practicable from a technical and regulatory standpoint are presented below. No decision has been made concerning the method that will be used to dispose of

processed water, and the alternatives described below may be used singly or in combination with each other.

7.2.3.1 Transport Offsite as Solid

Under this alternative, the water would be immobilized with cement, packaged, and transported offsite to a low-level waste disposal facility. To characterize this alternative, the staff assumed that the water would be packaged in 55-gallon drums, with each drum containing 30 gallons of immobilized water.

About 51,300 drums would have to be transported offsite in 640 shipments. From 12 to 18 months would be required to design and construct the necessary solidification facilities. At an average throughput of 4 drums per hour, or 96 drums per day, about 18 months would be required to package the 51,300 drums. This is based on operation 24 hours a day, seven days a week.

There are no technical or Federal regulatory obstacles to implementing this alternative. There are, however, potential State restrictions on the use of low-level waste disposal facilities to dispose of this material. There is a shortage of shallow land burial facilities, and it is doubtful that relatively innocuous tritiated waste will be given space that could otherwise be used for other radioactive waste.

7.2.3.2 Transport Offsite as Bulk Liquid

Under this alternative, processed accident water would be packaged as bulk liquid in tank trucks and transported away from the site. It could then be disposed of in a deep injection well, packaged for ocean disposal, or released at another location.

Current Federal regulations do not prohibit bulk liquid shipments of tritiated water. Under the LSA definitions in 10 CFR Part 71, tritium oxide in aqueous solution with concentrations up to 5 mCi/mL qualifies as LSA material. Under 49 CFR Part 173.392, bulk liquids may be transported in tank trucks provided the concentrations are less than 10 percent of the LSA concentrations. Thus, water with tritium concentrations up to 500 μCi/mL could be transported in bulk. The maximum tritium concentration expected from any source at TMI-2 is 0.95 μCi/mL. These shipments also would have to conform to other transportation criteria in 49 CFR Part 173. Tritium is a Type IV radionuclide, so each shipment would be limited to 20 curies to ensure that Type A quantity limits were not exceeded. Transport of water with the maximum tritium level of 0.95 μCi/mL in a 4,500-gallon tank truck would result in about 16 curies of tritium per truck. For the average concentration of 0.5 μCi/mL, the tritium inventory per truck will be about 8 Ci. About 340 shipments would be required to transport 1.54 million gallons.

Two alternatives were considered for disposition of the water after removal from the site. One alternative involves disposal in a deep injection well. Commercial injection wells used for disposal of hazardous waste are in operation in 25 states. These hazardous chemical wastes include solutions containing heavy metals, organic compounds, and pesticides.

Shipment of bulk liquids to a deep injection well or to a facility which prepares the processed water for ocean disposal is practical. Moreover, since immobilization with cement and packaging in drums would negate the potential for use of a deep injection well and possibly for ocean disposal, alternatives which involve bulk transport of liquids are preferred. However, transportation from the site is merely the initial step toward ultimate disposition of the processed water. Disposal through use of a deep injection well or burial at sea both involve solving potential regulatory problems in relatively new fields of regulated activities.

Disposal of hazardous and radioactive waste in deep wells is regulated by the EPA through its Underground Injection Control (UIC) Program under the Safe Drinking Water Act. This program is still evolving. Technical criteria and standards are contained in 40 CFR Part 146; proposed procedural requirements for State UIC permit programs are contained in 40 CFR Part 122; procedures for approving State programs in 40 CFR Part 123; and procedures for issuing State permits in 40 CFR Part 124. Within these regulations, Class I wells are those used to inject industrial, nuclear, and municipal wastes beneath the deepest stratum containing an underground drinking water source. These wells would be regulated by the State in which they are located and would be regulated by the State authority responsible for groundwater quality. Historically these wells have not been used for disposal of nuclear waste, and a special permit would have to be obtained to use such a well for the processed water from TMI-2. There is no certainty that a cognizant

State authority would grant a permit to dispose of processed TMI-2 accident water through a Class I well within its borders.

Similar regulatory obstacles would also have to be overcome to dispose of the processed water in the ocean. The regulations governing ocean dumping were promulgated under Title I of the Marine Protection Research and Sanctuaries Act (1972) and are currently found in 40 CFR Part 220 (EPA 1980). Under these regulations, a statement of the need for the proposed dumping and evaluation of alternative means of disposal, treatment, or recycle of the material must be included. Also, an assessment of the anticipated environmental impacts must be part of the application. This could require several years of effort.

In addition, under EPA's proposed regulations the tritiated water would have to be packaged in containers which would retain their integrity until radiodecay reduced the activity to "environmentally innocuous" levels. Satisfying this container criteria could also mitigate against this alternative dependent on what level of tritium activity is deemed innocuous in an ocean environment. Previous ocean disposal practices would permit immobilizing the liquid and packaging it in drums. Since these practices have been banned, some higher standard of packaging, as yet undefined is anticipated by the proposed regulations.

Therefore, although there are no regulatory obstacles to transporting the tritiated water off the site, there are potential regulatory obstacles to its disposition in either a deep injection well or the ocean, which make this option undesirable.

7.2.3.3 Discharge to the Susquehanna River

The release of water to the river is presently prohibited by the Commission's statement of May 25, 1979, the City of Lancaster Litigation Settlement Agreement, and the NRC Staff Order of February 11, 1980.

Criteria governing potential discharge of processed water to the river include; (1) 10 CFR Part 50 Appendix I criteria for offsite radiological exposure, (2) Clean Water Act criteria related to EPA's Primary Drinking Water Standards, (3) the plant's NPDES permit limitations for non-nuclear discharges to surface waters, (4) State and local ordinances governing point source discharges to the river, and (5) the requirements of 10 CFR Part 20.

In the event this alternative is pursued, specifications for radionuclide levels and for nonradioactive chemicals would have to be considered.

° <u>Nonradioactive Chemical Constraints</u>

Releases to the river are controlled by the plant's National Pollution Disharge Elimination System (NPDES) permit. This permit limits the pH to between 6 and 9 and the average boron concentration at the discharge point to 0.7 ppm.

° <u>Radionuclide Concentration and Dose Constraints</u>

For release to the river, the radionuclide concentrations in liquid effluents must be diluted to levels at the discharge point which are below 10 CFR Part 20 release limits for the specific radionuclides in the processed water. Another set of criteria are EPA's Primary Drinking Water Standards, which must not be exceeded at a drinking water intake downstream from the discharge point. Additionally, the dose design objectives of Appendix I of 10 CFR Part 50 must be met.

Processed water discharged to the river is mixed with service water cooling tower blowdown. The cooling tower blowdown flow is 38,000 gpm, but current technical specifications limit the flow available for dilution to 90 percent of this value, or about 34,000 gpm. Thus, processed water discharged at 1 gpm can be diluted by a factor of 34,000. A change in the Technical Specifications could increase this dilution flow to 140,000 gpm.

To consider discharges of the processed water shown in Table 7.25 a discharge rate of 10 gpm was selected. This discharge rate leads to a dilution factor of 3,400. At a continuous discharge of 10 gpm, the inventory of 1,540,000 gallons would require a release period of about 110 days. If the processed water is borated for defueling and is discharged in the borated condition, the boron concentration limitation in the NPDES permit will reduce this discharge rate. When 1,040,000 gallons of water borated to 3,650 ppm is mixed with the other processed water, a dilution factor of about 3,500 is required to satisfy the NPDES limit.

Since up to 140,000 gpm of dilution water is available, discharge rates can be varied to satisfy 10 CFR Part 20 release limits at the discharge point, Primary Drinking Water Standards at a downstream uptake for drinking water, and the NPDES limits. The dilution factor that satisfies the most limiting of these criteria will be used if this alternative is implemented.

7.2.3.4 Natural Evaporation from Ponds

The climatological conditions at the TMI-2 site are not conducive to natural evaporation as a means of reducing processed water volumes because the evaporation rates are equivalent to the amount of rainfall the area receives. Natural evaporation can, however, reduce the tritium content of processed water which is ponded due to the losses of tritiated water vapor and the addition of rainfall.

To characterize this alternative, the staff assumed an evaporation pond with a capacity of 2.5 million gallons and a surface area of 60,000 ft^2. To assess this alternative the following criteria were established:

- Criterion 1 - Offsite doses due to airborne tritium releases from the pond shall conform to levels consistent with Appendix I guidelines.
- Criterion 2 - The airborne concentrations for tritium above the pond shall be less than levels in 10 CFR Part 20 Appendix B for occupational personnel (5×10^{-6} μCi/mL).
- Criterion 3 - The processed water shall be retained in the pond at least until the concentrations of tritium are equivalent to those which naturally occur in the Susquehanna River - 200 pCi/L to 400 pCi/L.

The methodology used to consider the natural evaporation alternative is presented in Appendix S. The conditions which satisfy the above criteria and also bound (1) tritium release rates to the atmosphere, (2) exposure to onsite workers, and (3) time frames to achieve tritium concentrations comparable to those in the river are defined in that appendix. These conditions are summarized below.

Under the assumed worst case conditions for offsite atmospheric releases of tritium (maximum evaporation rates - see S.2 of Appendix S), 80 percent of the tritium is released in 3 to 6 months after the processed water is placed in the pond. Maximum release rates are attained when ponding is started in July. Minimum release rates are attained when ponding is started in January. In both cases, essentially the entire inventory of about 2900 Ci of tritium is released to the atmosphere during the first year.

Under the assumed worst-case conditions for occupational exposure (static conditions above the pond, see S.3 of Appendix S) ponding of 0.5 μCi/mL water in July may not be acceptable. For the average relative humidity during this month, the tritium concentrations in the ponded water must be maintained below 0.36 μCi/mL to insure that airborne concentrations of tritium above the pond are acceptable for restricted use. An extreme worst-case climatological condition is represented by several 90°F days with 90 percent relative humidity. Under these conditions, the ponded water tritium concentration must be maintained below 0.17 μCi/mL.

Operating practices can avoid these conditions. The 1,540,000 gallons of processed water can be ponded during the winter months and the exchange between tritiated pond water and untritiated precipitation will reduce the level below 0.17 μCi/mL before the worst-case summer month of July. If this is not practicable, processed water can be diluted to reduce the average concentration during the first July below 0.17 μCi/mL. In both cases, tritium concentrations in the pond can be controlled to maintain airborne concentrations below 10 CFR Part 20 Appendix B occupational exposure limits for restricted use under poor climatological conditions.

The time frames to achieve tritium concentrations in the 200 to 400 pCi/L range were determined under maximum and minimum evaporation rate conditions. Under maximum evaporation rate conditions (case one in Appendix S), tritium concentrations approach these levels in 1½ to 2 years. When the pond is filled to capacity (2.5 million gallons) these levels are attained in about 3 to 5 years.

While the tritium concentration reduction that takes place due to evaporation of tritiated water and dilution with precipitation is rather rapid, the concentrations of other radionuclides in the pond water would remain essentially unchanged. This does not present a problem for processed

accident water with characteristics similar to those represented by the SDS/EPICOR II treatment case. However, the relatively large inventory of cesium, strontium, and antimony present in processed accident water with characteristics similar to those represented by the SDS treatment case could be a problem because these radionuclides would settle to the bottom of the pond and be retained in the sediment. Therefore, the type and quality of radionuclides other than tritium should be considered if the evaporation alternative is implemented.

In summary, evaporation of accident water from a natural evaporation pond will substantially reduce tritium concentrations over relatively short time periods and is considered a practicable alternative when coupled with retention of the ponded water until tritium concentrations approach natural levels.

7.2.3.5 Forced Evaporation Through Cooling Tower

Under this alternative, processed water would be evaporated using the mechanical draft service water cooling tower. Figure 7.11 presents a simplified flow diagram for disposition of the processed water using this alternative. As shown, the action of the cooling tower will lead to the formation of water vapor containing tritium. This tritiated vapor will be released to the atmosphere. Most of the fission product contaminants in the processed water will remain in the concentrated cooling tower blowdown, but a small fraction of these contaminants in the cooling tower blowdown would be diluted to the extent required to satisfy 10 CFR Part 20 criteria prior to discharge to the river. Typically 90 to 95 percent of the tritium in the processed water is released to the atmosphere, while the remainder is discharged with the blowdown. The reverse is true for fission product contaminants, in which 20 to 30 percent of the radionuclides are released to the atmosphere via the drift while 70 to 80 percent of the radionuclides are discharged with the blowdown.

To evaluate this alternative, the following assumptions relative to cooling tower performance were made.

- The processed water feed rate to the tower would be 100 gpm.
- The evaporation rate would be 93 gpm and the drift rate would be 2 gpm.
- The air flow would be 2.25×10^6 cfm.
- The blowdown rate would be 5 gpm.

Based on these assumptions, 95 percent of the processed water will be released to the atmosphere in the form of vapor. This vapor will contain about 95 percent of the tritium, originally within the processed water feed and about 30 percent of the fission product contaminants in the processed water feed. The radionuclides not released to the atmosphere will be present in the cooling tower blowdown in concentrated form. For the stated assumptions, the concentration factor across the cooling tower is 14.3. Therefore the concentrations of radionuclides in the cooling tower blowdown will be 14.3 times these concentrations in the processed water feed. Based on current technical specifications, up to 34,000 gallons of clean water would be available to dilute the blowdown. At a blowdown rate of 5 gpm, discharges to the river can be diluted by a factor of 6800. With a change in technical specifications, the water available for dilution could be increased to about 140,000 gpm without additional facilities.

The cooling tower would be operated intermittently under favorable meteorological conditions. At the assumed feed rate of 100 gpm, about 260 hours of cooling tower operation are needed to process the 1,540,000 gallon inventory. Releases would be made in accordance with the dose design objectives of Appendix I to 10 CFR Part 50 and the concentration levels of 10 CFR Part 20.

7.2.4 Effluents and Releases to the Environment

The nature and impacts of the releases that could arise from implementation of the processed-water disposal alternatives discussed above are summarized in this section. Releases under both normal and accident conditions are discussed.

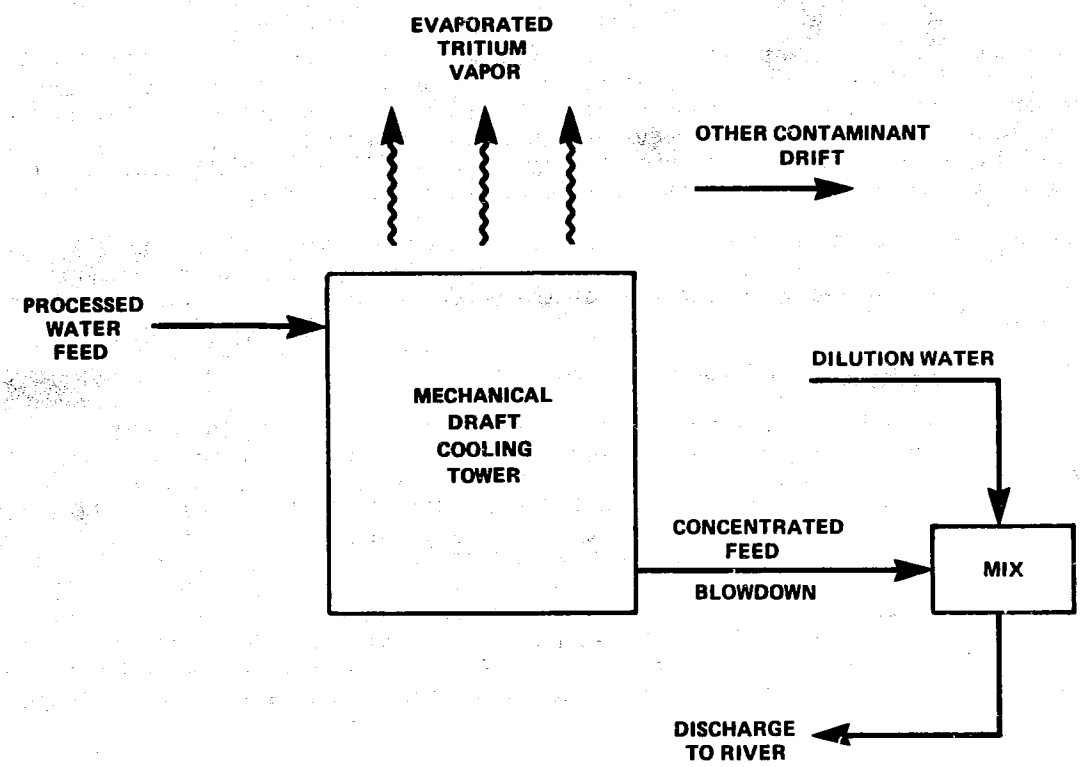

Figure 7.11. Simplified Process Diagram Forced Evaporation through Cooling Tower.

7.2.4.1 Effluents and Releases from Normal Operations

The alternatives described above which involve bulk liquid storage in the processed water storage tanks, immobilization, discharge to the Susquehanna River, natural evaporation in a lined pond, and forced evaporation through the cooling tower lead to releases under normal conditions of operations.

Storage in Processed Water Storage Tanks

During the transfer of processed accident water to and between processed water storage tanks, gaseous effluents could be released through the tank vents. This can occur during a tank filling operation because the air is displaced through the tank vent. Assuming that the air in the tank is in equilibrium with the water at 100 percent relative humidity and the tritium concentration in the water is 0.95 μCi/mL (the maximum), the air leaving the vent would contain 34×10^{-6} μCi/mL of tritium. For a filling rate of 30 gpm, the tank venting rate would be about 4 cfm or 3.8 μCi of tritium per minute, which would result in a total discharge of 0.13 Ci of tritium.

Discharges to the Susquehanna River

The characteristics of the effluents released to the river depend on the quality of the processed accident water and the dilution factor. Section 7.3.3.2 assumed the maximum volume of water available for dilution would be 34,000 gpm of cooling water. This quantity of dilution water and a discharge rate of 10 gpm were used to determine the characteristics of effluents at the discharge point. These characteristics are shown on Table 7.25, the assumed best-case conditions. Similar information on the assumed worst-case condition is presented in Table 7.26.

Table 7.25. Releases to the Susquehanna River for the Best-Case--SDS/EPICOR II[a]

Radionuclide	Average Concentration[b] (μCi/mL)	Radionuclide Inventory (curies)	Average Concentration After Dilution[c] (μCi/mL)
H-3	0.50	2.9×10^{3}	1.5×10^{-4}
Cs-137	9.8×10^{-5}	5.7×10^{-1}	2.9×10^{-8}
Cs-134	4.3×10^{-5}	2.5×10^{-1}	1.3×10^{-8}
Sr-90	1.5×10^{-8}	8.7×10^{-5}	4.4×10^{-12}
Sr-89	1.0×10^{-9}	5.9×10^{-6}	2.9×10^{-13}
Ru-106	7.2×10^{-6}	4.2×10^{-2}	2.1×10^{-9}
Sb-125	1×10^{-5}	7×10^{-2}	2.9×10^{-9}
Te-127m	1.7×10^{-5}	1×10^{-1}	5×10^{-9}
Ce-144	3.9×10^{-6}	2.3×10^{-2}	1.1×10^{-9}

[a]Processed water with average characteristics shown in Table 7.23.

[b]Based on volume of 1,540,000 gallons.

[c]Dilution factor of 3400 based on 10 gpm discharge.

Table 7.26. Release to the Susquehanna River for the Worst-Case SDS[a]

Radonuclide	Average Concentration[b] (μCi/mL)	Radionuclide Inventory (curies)	Average Concentration After Dilution[c] (μCi/mL)
H-3	0.50	2.9×10^{3}	1.5×10^{-4}
Cs-137	8.6×10^{-4}	5	2.5×10^{-7}
Cs-134	1.6×10^{-4}	9.3×10^{-1}	4.7×10^{-8}
Sr-90	1.5×10^{-3}	8.7	4.4×10^{-7}
Sr-89	1×10^{-4}	6×10^{-1}	3×10^{-8}
Ru-106	3.6×10^{-3}	21	1.1×10^{-6}
Sb-125	9.3×10^{-3}	54	2.7×10^{-6}
Te-127m	8.8×10^{-3}	51	2.6×10^{-6}
Ce-144	9.1×10^{-4}	5.3	2.7×10^{-7}

[a]Water treated with SDS on once-through basis will have to be recycled to achieve water quality acceptable for discharge in accordance with Appendix I effluent technical specifications.

[b]Based on volume of 1,540,000 gallons.

[c]Dilution factor of 3400.

Evaporation from the Lined Ponds

The only effluent of concern released during normal operations of the lined evaporation ponds would be tritiated water vapor. The amount of tritiated effluent released depends on varying meteorological conditions, i.e. time of ponding, and initial concentration of the water. For the case of July ponding of 0.50 μCi/mL water, approximately 1300 Ci could evaporate during the first month, 700 Ci during August and about 300 Ci during September. The remaining inventory would continue to be diluted by evaporation and precipitation. If January ponding of the water is implemented, it would take about six months to release this same 2300 Ci. In either case, most of the tritium evaporates within the first year after ponding. The total tritium inventory of 2910 curies would ultimately be evaporated over a period of several years, to a concentration level comparable to that in the Susquehanna River (200 to 400 pCi/L).

Forced Evaporation in Cooling Tower

The assumptions presented in Section 7.2.3.5 for cooling tower performance were used to estimate releases. Table 7.27 shows estimated radionuclide releases to the atmosphere and the liquid releases via cooling tower blowdown discharges. The liquid releases would have concentrations 14.3 times the effluent concentrations, and would be diluted with up to 34,000 gpm of water before discharge to the river. Table 7.27 presents these releases for best- and worst-case processed water effluents to the cooling tower.

The quantities of radionuclides shown would be released intermittently over 260 hours of cooling tower operation during a 12- to 15-month period. The actual release rates would be dictated by compliance with levels which satisfy 10 CFR Part 20 criteria.

Table 7.27. Normal Releases from Cooling Tower Evaporation of Process Water

Radionuclide	Curies Released		
	In Influent	To Atmosphere	To Blowdown[b]
Best-Case Influent SDS/EPICOR II			
H-3	2910	2710	200
Cs-137	0.57	0.16	0.41
Cs-134	< 0.25	< 0.07	< 0.18
Sr-90	8.7×10^{-5}	2.5×10^{-5}	6.2×10^{-5}
Sr-89	5.9×10^{-6}	1.7×10^{-6}	4.2×10^{-6}
Sb-125	7.1×10^{-2}	2×10^{-2}	5.1×10^{-2}
Ru-106	4.2×10^{-2}	1.2×10^{-2}	3.0×10^{-2}
Te-127m	1×10^{-1}	2.9×10^{-2}	7.1×10^{-2}
Ce-144	2.3×10^{-2}	7×10^{-3}	1.6×10^{-2}
Worst-Case Influent SDS [a]			
H-3	2910	2710	200
Cs-137	5.0	1.4	3.6
Cs-134	9.3×10^{-1}	2.7×10^{-1}	6.6×10^{-1}
Sr-90	8.7	2.5	6.2
Sr-89	6×10^{-1}	1.7×10^{-1}	4.3×10^{-1}
Sb-125	54	15.4	38.6
Ru-106	21	6.1	14.9
Te-127m	51	14.6	36.4
Ce-144	5.3	1.5	3.8

[a]Water treated with SDS on once-through basis will have to be recycled achieve water quality acceptable for discharge in accordance with Appendix I effluent technical specifications.

[b]Can be diluted as needed prior to release.

7.2.4.2 Accident Releases

The worst-case accident arises due to rupture of an outside processed water storage tank containing quantities of residual radioactivity at the upper limit of the storage criteria discussed in Section 7.2.2.1. For this evaluation a 500,000-gallon water storage tank containing a composite of processed AFHB water and RCS and reactor building sump water processed by the SDS system is assumed to rupture and release its contents instantaneously.

Water from the tank would be partly absorbed into the ground and partly carried over several hundred yards (approximately 250 meters) via the site surface drainage system to the flood control facility located along the east channel of the island. The water reaching the collection basin of the flood control facility would be prevented from entering the east channel of the river by a mechanical sluice gate which is normally closed. Even if the sluice gate was inadvertently left in the open position, it could be closed relatively quickly following a tank rupture. For the purpose of this evaluation, however, it was conservatively assumed that half of the water from the ruptured tank would enter the river through an open sluice gate and half would be absorbed into the ground.* Based on qualitative consideration of the following factors the processed water (250,000 gallons) is assumed to enter the river over a two hour period.

- The rate of spillage from the ruptured tank,
- Physical obstructions between the tank and collection basin,
- The volume of water in the tank at the time of the break,
- The quantity of fresh water drainage from the site (e.g., if rupture occurred during a rainstorm).

A two-hour period is considered to be a lower, and therefore conservative, limit of the time period over which the river release would be expected to occur.

Once the processed water entered the east channel it would be either carried downstream or retained behind Red Hill Dam. The NRC staff estimates that Red Hill Dam would overtop at Susquehanna River flow rates in excess of 16,000 ft^3/sec. For the situation where the dam was not being overtopped, the radioactive contamination would mix over a period of days with the bulk of the water in the east channel. (Drinking water is not taken from this source.) When the dam was finally overtopped, the water would be carried downstream, but at a very low concentration as a result of mixing with the river flow. For the situation where the east channel was flowing and overtopping the dam, contaminated water would be carried downstream as soon as it enters the channel.

The two nearest downstream drinking water intakes are the Brunner Island electric station, approximately 5 miles from the site, and the Columbia intake, approximately 16 miles from the site. The Brunner Island intake, which is not a public source of drinking water, is on the west side of the river, whereas the contaminated water would be released along the east side of the river. Therefore, the upper bound estimate of the concentrations of radionuclides at this intake is determined based on the assumption that the radionuclides become uniformly mixed over the cross section of the channel. Longitudinal mixing, however, would be unlikely.

In order to estimate the maximum amount of radioactivity which could be released to the river as a result of tank rupture it is necessary to apply the following criterion discussed in Section 7.2.2.1.

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*The radionuclides in the processed water which is absorbed into the ground will not enter the river for a year or more (see Appendix V). The gradual release of these nuclides will result in insignificant environmental impacts (i.e., doses which are a small fraction of Appendix I to 10 CFR Part 50, and 10 CFR Part 20).

Specifically this criterion requires the licensee to limit processed water storage tank content such that if the tank is breached, the following equation is satisfied.

$$\Sigma \frac{Ci}{MPCi} \leq 1$$

where

Ci = concentration (μCi/mL) of the ith nuclide at the nearest downstream drinking water intake (Brunner Island)
MPCi = maximum permissable concentration (10 CFR Part 20, Table II, Col. 2 - μCi/mL).

The concentration at the nearest drinking water intake (Ci, μCi/mL) is determined by:

$$Ci = \frac{(0.5)\ Ai}{Q\Delta t} = \frac{Ai}{6.4 * 10^6}$$

where

Q = minimum river flow (16,000 ft^3/sec, cfs) to overtop Red Hill Dam
Δt = release period (2 hours)
Ai = tank activity prior to rupture (curies)
0.5 = fraction of the tank volume that discharges to the river
6.4×10^6 is a constant (Ci/μCi/mL)

Combining these two equations yields the limiting storage requirement expressed by the criterion listed above.

$$\Sigma \frac{Ai}{MPCi} \leq 6.4 * 10^6$$

Thus, this expression defines the maximum release of radioactivity to the river which could result from a rupture of an outside process water storage tank.

Based on this, Table 7.28 presents a maximum source term for this accident. These values were calculated using the relative radionuclide spectrum and quantities listed in Table 7.24 for a composite of processed AFHB water and RCS and reactor building sump water processed by the SDS system. Table 7.28 represents an upper limit estimate of radionuclide inventory which could be released to the east channel as a result of a tank rupture. It should be noted that, in accordance with the tank failure model (i.e., 50 percent of the tank inventory is discharged to the river), the radionuclide inventory in Table 7.28 is only 50 percent of the inventory that could be stored in the tank.

7.2.5 Environmental Impacts

7.2.5.1 Occupational Doses

The principal radionuclide in the processed water is tritium, a soft beta emitter, which does not constitute a direct radiation source. Tritium is, however, an internal source of radiation if ingested by workers. Since it is not contemplated that the workers will ingest processed water, there are no occupational doses attributable to processed water disposal.

Table 7.28. Maximum Activity Released to the Susquehanna River due to a Rupture of a Processed Water Storage Tank

Radionuclide	Activity (Ci)[a]
H-3	280
Cs-137	0.49
Cs-134	0.09
Sr-90	0.84
Sr-89	0.058
Ru-106	2.0
Sb-125	5.2
Te-127m	5.0
Ce-144	0.51

[a]These numbers represent one-half the maximum activity which could be stored in an outside processed water storage tank in accordance with the model described in this section and in Section 7.2.2.1.

7.2.5.2 Offsite Doses

The dose estimates presented here for disposal of the processed water are based on the source terms of Section 7.2.4. The calculational models used to make these estimates are described in Appendix W. The environmental impacts of the four options for disposing of the processed water are considered here. The four options are (1) discharging directly to the river, (2) discharging to the atmosphere via natural evaporation from a holding pond, (3) combined discharge to the atmosphere and river by forced evaporation and blowdown, and (4) ship offsite to a waste disposal site. The significance of the doses described below and their human health and environmental consequences are discussed in Section 10.3.

The option of shipping the processed water offsite would result in no exposure to the general public as the only nuclide of significant concentration is tritium. No exposure to the public would occur because tritium emits beta radiation which would not penetrate the shipping containers.

The dose estimates for the controlled release to the river option for the maximum exposed individual are listed in Tables 7.29 and 7.30 for the SDS/EPICOR II and SDS alternatives, respectively. The downstream population dose for the SDS/EPICOR II processing option was estimated to be 30 person-rems and for the SDS processing option was estimated to be 900 person-rems.

The dose estimates for releasing the processed water to the atmosphere via natural evaporation are based on the assumption that 2910 curies of tritium are released uniformly over the period of one year. The dose received by the maximum exposed individual is listed in Table 7.31. Note that for tritium the dose to each body organ is the same as that for the total-body. The 50-mile total population dose was estimated to be 30 person-rems.

Table 7.29. Dose Estimates to the Maximum Exposed Individual for Releasing Processed Water to the River - SDS/EPICOR-II Option

Pathway[a]	Dose (mrem)[a]		
	Total-Body	Bone	Liver
Fish consumption	1.1	1.2	1.5
Drinking water	2.4×10^{-2}	1.1×10^{-2}	2.0×10^{-2}
Shoreline exposure	8.5×10^{-4}	9.9×10^{-4}	4.8×10^{-3}
Total	1.1	1.2	1.6

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children and infants. The highest dose estimates for each age group are listed. The total-body values are for adults, the bone values are for children, and the liver values are for teenagers.

[b]The basis for selecting the special locations is described in Appendix W. The assumption is that drinking water is from the nearest downstream municipal intake, and shoreline exposure and fish consumption is from the center channel in the river.

Table 7.30. Dose Estimates to the Maximum Exposed Individual for Releasing Processed Water to the River - SDS Option

Pathway[a]	Dose (mrem)[a]		
	Total-Body	Bone	Liver
Fish consumption	9.1	18	11
Drinking water	6.4×10^{-1}	4.7	6.4×10^{-2}
Shoreline exposure	2.1×10^{-2}	2.4×10^{-2}	1.1×10^{-1}
Total	9.8	23	11

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children and infants. The highest dose estimates for each age group are listed. The total-body values are for adults, the bone values are for children, and the liver values are for teenagers.

[b]The basis for selecting the special locations is described in Appendix W. The assumption is that drinking water is from the nearest downstream municipal intake, and shoreline exposure and fish consumption is from the center channel in the river.

Table 7.31. Dose Estimates to the Maximum Exposed Individual for Releasing Processed Water to the Atmosphere via Natural Evaporation

Location	Pathway	Dose (mrem)[a] Total-Body
Nearest garden[b]	Inhalation	2.0×10^{-1}
	Ground Shine	0
	Vegetable Use	7.0×10^{-1}
	Total	9.0×10^{-1}
Nearest milk goat	Inhalation	1.0×10^{-1}
	Ground Shine	0
	Goat Milk Use	7.9×10^{-1}
	Total	8.9×10^{-1}
Nearest cow and garden	Inhalation	2.2×10^{-1}
	Ground Shine	0
	Vegetable Use	7.7×10^{-1}
	Cow Milk Use	3.1×10^{-1}
	Total	1.3

[a]Doses were calculated for Gi-tract, bone, liver, kidney, thyroid, lung and skin. The dose estimates for all organs except bone were the same as that for total-body. The estimates for bone were zero. Doses were calculated for four age groups; adults, teenagers, children, and infants. The largest of each group are listed in this table. Infant doses are listed for the nearest milk goat location and child doses are listed for the other two.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 mile east-northeast, nearest milk goat = 1.02 mile north, and nearest cow and garden = 1.05 mile east.

The dose estimates for the forced evaporation option for the maximum exposed individual are listed in Tables 7.32 and 7.33.

The combined 50-mile population dose and downstream population dose were estimated to be 30 person-rems for the SDS/EPICOR II processing option and 1000 person-rems for the SDS processing option, respectively.*

*The population dose estimates have been rounded to one significant figure. The combined 50-mile population dose and downstream population dose of 30 person-rems for the forced evaporation SDS/EPICOR II processing option was based on a dose of 7 person-rems from the liquid pathways and a dose of 26 person-rems from the atmospheric pathways. The combined 50-mile population dose and downstream population dose of 1000 person-rems for the forced evaporation SDS processing option was based on a dose of 590 person-rems from the liquid pathways and 440 person-rems from the atmospheric pathways.

Table 7.32. Dose Estimates for the Maximum Exposed Individual for Releasing Processed Water Via the Forced Evaporation Option - SDS/EPICOR II Processing Option

Location	Pathway	Dose (mrem)[a] Total-Body	Bone	Liver
Atmospheric Releases:				
Nearest garden[b]	Inhalation	1.9×10^{-1}	1.0×10^{-2}	2.0×10^{-1}
	Ground Shine	2.7×10^{-2}	2.7×10^{-2}	2.7×10^{-2}
	Vegetable Use	7.4×10^{-1}	4.7×10^{-1}	1.2
	Total	9.6×10^{-1}	5.1×10^{-1}	1.4
Nearest milk goat	Inhalation	2.0×10^{-1}	5.6×10^{-3}	1.0×10^{-1}
	Ground Shine	2.6×10^{-2}	2.6×10^{-2}	2.6×10^{-2}
	Goat Milk Use	6.3×10^{-1}	2.3	3.8
	Total	8.6×10^{-1}	2.3	3.9
Nearest cow and garden	Inhalation	2.1×10^{-1}	1.1×10^{-2}	2.2×10^{-1}
	Ground Shine	4.0×10^{-2}	4.0×10^{-2}	4.0×10^{-2}
	Vegetable Use	8.5×10^{-1}	6.9×10^{-1}	1.5
	Cow Milk Use	3.9×10^{-1}	5.4×10^{-1}	8.9×10^{-1}
	Total	1.5	1.3	2.7
Liquid Releases:				
Near TMI	Fish Consumption	7.6×10^{-1}	8.7×10^{-1}	1.1
	Drinking Water	4.7×10^{-3}	8.1×10^{-3}	5.3×10^{-3}
	Shoreline Exposure	6.1×10^{-4}	7.1×10^{-4}	3.4×10^{-3}
	Total	7.7×10^{-1}	8.8×10^{-1}	1.1

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. For the atmospheric releases, the doses are for children for the nearest garden and nearest cow and garden locations, and for the nearest milk goat location are for adults for total-body and for infants for bone and liver. For liquid releases, the doses for total-body are for adults, for bone are for children, and for liver are for teenagers.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations for atmospheric releases are: nearest garden = 1.05 mile east-north-east, nearest milk goat = 1.02 mile north, and nearest cow and garden = 1.05 mile east. For liquid releases the assumption was made that drinking water is from the nearest downstream municipal intake, and shoreline exposure and fish consumption is from the center channel in the river.

Table 7.33. Dose Estimates for the Maximum Exposed Individual for Releasing Processed Water Via the Forced Evaporation Option - SDS Processing Option

Location	Pathway	Dose (mrem)[a] Total-Body	Bone	Liver
Atmospheric Releases:				
Nearest garden[b]	Inhalation	1.1	14	4.3×10^{-1}
	Ground Shine	1.8×10^{-1}	1.8×10^{-1}	1.8×10^{-1}
	Vegetable Use	77	300	4.4
	Total	78	310	5.0
Nearest milk goat	Inhalation	4.3×10^{-1}	5.2	2.4×10^{-1}
	Ground Shine	1.7×10^{-1}	1.7×10^{-1}	1.7×10^{-1}
	Goat Milk Use	14	64	23
	Total	15	69	23
Nearest cow and garden	Inhalation	1.2	16	4.8×10^{-1}
	Ground Shine	2.6×10^{-1}	2.6×10^{-1}	2.6×10^{-1}
	Vegetable Use	110	440	6.2
	Cow Milk Use	6.7	27	4.5
	Total	120	480	11
Liquid Releases:				
Near TMI	Fish Consumption	6.5	13	7.9
	Drinking Water	4.4×10^{-1}	3.3	3.7×10^{-2}
	Shoreline Exposure	1.5×10^{-2}	1.7×10^{-2}	8.2×10^{-2}
	Total	7.0	16	8.0

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. For the atmospheric releases, the doses are for children for the nearest garden and nearest cow and garden locations, and for the nearest milk goat location are for adults for total-body and for infants for bone and liver. For liquid releases, the doses for total-body are for adults, for bone are for children, and for liver are for teenagers.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations for atmospheric releases are: nearest garden = 1.05 mile east-north-east, nearest milk goat = 1.02 mile north, and nearest cow and garden = 1.05 mile east. For liquid releases the assumption was made that drinking water is from the nearest downstream municipal intake, and shoreline exposure and fish consumption is from the center channel in the river.

7.2.5.3 Postulated Accident Effects

The accident described in Section 7.3.4.2 that involves releases of processed water to the river is breaking of a processed water storage tank. For this accident scenario, 50 percent of the released water will run over the surface to the flood control facility and drain into the east channel of the Susquehanna River. The potential offsite dose consequences to humans is dependent upon whether or not Red Hill Dam is overtopping. If Red Hill Dam is overtopping, the released water will be diluted with the flow of the Susquehanna River resulting in doses to humans that are fairly low and within annual limits for routine operation. However, if Red Hill Dam is not overtopping, the released radioactivity could reside in the east channel for an extended period of time at typically fairly high concentrations. Dose calculations are presented here for both river flow situations.

Flowing East Channel

The source terms that were used for the tank break accident are listed in Table 7.28. The calculations are based on the assumption that the SDS processing option is used for reactor building sump and RCS accident water. This is the least effective processing option, hence, if a more effective processing option is chosen, an accidental release would probably result in lower doses than presented here. Table 7.34 lists the offsite dose estimates to the maximum exposed individual resulting from the tank rupture accident scenario. These dose estimates were based on the calculational methods described in Appendix W. The significance of these doses related to humans are discussed in Section 10.4. The ecological significance of these accidents is discussed in Section 7.2.5.4.

Stagnant East Channel

The dose estimates for the accident situation where it is assumed Red Hill Dam is not overtopping were calculated based on the estimates for the situation where it was overtopping, except that a correction factor was employed to take into consideration the decreased dilution of the east channel, and an adjustment was made to present dose estimates on a unit consumption basis. Table 7.35 lists dose estimates for consumption of 1 liter of water or 1 kg of fish flesh from the east channel for the and SDS processing option. The fish consumption doses are fairly high, even for the lowest case. However, should an accident occur, consumption of fish caught from this pond could certainly be avoided. The significance of these doses related to humans and possible mitigative measures are discussed in Section 10.4. The ecological significance of these accidents is discussed in Section 7.2.5.4.

7.2.5.4 Radioecological Consequences of Discharge to the Susquehanna River

Additional factors which must be considered in the evaluation of the consequences of liquid radioactive discharges from the cleanup include the transport pathways, environmental removal mechanisms, accumulation and persistance of the radionuclides and consequences of radiation exposure to organisms other than man. The radionuclides of primary concern in these analyses are Cs-137, Sr-90 and H-3. Cesium-134, Sr-89, Sb-125 and traces of other radionuclides are contaminants present in radioactive waste streams but are of lesser concern, either because they are in low concentrations or have low dose factors. Further details on the significance of the nuclide mix are given in Appendix J.

Release Scenarios

The staff has postulated several mechanisms for the release of treated radioactive water from TMI and evaluated the radioecological consequences to the biota of the Susquehanna River and Chesapeake Bay.

Controlled Release

Effluent discharges to the Susquehanna River from TMI during planned controlled releases of treated water would occur at the combined two-unit discharge structure on the west shore of Three Mile Island (Figure 1.3). This river discharge structure, therefore, would release effluents into the center channel of the Susquehanna River as it flows south toward the York Haven Dam and

Table 7.34. Dose Estimates to the Maximum Exposed Individual Caused by Rupture of a Processed Water Storage Tank during High River Flows - SDS Processing Option[a]

Pathway[c]	Dose (mrem)[b]		
	Total-Body	Bone	Liver
Fish consumption	8.9×10^{-1}	1.7	1.1
Drinking water	6.2×10^{-2}	4.6×10^{-1}	6.2×10^{-3}
Shoreline exposure	2.0×10^{-3}	2.3×10^{-3}	1.1×10^{-2}
Total	9.5×10^{-1}	2.2	1.1

[a]These doses are based on the assumption that the flow in the river during the accident is sufficient to overtop Red Hill Dam.

[b]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung and skin. The maximum three organ doses are listed in this table. Doses were calculated for four age groups; adults, teenagers, children and infants. The highest dose estimates for each age group are listed. The dose estimates for total-body are for adults, for bone are for children, and for liver are for teenagers.

[c]The basis for selecting the special locations is described in Appendix W. The assumption is that drinking water is from the nearest downstream municipal intake, and shoreline exposure and fish consumption is from an area downstream of TMI.

Table 7.35. Dose Estimates for Consumption of Drinking Water or Fish Flesh from the East Channel due to an Accidental Processed Water Storage Tank Rupture during Low River Flows[a]

Processing Option	Adult Total-Body Dose from Consuming 1 liter of Water from East Channel (mrem)	Adult Total-Body Dose from Consuming 1 kg of Fish from East Channel (mrem)
SDS/EPICOR II[c]	0.14	56
SDS[b]	3.8	470

[a]The dose estimates here are based on the assumption that the tank ruptures during an extended period of low-river flow such that Red Hill Dam is not being overtopped.

[b]Values determined by multiplying appropriate values in Table 7.34 by the increased concentration factor (44,900 for drinking water and 11,200 for fish consumption), divided by the annual usage rate (730 l/yr for drinking water and 21 kg/yr for fish consumption).

[c]These values were estimated by scaling the values for the SDS processing option in this table by the ratio of the appropriate values in Table 7.29 to the appropriate values in Table 7.30. The scaling factor for drinking water was determined by dividing the total-body dose for drinking water in Table 7.29 by the total-body dose in Table 7.30 ($2.4 \times 10^{-2}/6.4 \times 10^{-1} = 0.038$). In a similar fashion the scaling factor for fish consumption was estimated (1.1/9.1 = .12).

the York Haven Generating Station (hydroelectric). Studies of river flow and effluent plume behavior at TMI have shown that the plume stays very close to the shoreline as it moves south with the center channel river flow. The wide shallow nature of the river apparently results in low initial mixing of tributary waters with river flow.

This type of flow phenomena restricts the effluent from TMI to very near the west shore of TMI[17] until it either passes over the York Haven Dam or is discharged from York Haven Pond through the hydroelectric station. After exiting the Pond, the effluent receives a considerable amount of dilution with the River flow. The most productive spawning and nursery areas of important fishes in the Pond are in the east and west river channels (Appendix E) and the areas most frequently fished by anglers are upstream of TMI between Hill, Shelley, and Fall Islands. Therefore, during planned releases, the effluent from the nuclear station discharge would not intercept or directly contact those important areas of York Haven Pond.

To illustrate controlled releases, 10 gpm of treated water was assumed to be diluted with 34,000 gpm of cooling tower blowdown before release to the center channel of the river through the plant discharge structure. The river flow during this release was assumed to be 12,600 CFS, which is a weighted average flow for dilution, whose derivation is described in Section 3.4.1. The center channel of the Susquehanna River is presumed to be carrying 25 percent of the total river flow. Exposure to biota living above York Haven Dam are calculated on the basis of the flowrate in the center channel, while exposure to organisms downstream of the dam are based on the total river flow.

The expected activities of the radionuclides of concern in the processed water are listed in Table 7.25 and 7.26 for the SDS/EPICOR II and SDS processing options, respectively. Table 7.36 lists the background concentrations of H-3, Cs-137 and Sr-90 in the water of the Susquehanna River and the detection capability of the Licensee's monitoring program. Also listed in Table 7.36 are the annual average and instantaneous concentrations in the river below York Haven Dam resulting from the release of water treated by the two options based on the 10 gpm release rate and 12,600 CFS river flow. Instantaneous concentrations due to the releases of treated effluent from the plant would, of course, depend on the rate of release and the river flow rate at that time.

Table 7.36 reveals that only H-3 and possibly Sr-90 (SDS processing) could be detected in levels above background for either treatment option. Since doses and exposures to man and other organisms are calculated on a yearly average basis, the yearly average concentrations in Table 7.36 are more useful from the standpoint of impact evaluation than the instantaneous values.

Accidental Release

The other scenario considered for the release of the treated water involves the instantaneous rupture and spillage of a 500,000-gallon holding tank. Water from the tank would run over surface to the flood control facility and be routed to the east channel of the river above Red Hill Dam, an important spawning and nursery area of the river. The concentration of radionuclides in Red Hill Pond would depend on whether or not the flow rate in the Susquehanna River was sufficient to overtop Red Hill Dam. For flows below 16,000 CFS, the dam does not overtop and there would be only limited exchange of water between the pond and the Susquehanna River at the upstream end of Three Mile Island. The concentration in the pond for this case was based on the assumption that one-half the total contents of the 500,000-gallon holding tank spilled into and was diluted by the approximately 65-million-gallon volume of Red Hill Pond. The concentrations of radionuclides were assumed to remain at these levels until the river stage was sufficient to overtop Red Hill Dam. For flows greater than 16,000 CFS, the dam would be overtopped and the radionuclides would be carried downstream.

Table 7.36. Background and Added Concentrations of Radionuclides in Susquehanna River Below York Haven Dam due to Controlled Releases from SDS/EPICOR II and SDS Treatment Systems[a]

	River Concentrations (μCi/mL)					
		SDS/EPICOR II		SDS		
Radionuclide	Background Concentration[b]	Instantaneous River Concentration[c]	Annual Average River Concentration[d]	Instantaneous River Concentration[c]	Annual Average River Concentration[d]	Licensee's Lower Limit of Detection[e] (μCi/mL)
H-3	1.8×10^{-7}	8.9×10^{-7}	2.6×10^{-7}	8.9×10^{-7}	2.6×10^{-7}	3.3×10^{-7}
Cs-137	2.8×10^{-9}	1.7×10^{-10}	5.1×10^{-11}	1.5×10^{-9}	4.5×10^{-10}	6.0×10^{-9}
Sr-90	2.3×10^{-9}	2.7×10^{-14}	7.8×10^{-15}	2.7×10^{-9}	7.8×10^{-10}	1.0×10^{-9}

[a]The annual average concentration and doses also apply to the accidental release scenarios downstream of Red Hill Dam.

[b]From Section 3 of this document.

[c]Based on source terms of Tables 7.25 and 7.26 and the volumes of Tables 7.23 and 7.24 released at a rate of 10 gpm into a river flow of 12,600 cfs.

[d]Based on all activity of Tables 7.25 and 7.26 released over a one-year period into a river flow of 12,600 cfs.

[e]From "Three Mile Island Nuclear Station Radiological Environmental Monitoring Program, Annual Report, 1979," Metropolitan Edison Company, April 1980, p. 12.

Behavior of Radionuclides in Susquehanna River

Radionuclides entering the Susquehanna River at TMI-2 will move in the downstream direction with the river flow. Some radioactivity will deposit in bottom sediments and some will be taken up by aquatic organisms. Tritium discharged to the river will be in the form of tritiated water (HTO as compared to normal water, H_2O). The chemical and physical behavior of tritiated water is basically that of ordinary water with regard to mobility, interaction with suspended particles, and general dilution.

In contrast, the isotopes Cs-134 and Cs-137 have an appreciable tendency to be adsorbed onto suspended sediments and to concentrate in aquatic organisms. Dissolved cesium introduced to the river will partition itself between the water and the suspended sediment. Contaminated sediments would collect to a certain degree behind the York Haven, Safe Harbor, Holtwood and Conowingo Dams, but would not reside there indefinitely since much of it would be flushed into the Chesapeake Bay during major floods. Although radioactively contaminated sediments would be resuspended during floods, they would simultaneously be highly diluted by the great quantity of flowing water.

As suggested in Table 7.36, it is highly unlikely that any of the radiocesium released from the TMI cleanup could be detected in the Susquehanna River, except possibly in the immediate area of the site. Radiocesium levels would be masked by background levels caused by nuclear explosive testing fallout, previous normal operating releases from Three Mile Island Units 1 and 2, and current releases from Peach Bottom Units 2 and 3.

The isotopes of strontium have a much smaller tendency than cesium to adsorb onto sediments, and thus will remain almost completely in the water. The effects of sediment on the radioecology of Susquehanna River and Chesapeake Bay are expected to be inconsequential. Details of the importance of sediment are given in Appendix T.

Behavior of Radionuclides in the Chesapeake Bay

The upper Bay area, including the Susquehanna Flats and its tributaries, are major spawning and nursery grounds for many important species of fishes and other aquatic biota. Some of the radionuclides discharged to the Susquehanna River at Three Mile Island will ultimately be carried downstream to the Chesapeake Bay either in dissolved form or on suspended sediment. The Chesapeake Bay (Figure 7.12) is an estuary where fresh waters from the rivers and salt water from the ocean mix. Salinity varies from practically zero at the mouths of the rivers up to about 35 parts per thousand (PPT) in the open ocean.

Substances which enter Chesapeake Bay mostly or entirely in the dissolved form such as all of the tritium, most of the cesium, and practically all of the strontium will eventually be transported to the sea. The transport would be caused by the combined effects of advection by fresh water and dispersion caused by the astronomical tides and wind wave activity. The "flushing time," in the Bay is approximately one year. The flushing time is indicative of the rate at which a dissolved pollutant would be purged from the Bay, but because of the effects of sediments and the nonideal mixing and entrapment in the complicated backwater areas of the Bay, small traces of radioactive contamination would probably linger for several years.

The bulk of the sediments transported into the Chesapeake Bay will be deposited in the upper Bay in the region known as the "turbidity maximum."[22] Eaton suggests that some Susquehanna River-derived sediment would be deposited as far down-Bay as the mouth of the Potomac River.[23]

Direct measurements of radionuclides released from the Peach Bottom Nuclear Plant in Conowingo Pond suggests that the maximum sediment deposition of cesium is at the mouth of the Susquehanna River, and has decreased by two orders of magnitude by the Sassafras River. Radiocesium levels are undetectable beyond that point.[24] The measurements may not have included the periods of major sediment transport during floods, which would both dilute and disperse contaminated sediment further into the Bay to levels below detection limits.

Once the contaminated sediments enter the brackish waters of the Bay, a portion of the Cs-134 which was previously adsorbed will be released back into the water column in a dissolved state.[25] Estimates of radiation doses made below for the Susquehanna River represent maximum expected values. Even with desorption of radionuclides into the water column, radiocesium concentrations are expected to be lower than in the river.

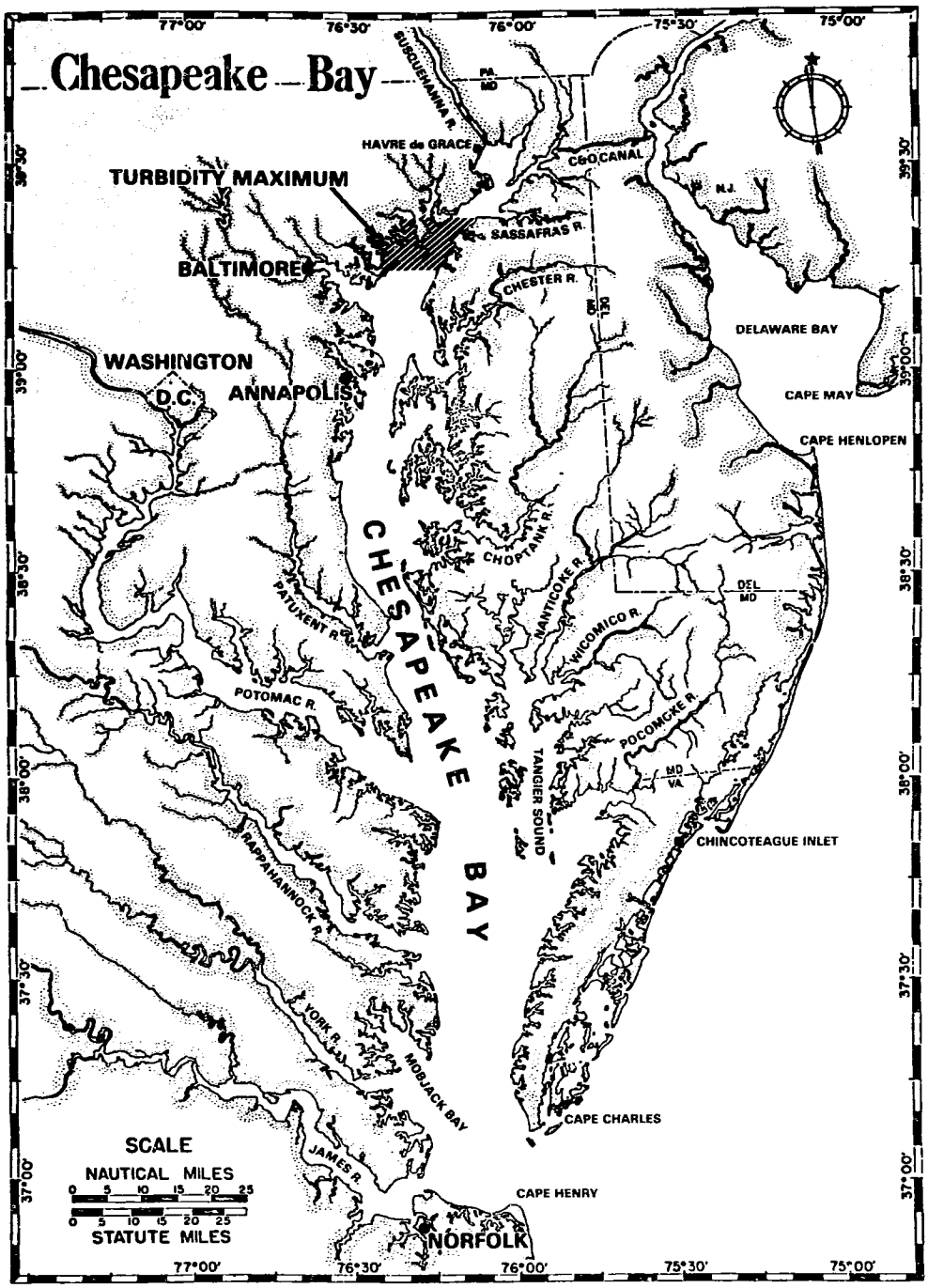

Figure 7.12. Chesapeake Bay Area.

Chesapeake Bay is habitat for a great variety of aquatic life, some of which is consumed by man. Maximum concentrations of radionuclides in the Bay water would not exceed those in the Susquehanna River because the river water undergoes further dilution with salt water and other sources of freshwater in the Bay.

The Susquehanna River discharges approximately 52 percent of the total fresh water input to the Chesapeake Bay, and about 80 percent of freshwater input above the Potomac River.[22] For releases of long duration it is possible to relate the steady state dilution in Chesapeake Bay of dissolved substances contained in Susquehanna River water directly to the observed salinity. Table 7.37 gives the additional dilution factors for Susquehanna River water as a function of salinity in the Bay. These factors are underestimates of dilution because they are based on the assumption that all fresh water in the bay comes from the Susquehanna River, while there actually are other substantial sources of freshwater input. Figures 7.13a and 7.13b show the salinity along the axis of the Chesapeake Bay for periods of high flow and low to moderate flow respectively.[26] Since doses are calculated on an average annual basis, these dilution factors may also be applied to radioactive releases of short duration. It should also be noted, while it appears that the additional dilution in the Bay would be larger for the lower river flow of Figure 7.13a, the initial dilution of TMI effluent by the river water flowing by the site would initially be lower.

Table 7.37. Conservative Dilution of Susquehanna River Water in Chesapeake Bay[a]

Salinity - Parts per Thousands	Additional Dilution of Susquehanna River Water[b]
0 (Fresh water)	1
2	1.06
5	1.17
10	1.4
15	1.75
20	2.33
25	3.5
30	7
32	11.7
35 (sea water)	Infinity

[a]For a steady release of fresh water. Conservatively assumes all fresh water input to Bay due to Susquehanna River.

[b]Example: If the Susquehanna River Concentration of Cs-137 were 1.0 μCi/L, the calculated Chesapeake Bay concentration where the salinity were 10 parts per thousand would be 1.0/1.4 = 0.714 μCi/L.

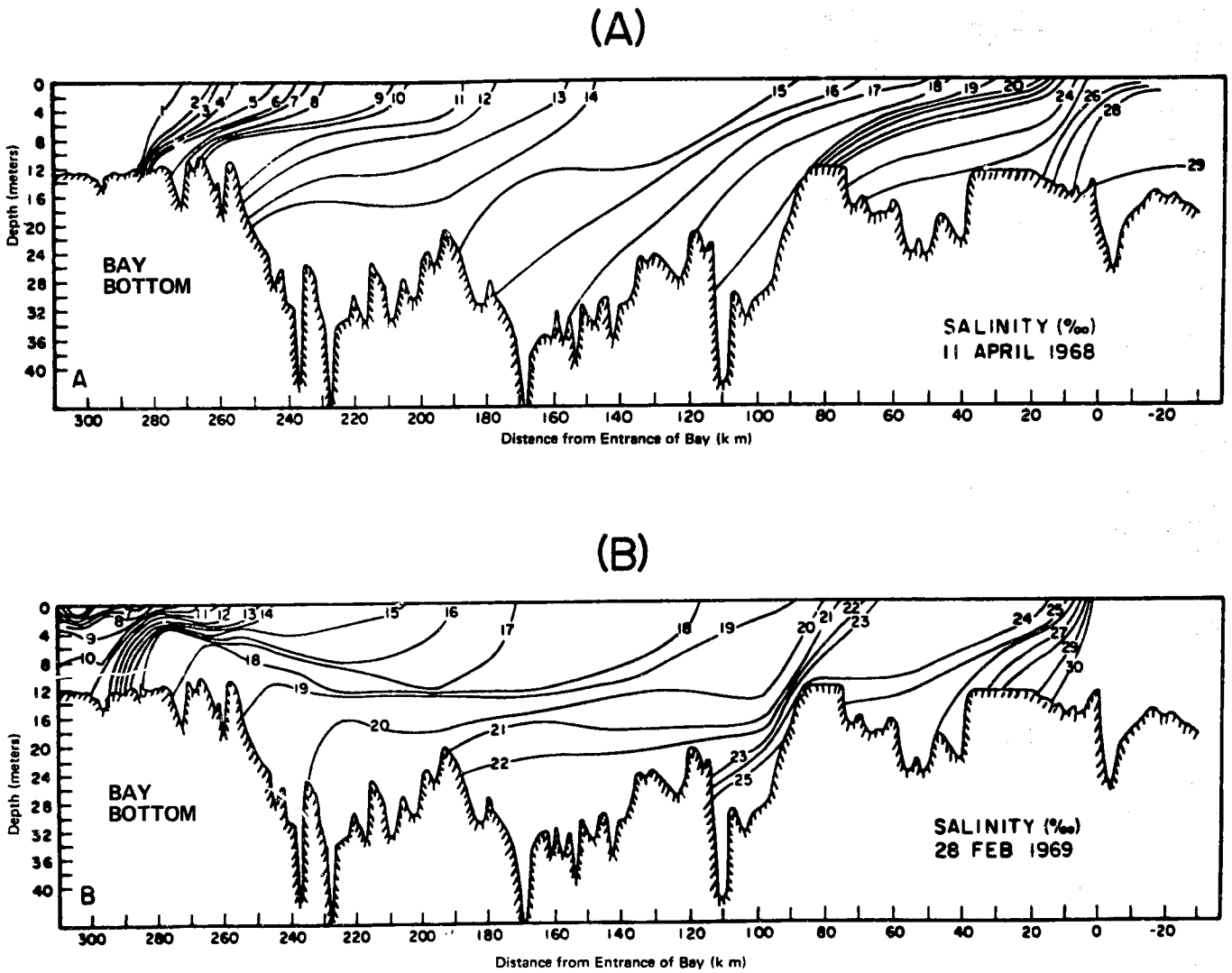

Figure 7.13. Longitudinal Salinity Distribution along Axis of Chesapeake Bay during (A) Period of High River Flow and (B) Period of Low to Moderate River Flow.

Water exchange and transport in the northern bay area also occurs through the Chesapeake and Delaware Canal, which is the sole direct link between the upper Chesapeake Bay and the Delaware River/Bay system.[27] Water flow in the canal can be erratic, with water transported from either bay through the entire length of the canal and discharged into the other bay. A net difference in mean tidal level between the Chesapeake and Delaware ends of the canal results in net transport of water to the eastward at an estimated average of 2450 cfs (~69 m^3/sec), in effect making the canal a tributary of the Delaware River.[28,29] Therefore, it seems probable that some radionculides of TMI-2 origin that enter the Chesapeake Bay could be expected to be transported into the canal and thereby into the Delaware River, where they could be accumulated by aquatic species present there (Appendix E).

Doses to Fishes and Other Biota in the Susquehanna River and Chesapeake Bay

Controlled Releases

The dose to fishes and other biota in the Susquehanna River is directly proportional to the concentration of radionuclides in the river water and to the time period over which the organisms are exposed. The concentration of radionuclide in the river is inversely related to the river flow during discharge of the radionuclide material. If it is assumed that the flow in the river is the same for a short-term release as for a long-term release, the annual average doses to fishes below York Haven Dam would be those listed in Table 7.38.

For a controlled release of treated effluent, the maximum nearsite dose to fish presented in Table 7.38 was calculated considering the dilution in the center channel, which is assumed to be carrying only 25 percent of the total river flow. As described in Appendix E, fishes important to the sport fishery in the Susquehanna River exhibit movement upstream, downstream and across the channel, so they would be exposed to an average concentration rather than the concentration at a particular point. The dose to fishes downstream of York Haven Dam assumes total mixing in all the waters of the Susquehanna River. The calculational methods used to make these dose estimates are described in WASH-1258.[30] Bioaccumulation factors are taken from Regulatory Guide 1.109.

Table 7.38. Average Annual Doses to Fishes in the Susquehanna River Downstream of York Haven Dam Resulting from Normal or Accidental Releases of Processed Water

Scenario	Dose (mrad)[a]	
	Near Site	Downstream
Normal Releases		
SDS/EPICOR II processing option[b]	6.0	1.5
SDS processing option[c]	99	25
Accidental Releases		
Tank failure - SDS processing option[d]	9.6	2.4

[a]A river dilution flow of 3150 cfs was used for nearsite estimates and 12,600 cfs for downstream estimates.

[b]See Table 7.25 for radionuclide source term.

[c]See Table 7.26 for radionuclide source term.

[d]See Table 7.28 for radionuclide source term.

Limited data are available from water and fish samples in the Susquehanna River for estimating the bioaccumulation factors of cesium and strontium. Bioaccumulation factors were estimated from background levels of Cs-137 and Sr-90 taken at control locations upstream from the site. Bioaccumulation factors for cesium were also independently estimated from the concentrations of Cs-134 in fish caught in Conowingo Pond resulting from the releases of Peach Bottom Units 2 and 3. Water concentrations in Conowingo Pond are estimated from known Cs-134 releases and an assumed river flow rate of 12,600 cfs. The results are tabulated in Table 7.39, and demonstrate that the bioaccumulation factors from Regulatory Guide 1.109, used in the staff's dose assessments, overestimate site-specific values.

Table 7.39. Bioaccumulation Factors Estimated from Measured Concentrations in Fish and Waters of the Susquehanna River

Radionuclide	Fish Concentration (pCi/g)	Water Concentration (pCi/mL)	Bioaccumulation Factor (pCi/g)/(pCi/mL)	Regulatory Guide 1.109 Value[a] (pCi/g)/(pCi/mL)
Sr-90	0.0217[b] (range 0.0004-0.075)	0.0023[c] (range 0.0003-0.0006)[b]	9.4[d] (range 1.3-250)	30
Cs-134	0.23[e]	0.00058[f]	400	2000
Cs-137	0.054[b] (range 0.01-0.145)[b]	0.0028[c] (range 0.002-0.02)[b]	19 (range 0.5-73)	2000

[a]Regulatory Guide 1.109, Rev. 1, p. 13.

[b]Values for control locations, taken from "Three Mile Island Nuclear Station, Radiological Environmental Monitoring Program, Annual Report 1979," Metropolitan Edison Company, April 1980, p. 13.

[c]Values reported in Section 3.4.1 of this document.

[d]Fish concentration divided by water concentration.

[e]Highest value reported in Conowingo Pond (Reference 24).

[f]Estimated average concentration based on 1.62 Ci released from Peach Bottom in Second Quarter, 1979 and 12,600 cfs average river flow.

Effects of Accidental Releases

The highest doses to aquatic organisms living in the Susquehanna River would be caused by the postulated accidental release of the contents of the processed water storage tank during an extended period of low river flow for which there would be no flow over Red Hill Dam. Table 3.1 demonstrates that statistically, river flows lower than that required to overtop Red Hill Dam and lasting for a half year could have a return period of less than two years. An extended period of non-flowing conditions in Red Hill Pond is therefore a reasonable assumption.

In order to demonstrate the upper limit of possible environmental damage to the fisheries of Red Hill Pond, the staff postulated a one-year period for which Susquehanna River flow was lower than 16,000 cfs. Neglecting exchange of water at the upstream end, the concentration of radionuclides in the pond was calculated by assuming that one-half the total contents of one of the 500,000-gallon tanks spilled into the approximately 65-million-gallon volume of the pond. The concentration of radionuclides in the pond was assumed to remain at this level for one year. The doses received by organisms (fish and fish eggs and larvae) living in the pond were calculated assuming the same bioaccumulation factors used for the controlled release scenario, and are presented in Table 7.40. For adult fish, average annual doses are presented in Table 7.40, and for eggs and larvae, doses over the 60-day developmental period from eggs to juviniles are present in the table. The 60-day developmental period from eggs to juviniles is considered to be the most radiosensitive period for fishes.

Table 7.40. Doses to Adult Fishes, Eggs, Embryos and Larvae in Red Hill Pond Caused by Failure of a Processed Water Storage Tank[a]

Accident[b] Type	Dose to Adult[c] Fishes - Rad	Dose to Eggs[d] or Larvae - Rad
Failure of processed water storage tank[e]	110	18

[a]Based on dilution of one-half contents of 500,000 gallon holding tank or evaporation pond with 65×10^6 gallon content of Red Hill Pond, stagnant for 1 year, no interchange with other Susquehanna River water. (See Table 7.28 for radionuclide inventory.)

[b]All dose estimates based on SDS processing option. Other options would result in lower doses.

[c]One-year exposure based on concentration in pond and Regulatory Guide 1.109 bioaccumulation factors.

[d]60-day exposure as above.

[e]Doses were estimated by multiplying the downstream accidental release value of Table 7.38 (2.4 mrad) by a factor to take into consideration the decreased dilution in the east channel under stagnant flow conditions (decreased dilution = flow in river in one year/volume in east channel = (12,600 cfs × 3.1×10^7 sec/yr × 7.48 gal/ft^3)/(6.5×10^7 gal) = (4.5×10^4). Value for eggs and larvae was estimated to be 1/6 that for mature fishes.

Doses to Fishes in Chesapeake Bay

Doses to fishes in the Chesapeake Bay would be greatest at the mouth of the Susquehanna River and would diminish with distance due to the combined effects of increased dilution and the markedly decreased biological accumulation factors for most radionuclides in saline water.[25] Annual average doses to fishes would be lower in the Chesapeake Bay than to those fishes residing in the Susquehanna River below York Haven Dam shown in Table 7.38. This conclusion is confirmed by radiological monitoring by the State of Maryland in Conowingo Pond which shows that the maximum concentration in fish due to radiocesium which has been released from the Peach Bottom nuclear plant is found in the pond itself, and that concentrations decrease markedly with distance down the bay.[31]

Radiosensitivity of Fish[32]

Figure 7.14 and Table 7.41 show that, as a group, fish are the most radiosensitive of the poikilothermic (coldblooded) aquatic animals, although some crustaceans exhibit similar sensitivities. The data available indicate that the average median lethal dose within 30 days of an acute radiation exposure ($LD_{50/30}$) is about 2,400 ± 1,500 rads. For larvae and embryos of fish the LD_{50} is considerably lower, about 200 rads.

Radiosensitivity of Crustaceans[32]

Crustaceans exhibit a range of radiation sensitivities covering approximately two orders of magnitude (Figure 7.14, Table 7.41). Various amphipods and shrimp, exhibit $LD_{50/30}$s of about 2,300 ± 1,400 rads--not significantly different from those of typical fishes. Some species of crabs exhibit $LD_{50/30}$s of about 23,000 ± 17,000 rads. Because the crabs live significantly longer and may not show lethal effects as soon as smaller crustaceans, a conservative estimate of the radiation dose inducing somatic effects on adult crustacea would be approximately the same as for fishes. Thus, an LD_{100} of 1500 rads is assumed for adults. This estimate considerably overestimates the impact on certain commercially and recreationally important species, such as the blue crab (*Callinectes sapidus*).

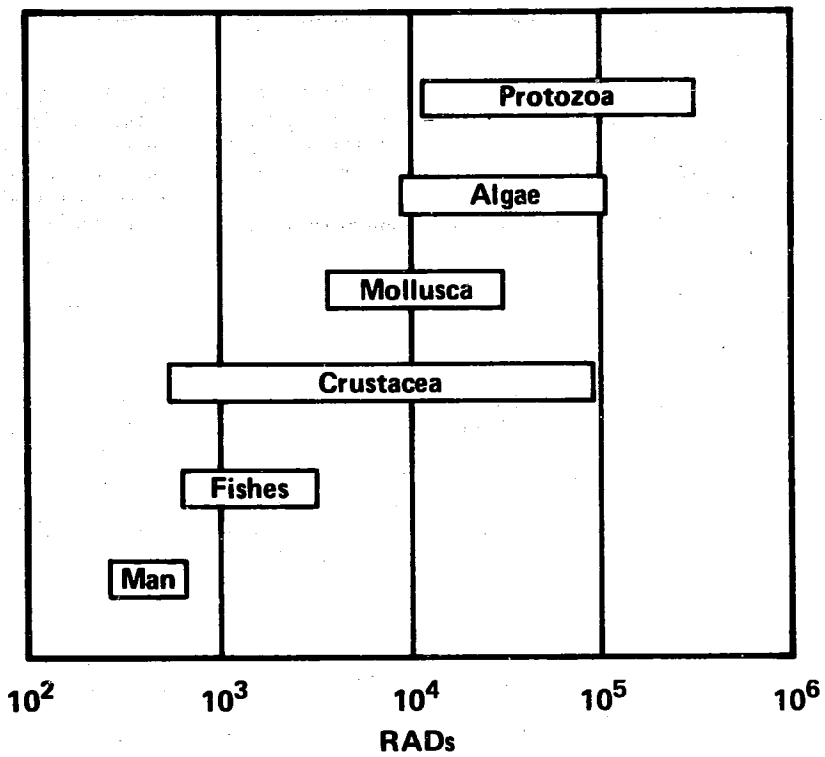

Figure 7.14. Tolerance of Aquatic Organisms to Radiation. (Relative tolerances of different groups of aquatic organisms to radiation in terms of dosages of radiation required to kill 50% of the exposed individuals in a given period of time.)

Table 7.41. Comparison of Acute Lethal Radiation Doses for Various Groups of Organisms[a]

Type of Organism	Dose (Kilorads)	Ranges of LDs
Bacteria	4.5 - 735	(LD_{90})
Blue green algae	<400 - >1200	(LD_{90})
Other algae	3 - 120	(LD_{50})
Protozoa	? - 600	(LD_{50})
Mollusks	20 - 109	($LD_{50/30}$)
Crustaceans	1.5 - 56.6	($LD_{50/30}$)
Fish	1.1 - 5.6	($LD_{50/30}$)

[a]I.L. Ophel et al., "Effects of Ionizing Radiation on Aquatic Organisms," in: "Effects of Ionizing Radiation on Aquatic Organisms and Ecosystem," Technical Report 172, IAEA, Vienna, 1976.

Radiosensitivity of Mollusks[32]

Mollusks generally exhibit a much lower radiosensitivity than fish, although there is comparability with some crustaceans, such as the blue crab (Figure 7.14, Table 7.41). The $LD_{50/30}$ for clams and oysters is about 100,000 rads. Like crustaceans, they have a much slower response to radiation than mammals. This results in the LD_{50} for long periods of time being much lower. For observation periods ranging from 50 to 90 days, the LD_{50} for adult mollusks is about 28,000 ± 23,000 rads. It is assumed that a reasonable $LD_{100/180}$ for adult mollusks would be about 50,000 rads.

Evaluation of Radiation Effects

Controlled Releases

The annual average downstream doses listed in Table 7.38 do not present a significant hazard to biota of the Susquehanna River and Chesapeake Bay, and are within levels generally established to be safe for humans. Although guidelines have not been established for acceptable limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for other species. Experience has shown that it is the maintenance of population stability that is crucial to the survival of a species, and species in most ecosystems suffer rather high mortality rates from natural causes. While the existence of extremely radiosensitive biota is possible, and whereas increased radiosensitivity in organisms may result from environmental interactions with other stresses (e.g., heat, biocides, etc.), no biota have have yet been discovered that show a sensitivity (in terms of increased morbidity or mortality) to radiation exposures as low as those expected from normal operations in the area surrounding and downstream from the Three Mile Island nuclear power plant. Furthermore, in all the plants for which an analysis of radiation exposure to biota other than man has been made, there have been no cases of exposures that can be considered significant in terms of harm to the species, or that approach the exposure limits to members of the public permitted by 10 CFR Part 20.[33] Since the BEIR-III Report[34] concluded that the evidence to date indicates that no other living organisms are very much more radiosensitive than man, no significantly radiological impact on populations of biota is expected as a result of the routine decontamination of this plant.

Accidental Releases

Doses to biota residing in Red Hill Pond for the postulated holding tank rupture or scenerio would be much higher than doses expected due to the controlled release scenerios. Comparison of the doses estimated for the SDS system (the system which gave the highest doses) in Table 7.40 to the ranges of senitivities estimated in Table 7.41 suggests that the accidental doses are well below levels that would significantly affect the adult fish, eggs, embryos or larvae in Red Hill Pond. Annual average doses to fishes downstream of Red Hill Dam for the case where the dam is overtopped would be 10 times less than the controlled release scenerio listed in Table 7.38, as for the SDS processing option, only about 20 percent of the total activity can be stored in the tank at a given time (one-half is assumed to immediately reach the river in an accident).

7.2.5.5 Psychological - Socioeconomic

The staff considers four methods for disposing of processed water to be practicable; these methods are: onsite storage, evaporation, offsite shipment, and discharge to the Susquehanna River. Although each would have negligible offsite radiological impacts under normal operating considerations, they can have varying psychological and socioeconomic impacts for different groups of people. Onsite storage of the processed water either in tanks or mixed with cement to form concrete blocks for a long period of time may generate chronic psychological distress and socioeconomic impacts among people living in communities near the station. In fact, the extent of potential distress and impact can be related to the length of time required for storage. The possible association between time of storage and distress reflects some individual's belief that there is an increasing probability of accidents over time and that long-term storage represents neither decontamination nor disposal. This latter point is especially important because some residents believe that storage, even long-term storage, is only a stage preparatory to the ultimate disposition of the radioactive waste. A strategy of long-term, onsite storage has the potential to prolong anxiety.

Release to the atmosphere through either forced or natural evaporation from the island are possible alternatives. As with the long-term storage option discussed above, the extent of anxiety may be positively related to the length of time required for evaporation. Moreover, the population impacted by the evaporation option is expected to be similar to that discussed above. Depending on processing scenarios, shipment of processed water offsite in solidified form would involve 640 truck shipments of 51,300 drums, or 340 shipments would be required if the processed water were shipped offsite in bulk liquid form.

Shipping the processed water to an offsite disposal area can be generally less stressful locally than either the long-term storage or evaporation options because it removes the believed source of threat from the TMI area. However, those living along or in proximately to the truck route may be subjected to stress. Such potential stress would be related to increased awareness of residents, and the perceived possibility of on-the-road accidents. The transportation of waste, particularly through Middletown where population density is higher and people live closer to the roadway than other locations in the local impact area, can stimulate greater stress. It is the staff's judgment that the marketability of residential property abutting the route through Middletown conceivably could be adversely impacted during the extended period of shipments.

The final disposal alternative considered practicable by the staff is the controlled release of processed water to the Susquehanna River. The staff estimates that a minimum of 110 days would be required to release the entire inventory of processed water. Although disposal to the Susquehanna would present negligible public health hazards, some residents of local and downriver (including Chesapeake Bay) communities find this alternative both threatening and distressful. In short, some members of the public do not believe [illegible] even the low concentration of tritium and even lower quantities of other radionuclides remaining after dilution of the processed water are safe. The staff concludes that these concerns would be concentrated among those who use the Susquehanna River/Chesapeake Bay system as a water supply, as a source of income and enjoyment, and those who consume the Bay's resources.

Although models to predict behavior from attitudes are difficult to specify under the best of circumstances, the staff has concluded that disposal of processed water into the Susquehanna River has the potential for producing widespread socioeconomic impacts. These impacts may include the following:

(1) temporary avoidance of drinking water originating from either the Susquehanna River or the Chesapeake Bay and the substitution of other liquids;

(2) losses to recreation as people cease their use of surface water for swimming and boating;

(3) losses to river and bay recreational fishing and hunting of waterfowl;

(4) avoidance of game products;

(5) consumer avoidance of river, bay, and ocean shellfish and finfish; and

(6) economic losses to watermen, onshore support activities, seafood processors, restaurants, and retail and service outlets associated with Bay products and activities (bait and tackle stores, motels, charter boat captains, etc.).

Potentially, the most consequential economic impacts of the Susquehanna River disposal options for processed water can result if widespread consumer avoidance of Chesapeake Bay resources occurs.* Consumer avoidance would be related to a number of factors including general attitudes toward radiation. There is some evidence to suggest that the public perceives nuclear power to be more dangerous than many other technologies or activities including alcoholic beverages, general aviation, motor vehicles, handguns, and smoking. This belief is said to be derived from the expectation that nuclear power activities can produce an enormous number of deaths and threaten the survival of the human race (pp. 25-34 of Reference 35).

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*"Consumer" in this context refers not only to customers of retail establishments but also to to watermen, wholesalers, hunters.

A second factor which would underlie consumer avoidance behavior is the confusion and distrust in a segment of the public that believes the risk of nuclear power to be considerably greater than the experts' assessments indicate (p. 15 of Ref. 35). A survey of residents within 20 miles of TMI to determine levels of trust in authorities indicated a leaning toward distrust.[36] Although the tendency toward distrust indicated in April 1979 had declined by August 1979, the tendency was still for opinions to lean on the average toward distrust and appears to have been above the national average. Although data are not available for a larger area around TMI, the staff has assumed that the population, particularly those downriver of TMI who might be indirectly impacted by the disposal of processed water in the Susquehanna, would share in levels of distrust characteristic of those living within 20 miles of TMI.

The potential for consumer avoidance and economic loss would also be related to two characteristics of seafood and the seafood industry. As indicated in Section 3.6.1.3, shellfish represents approximately 90% of the landed value from the Chesapeake fishery. Although accurate data are not available, a substantial portion of the total shellfish harvest is believed to be consumed in restaurants. Those who have a long familiarity with seafood marketing consider fresh shellfish a "luxury" item of food consumption for which substitutions are easily made.[37] Secondly, the industry is subjected to competitive pressures not only from fresh products originating in Gulf of Mexico and other waters but also from frozen, processed products. Those familiar with the Chesapeake seafood industry believe that sales losses are difficult, if not impossible, to recapture from competition.[37]

Finally, historical evidence strongly points to a link between water quality and consumer demand for fish and shellfish. Consumer scepticism concerning the safety of seafood which is served uncooked rises with outbreaks of hepatitis following consumption of these products.[38] Sport fisheries, in particular, are adversely affected by highly publicized examples of "contaminated" waters; such examples include mercury and PCB incidents in Japan; Kepone in the James River and lower Chesapeake Bay, Virginia; PCB and DDT in the Hudson River, New York and throughout the Great Lakes; Mirex in Lake Ontario, New York; mercury in Lake St. Claire, Lake Erie and mainstream TVA lakes[37] and natural phenomena such as red tides. In the months following the accident at TMI, local fishing activity in the vicinity of the plant declined in response to actual and believed changes in water quality levels (see Section 3.6.2.9).

Several experts in the field of marine economics and marketing suggested that the media plays an important role in determining the public's perception of water quality changes and, ultimately, consumer behavior.[40] If the media lacks adequate knowledge and context within which to evaluate misleading statements, such statements could be passed on to the public as fact. The end results are the generation of fears and consumer avoidance of seafood products. A concern for media reporting and its relationship to general public fears underlies the concern of the State of Maryland.[41]

The most recent and well-publicized incident of water contamination involves the insecticide kepone. During the 1970s, traces of kepone in amounts exceeding Federal limits were found in blue crabs, oysters, bluefish, and in several other species of finfish in the James, York, and Rappahannock Rivers in Virginia. Rather than risk the distribution of contaminated products, Federal and State officials closed the James River south of Richmond and adjacent portions of the lower Chesapeake Bay starting in December 1979 to the commercial harvesting of all seafood; recreational fishing was only permittted on a catch-and-release basis.[40] This ban was lifted for most species in November 1980. Prior to and following fishing restrictions, numerous newspaper articles emphasized the possibility of seafood contamination.[41] Efforts were made by Maryland officials to publicly draw a distinction between Maryland seafood and fish--which contained levels of kepone below Federal limits--and Virginia products.[42]

Although the direct and indirect economic effects of kepone contamination apparently have not been systematically analyzed, several sources of information indicate widespread impact. In Virginia, consumers refused to buy bluefish, restaurants refused to serve them, and fishermen had no market for the fish. Purchases of bluefish were made only when the restaurant or retailer could assure his customer that the fish was caught in the Atlantic Ocean or in North Carolina waters.[43] As a result of incomplete information and extensive news coverage, the sale of oysters harvested from areas which were completely isolated from James River was adversely affected in Baltimore's wholesale fish market, some 200 miles north of the James River.* During 1976, sport

*Researchers (see Ref. 44) calculated a 5% loss of oyster sales in Baltimore during a period when media attention was focused on kepone contamination in Virginia. This loss occurred in a market 200 miles from the James River and to a product which is unrelated to the James River oyster.

fishery charter boat captains from the lower Maryland portion of the Chesapeake Bay, including Chesapeake Beach (100 miles north of the James River), reported many trip cancellations due to angler concern with contaminated bluefish.[45] Although trip cancellations among charter boats based around the Annapolis area were minimal, a small number of captains operating off the New Jersey coast implicated kepone contamination of fish as the reason that some fishermen were reluctant to charter trips.[46]

The influence of contamination and media coverage on recreation has been demonstrated in a study which has relevance for waterfowl hunting in the Chesapeake Bay.[47] In examining the effects of mercury contamination on pheasant hunting, the researchers concluded that knowledge of the presence and dangers of mercury in pheasants helped to explain much of the loss in Oregon pheasant hunters during a two-year period.

Although the staff is unable to specify the quantitative or temporal extent of potential economic losses to Chesapeake Bay activities as a result of disposing tritiated water in the Susquehanna River, three conclusions appear to be justified at this time. First, changes in the perceived quality of water, whether or not such changes represent an actual health hazard to either the fauna or human consumers, is linked to altered patterns of consumption, most notably consumer avoidance. Second, the accuracy, tone, and quantity of media coverage will exert a powerful influence on how the public perceives changes in water quality and, ultimately, on human behavior and consumption patterns. Finally, because imperfect or limited information creates uncertainty, the demand for products harvested from areas not affected, even remotely, could also decline; that is, consumer avoidance could impact the demand for fish and waterfowl from areas abutting or beyond the Chesapeake region.

The State of Maryland has informed the NRC that it will be undertaking a comprehensive study of potential economic losses to Chesapeake Bay activities which would result from the alternative of disposing of TMI-2 processed water by dilution and controlled release to the Susquehanna River which is expected to be completed in one year. The NRC staff is of the opinion that at least until such a study is completed and the State of Maryland provides the results to the public and the NRC staff, no decision should be made regarding disposition of the TMI-2 processed accident water. This assumes no emergency arises which would require disposition of this water in the interim. The NRC staff believes any such disposal actions taken in advance of completing this study would be premature and without the benefit of information on societal and socioeconomic factors important to the decision-making process.* However, processing of the water to remove and immobilize the hazardous radionuclides should, in the staff's opinion, proceed as expeditiously as safety permits. No options for disposition of the processed water (as well as the separated radionuclides) are foreclosed by the expeditious treatment of the highly radioactive accident water in the reactor building sump and reactor coolant system.

As discussed elsewhere in this chapter, adequate onsite storage capacity already exists. In consideration of this, the licensee has recently announced that no proposal for processed water disposal will be made to the NRC prior to 1982.

7.2.6 Liquid Disposal Costs

Table 7.42 provides the estimated liquid disposal costs for the alternatives presented. The long-term onsite storage alternatives include the costs for up to 25 years of storage. Ultimate disposal after the 25-year storage period have not been estimated.

The costs for the offsite disposal alternatives include transportation costs to the final disposal site.

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*The basic regulatory criteria for discharge of radioactive liquids is provided in 10 CFR Part 20. Specifically, paragraph 20.1.c states that licensees should maintain releases of radioactive materials to "as low as is reasonably achievable" (ALARA) levels and further defines the term ALRA as meaning "....as low as reasonably achievable taking into account the state of technology and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations, and in relation to the utilization of atomic energy in the public interest."

Table 7.42. Disposal Costs for TMI-2 Liquids
(thousands of dollars)

Disposal Alternative Considered	High		Low
1. Long-term onsite storage[a]			
(a) Stored in tanks		5,600	
(b) Immobilized as concrete slabs	2,800		2,300
2. Discharge to the environs			
(a) River discharge	150		100
(b) Evaporation ponds		500	
(c) Forced evaporation		250	
3. Offsite disposal			
(a) Immobilized and shipped[b]	7,400		5,400
(b) Immobilized and shipped[c]	5,600		4,100
(c) Deep well injection	5,100		3,700
(d) Ocean disposal	10,500		7,700
4. Onsite disposal			
(a) SLB trenches	1,900		1,400
(b) Underground injection		250	

[a]Assumes 25-year storage.

[b]Richland, Washington, destination.

[c]West Valley, New York, destination.

While a range of cost estimates is presented, the range is intended to include the uncertainty in the actual volume of processed water to be disposed of. Where a single cost value is presented, the costs have been considered to be relatively insensitive to the actual volume.

7.2.7 Comparison of Processed Water Disposal Alternatives

The alternatives discussed above were compared on a relative basis considering time to complete water disposal, radionuclide release pathways, potential regulatory obstacles, offsite doses and costs. Table 7.43 summarizes these parameters. This table numbers alternatives from 1 to 10 and these designations are referred to in the discussions presented below. The conclusions that can be drawn from this summary are as follows:

- Time to Complete Water Disposal

  To compare onsite and offsite alternatives, time to complete is the length of time required to either (1) reduce concentrations to innocuous levels while onsite or (2) perform the work necessary to either discharge or transport processed water from the site, including any necessary construction work required to implement the alternative.

  As discussed above, alternatives 1 and 2 are not practicable due to the long time periods involved. Four of the other alternatives (3, 4, 5, and 6) could take 5 years to complete due to regulatory obstacles and, thus, may not be suitable.

  Three of the remaining four alternatives (7, 8 and 10) can be readily completed while the fourth (alternative 9) is completed over a period of several years.

Table 7.43. Comparison of Alternatives for Disposal of Processed Water

Disposal Alternatives	Years to Complete	Release Pathways: To Atmosphere	To River	To Land	To Subsurface Water	To Ocean	Potential Regulatory Obstacle: NRC Licensing	EPA Permitting	State/Local	Offsite Doses[d] Person-rem	Cost[c] (10^3)	Permanent Disposition
Long-Term Onsite Storage												
1. In liquid tanks[a]	200	+								NA	5600	No
2. As concrete slabs	200	+		+						30[e]	2300	No
Onsite Disposal												
3. SLB trenches	5[b]			+			+		+	NA	1400	Yes
4. Underground injection	5[b]				+		+	+	+	NA	250	Yes
Offsite Disposal												
5. Deep well injection	5[b]				+		+	+	+	NA	3700	Yes
6. Ocean disposal	5[b]					+		+		NA	7700	Yes
7. SLB facility	1			+						NA	4100	Yes
Discharge to Environs												
8. Release to river	< 1		+							30	100	Yes
9. Natural evaporation	1	+	+							30	500	Yes
10. Forced evaporation	< 1	+	+							30	250	Yes

[a]After storage alternatives 2 through 10 are applicable.

[b]Based on potential licensing and permitting delays.

[c]Based on the low cost values in Table 7.42.

[d]Based on the SDS/EPICOR II process effluent.

[e]Based on loss of all tritium in the concrete slab.

- Release Pathways

All alternatives lead to releases to the environs and the major difference among them is the pathway(s) through which radionuclides are released. As shown in Table 7.43, alternatives (3, 4 5 6, 7 and 8) have a single pathway, while others (2, 9 and 10) have multiple pathways. The onsite concrete slab alternative (2) releases tritium to the atmosphere by diffusion through the concrete for several years after placement in the storage areas. The slabs are later placed in a shallow land burial facility. The forced evaporation alternative (10) simultaneously releases radionuclides to the atmosphere and the river, while the natural evaporation alternative (9) releases tritium to the atmosphere. The impacts of these releases depend on the pathway and the time period over which the release is made.

- Potential Regulatory Obstacles

As discussed previously, the onsite disposal alternatives (3 and 4) are not considered suitable because they would convert TMI-2 into a permanent waste repository and present unresolved regulatory issues. The offsite bulk liquid alternatives (5 and 6) could present different regulatory obstacles since new fields of regulation would be involved. It may not be practicable to proceed with these alternatives due to these regulatory obstacles.

The other alternatives (1, 2, 7, 8, 9, and 10) can be implemented in the framework of existing regulations.

- Offsite Doses

Offsite doses, based on the SDS/EPICOR II process effluent, were only determined for alternatives which affect the population within 50 miles of the TMI-2 site. Therefore, the doses arising from offsite disposal alternatives were not determined and are not shown on the table. However, if the processed water is transported to another location and either evaporated or discharged to a river, the doses from these operations would be comparable to those shown for similar onsite activities.

- Cost

The costs shown on Table 7.42 include the cost of any new facilities that are needed and the operating cost to perform the work required to complete the alternative. None of the costs shown include the cost of licensing for the various alternatives.

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8. SOLID WASTE PACKAGING AND HANDLING

The activities described in Sections 5, 6, and 7 result in the generation of radioactive materials that must be packaged and disposed of as radioactive waste. Most of this radioactive material will be low-activity radioactive waste that can be disposed of at licensed shallow land burial facilities used for disposal of wastes from other nuclear power plants. Some of these materials may be transuranic (TRU) radioactive wastes, and others will have characteristics, such as high specific activity, that warrant their handling and disposal as high-level waste on a special case-by-case basis. In the staff's judgment, the TRU and high-specific-activity wastes will need to be transported to federal government (U.S. Department of Energy) facilities for treatment and disposal. In this section, wastes are discussed in terms of their characteristics, the alternatives considered for their packaging and handling, and the impact of these operations.

The wastes that could be generated have been divided into four major categories that are related to their source or physical form:

Process Solids - The radioactive wastes generated from treatment of liquid wastes. These materials consist of accident sludge, spent filter cartridges, expended ion-exchange media in the form of zeolites* and organic resins, and evaporator bottoms. Bituminized materials generated from once-through treatment of liquid wastes also are included in this category.

Chemical Decontamination Solutions - The liquid radioactive wastes generated from "hands-on" decontamination activities and decontamination of systems that have not been treated to generate process solids.

Solid Materials - The dry radioactive waste generated during decontamination and defueling operations. These materials consist of trash, contaminated equipment and material, and irradiated, activated hardware.

Damaged Fuel Assemblies and Core Debris - The radioactive materials removed from the reactor vessel during defueling operations.

The waste types within each of these categories are discussed separately in Sections 8.1 through 8.4.

8.1 PROCESS SOLID WASTES

Packaging and handling of contaminated sludges generated by the accident and of process solid wastes that could arise from treatment of accident water and decontamination liquids are discussed in this section. These solid wastes would include accident sludges and, depending on the liquid processing alternatives implemented (see Sec. 7.1), filter cartridge assemblies, ion-exchange media (inorganic zeolites and organic resins), evaporator bottoms, and bituminized materials. The waste forms that could be generated from treatment of the various liquid waste sources are listed in Table 8.1 and discussed in detail below.

8.1.1 Status and Specific Considerations

The status of the process solid wastes generated through September 22, 1980, and the practices used to package, handle, and store these wastes are summarized in this subsection. The amounts of the various types of such wastes that could be generated by processing of liquid wastes and decontamination of various systems are estimated.

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*Other inorganic ion-exchange media may be used with or in place of zeolites.

Table 8.1. Process Solid Waste Summary

Source of Treated Liquid Waste	Curie Inventory in Untreated Liquid[a]		Process Solid Waste Forms[b,c]					
	Minimum	Maximum	Filters	Zeolites	Organic Resins	Evaporator Bottoms	Bitumen	Sludge
1. AFHB accident water	55,000[d]	55,000[d]	X	X[e]	X[e]			X
2. Reactor building sump water	500,000	500,000	X	X	X			X
3. RCS water	20,000	20,000		X	X			
4. RCS flush & drain water	20,000	100,000	X	X	X			
5. AFHB/reactor building decontamination solutions								
(a) Aqueous	90	90			X			
(b) Chemical	70	70				X	X	
6. RCS decontamination solutions								
(a) Aqueous[f]	2,000	20,000	X	X	X			
(b) Chemical[f]	2,000	20,000				X	X	
Total	600,000	700,000						

[a]Exclusive of H-3 and noble gases--rounded to two significant figures.

[b]Waste form combinations are alternative-dependent--see Section 7.1.3.

[c]X indicates process solid waste form could be generated and is considered.

[d]Curies removed by system through September 22, 1980.

[e]Some liners contain zeolites mixed with organic resins.

[f]Mutually exclusive alternatives; one waste form will be produced, not both.

8.1.1.1 Efforts to Date

The work performed through September 22, 1980, involved processing of AFHB accident water with the EPICOR II system and resulted in the generation of expended ion-exchange media. The characteristics of these media are summarized in Table 8.2. All the liners* listed in the table are being stored onsite in the EPICOR II storage facility (see Appendix Q for description). In compliance with the Commission's order of October 1979 to solidify these wastes prior to shipment, the licensee has been investigating the use of cement.[1] In addition, the licensee has been investigating alternatives to solidification of these wastes onsite.[2]

As contaminated liquids are passed through the EPICOR II system and the ion-exchange media become loaded with radioactive contaminants removed from the liquids, the expended media are replaced with new liners. The steps involved in the removal of expended liners from the EPICOR II system and their subsequent handling are as follows:

- Disconnecting the connections between the resin liners and the EPICOR II system
- Capping liner inlet, outlet and vent connections
- Transfer of the expended resin liner to a bottom-loaded, bell-type transfer cask mounted on an overhead monorail (mixed-bed third-stage liners are handled unshielded)
- Placement of the transfer cask liner into a top-loaded, reinforced-concrete shield mounted on a flatbed truck
- Transport (within the site boundaries) of the doubly shielded liner to the onsite storage facility
- Removal of the bell transfer cask from the truck-mounted shield with a crane
- Movement of the bell transfer cask to a position above the storage module
- Placement of the liner within a shielded storage module
- Closure of the storage module with a concrete shield plug.

The radiation exposure of personnel involved in transfer of these liners to the storage area has been about 6 person-rem.[3] This corresponds to about 85 person-mrem per liner transferred.

8.1.1.2 Projected Requirements

The different types of process solid wastes that could be generated and the curies of radioactivity that would be removed from various liquid waste streams and deposited on these solids were listed in Table 8.1.

To estimate the quantities of process solid wastes that could be generated, the staff defined best- and worst-case conditions that bound waste quantities as follows:

- Best-Case Conditions. Liquid wastes are assumed to be treated using the methods that generate minimum waste volumes. Where specific activity or curie content per unit of the process solid waste type could be variable, the maximum curie content is assumed. (The greater the amount of radioactive contaminants allowed to be accumulated on a single batch of resins, for example, the smaller the total volume of resins that will be required to remove a given amount of contaminants from the liquids.) These minimum volume projections for process solid waste generation are shown in Table 8.3.

- Worst-Case Conditions. Liquids are assumed to be treated using the methods that generate maximum waste volumes. Where specific activity or curie content per unit of the process solid waste type could be variable, the minimum curie content was assumed. (The smaller the amount of radioactive contaminants allowed to accumulate on a single batch of resins, for example, the greater the total volume of resins that will be required to remove a given amount of contaminants from the liquids.) These maximum volume projections for process solid waste generation are shown in Table 8.4.

The process solid waste volumes shown in Tables 8.3 and 8.4 were used to evaluate alternatives and characterize process solid wastes throughout this section.

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*The "liners" are the containers housing the ion-exchange material.

Table 8.2. Summary of EPICOR II Waste Packages[a]

Package	Total Ion-Exchange Media Volume (ft^3)	Liner[b] Size	Number of Liners	Range of Curie Loadings per Liner	Maximum Liner Radiation Level (R/hr)
Prefilter liner (30 ft^3/liner)	1470[c]	4'D × 4'H	49	100 to 1300[d]	1500[e]
Second-stage liner (30 ft^3/liner)	420	4'D × 4'H	14	1 to 60[f]	50
Third-stage liner (130 ft^3/liner)	780	6'D × 6'H	6	1 to 6	5

[a]These media are currently being stored in the dewatered condition.

[b]Liners are carbon steel.

[c]May be mixture of organic and inorganic.

[d]30 liners are in 1200 to 1300 Ci range, 14 liners are in the 800 to 1200 Ci range, 2 liners are in the 400 to 800 Ci range, 3 liners are in the 100 to 200 Ci range. These loadings do not include the daughter products of Cs and Sr which would increase the total Ci's by 50%.

[e]Maximum hot-spot radiation level on liners in 1200 to 1300 Ci range. Average radiation levels in R/hr are about 0.8 times curie content.

[f]Five liners are in the 10 to 60 Ci range, remainder are below 10 Ci.

Source: "Data Handout-TMI-2," as available to the TMI Working Group Meeting held at TMI, Middletown, PA, September 23, 1980.

8.1.2 Alternative Methods Considered

The techniques used to treat, package, and handle each of the various types of process solid waste materials that could be generated depend on the physical characteristics of the waste form and its specific activity or curie content.

8.1.2.1 Accident Sludge

The volume of accident sludge that could be generated from cleanup of the AFHB and reactor building is estimated to be about 250 ft^3. The estimated specific activity of this sludge could range from 1.6 Ci/ft^3 in the reactor building to 60 Ci/ft^3 in the AFHB.[4] Therefore, the packaging and handling operations for sludge will require the use of remote techniques.

Alternatives considered for sludge are dewatering or immobilization with cement or with vinyl ester styrene. In each case, the sludge could be packaged in 55-gallon drums or large liners. The maximum waste package production would involve immobilization with cement or vinyl ester styrene and packaging in 55-gallon drums. The minimum waste package production would involve packaging dewatered sludge in 4-ft-diameter by 4-ft-high liners with a capacity of 45 ft^3. The quantities, container characteristics, and maximum radiation levels of packaged sludge under these maximum and minimum conditions are given in Table 8.5. To date, no proposal has been made by the licensee on whether to dewater or immobilize the sludge.

The high specific activity of the AFHB sludge will lead to container radiation levels up to 2500 R/hr. Since these levels are in excess of those experienced to date with EPICOR II prefilter liners of the same size, they could present handling and personnel radiation exposure problems. Therefore, drums are the preferred container for AFHB sludge packaging.

Table 8.3. Minimum Generation of Process Solid Waste

Source of Treated Liquid Waste	Treatment Alternative[b]	Process Solid Waste Forms and Unpackaged Volume (ft^3)[a]					
		Filters	Zeolites	Organic Resins	Evaporator Bottoms	Bitumen	Sludge[c]
1. AFHB accident water	EPICOR II system	NA	NA[d]	2600[d]	NA	NA	150
2. Reactor building water	SDS/Modified SDS	110	50[e]	200	NA	NA	100
3. RCS water	SDS/Modified SDS	NA	20	200	NA	NA	NA
4. RCS flush and drain water[f]	SDS/Modified SDS	360	20	200	NA	NA	NA
5. AFHB/reactor building decontamination solutions (Chemical)	Evaporator/Resin	NA	NA	100	200	NA	NA
6. RCS decontamination solutions (Aqueous)[g]	CAN DECON/SDS Type	20	10	100	NA	NA	NA

[a]Volumes rounded to nearest 10 ft^3.

[b]Alternatives discussed in Section 7.1 and defined in Appendix G.

[c]Sludge generation is independent of treatment alternative.

[d]May be mixture of organic and inorganic.

[e]Volume shown based on specific activity up to 15,000 Ci/ft^3, the maximum loading considered for zeolites used in reactor building accident water treatment.

[f]Based on removal of 20,000 Ci.

[g]Based on removal of 2000 Ci.

Table 8.4. Maximum Generation of Process Solid Waste

Source of Treated Liquid Waste	Treatment Alternative[b]	Process Solid Waste Forms and Unpackaged Volume (ft^3)[a]					
		Filters	Zeolites[c]	Organic Resins	Evaporator Bottoms	Bitumen	Sludge[d]
1. AFHB accident water	EPICOR II	NA	NA[e]	2600[e]	NA	NA	150
2. Reactor building sump water	SDS/EPICOR II	110	430	390	NA	NA	100
3. RCS water	Modified EPICOR II	NA	50	540	NA	NA	NA
4. RCS flush and drain water[f]	Modified EPICOR II	450	100	1980	NA	NA	NA
5. AFHB/reactor building chemical decontamination solutions	Bituminization	NA	NA	150	NA	50	NA
6. RCS chemical decontamination solutions[g]	Evaporator/Resin	NA	NA	250	7000	NA	NA

[a]Volumes are rounded to nearest 10 ft^3.

[b]Alternatives discussed in Section 7.1 and defined in Appendix G.

[c]Volume based on specific activity of up to 1250 Ci/ft^3, the minimum considered for zeolites.

[d]Sludge quantities are independent of treatment alternatives.

[e]May be mixture of organic and inorganic.

[f]Based on removal of 100,000 Ci.

[g]Based on removal of 20,000 Ci.

Table 8.5. Accident Sludge Packages--Minimum and Maximum Waste Generation

Factor	AFHB Sludge		Reactor Building Sludge	
	Minimum	Maximum	Minimum	Maximum
Original volume (ft^3)	150	150	100	100
Process used	Dewater	Immobilize	Dewater	Immobilize
Volume increase factor	1	1.65	1	1.65
Container	50 ft^3[a]	Drum	50 ft^3[a]	Drum
Maximum Ci/package	2700	250	72	7
Maximum surface radiation level (R/hr)[b]	2200	200	60	6
Number of packages	∿ 3	36	∿ 2	∿ 24

[a]50-ft^3 liner with 45-ft^3 capacity.

[b]Based on 0.8 R/hr per curie.

8.1.2.2 Zeolites

Zeolites are effective ion-exchange media for removal of cesium. For the alternative treatment systems under consideration, zeolite ion-exchange vessels are used as the first stage in a multistage treatment system. Each vessel is made of stainless steel, has an internal volume of 10 ft^3 and is loaded with between 8 ft^3 and 10 ft^3 of zeolite. The amount of radioactivity that can be removed by zeolites is extremely high; specific activities in excess of 15,000 Ci/ft^3 of zeolite can be achieved without radiation damage to the ion-exchange material. Thus, the limits on the radioactivity in the zeolite vessels are based on shielding requirements and remote handling techniques rather than on ion-exchange media stability. It is expected that these zeolite ion-exchange vessels will be loaded with between 10,000 and 120,000 Ci each. The higher loadings will minimize container handling and resultant exposure, as well as the number of shipments and volume to be disposed. Even with the lower loadings of radioactive materials (predominated by 30-year half-life radioisotopes), the zeolite containers cannot be handled or disposed of as low-level radioactive waste. Present intentions are for interim storage of zeolite ion-exchange vessels at the plant site followed by transportation to a federal government facility for treatment and eventual disposal.[5]

Currently, the primary alternative for managing similar zeolites produced at federal government facilities, which also could be applied to zeolites from TMI, is to comingle the material with high-level waste sludge being stored in tanks pending the availability of facilities to immobilize the waste. Alternatively, the TMI-2 materials could be processed separately. Eventual immobilization in a borosilicate glass is now considered to be the preferred approach, although developmental work on ceramics continues. After immobilization and packaging, the waste would be stored until a geologic repository is developed.

Some developmental work has been done on elution of Cs-137 from zeolite resins, with subsequent storage of the eluate for eventual immobilization. In the event that waste forms containing zeolites are unacceptable for disposal in a high-level-waste repository, elution of the activity from zeolites may be required. However, applicability of this procedure to specific conditions of the TMI zeolite would have to be verified before this limited treatment option could be considered.

The volume of zeolites that could be generated depends on the treatment system selected and the amount of radioactivity placed on zeolites in each ion-exchange vessel. The higher the radioactivity loading per vessel, the lower the volume of packaged waste. The quantities and characteristics of zeolites that could be generated under best- and worst-case conditions are summarized in Table 8.6 and the conditions of maximum and minimum waste production are outlined below:

- Maximum Waste Production. Systems using zeolite ion-exchange vessels are assumed to be used for treatment of reactor building sump water, primary system water, reactor cooling system flush and drain liquid, and CAN DECON liquid waste. Each vessel would be loaded with 10,000 Ci of radioactive contaminants or about 1250 Ci/ft^3, the minimum loading desirable.

- Minimum Waste Production. Systems using zeolite ion-exchange vessels would be used for treatment of reactor building sump liquid, primary system water, and RCS flush and drain liquid. Each vessel used for reactor building sump water would be loaded with 120,000 Ci, or about 15,000 Ci/ft^3, the maximum loading considered. Vessels used to process other water would be loaded with 10,000 Ci, or 1250 Ci/ft^3.

Since no proposal has been made on the use of zeolites rather than other treatment methods, the quantities and characteristics shown in Table 8.6 bound the volumes of zeolite wastes that could be generated under the stated assumptions.

Table 8.6. Generation of Zeolite Waste--Minimum and Maximum Alternatives

Source of Treated Liquid Waste	Minimum Generation		Maximum Generation	
	Volume (ft^3)[a]	Curies[b]	Volume (ft^3)[a]	Curies[b]
1. Reactor building sump water[c]	48	500,000	432	500,000
2. RCS water[d]				
SDS[e]			48	20,000
Modified EPICOR II	16	20,000		
3. RCS flush and drain[d]				
SDS[e]			96	100,000
Modified EPICOR II	16	20,000		
4. RCS decontamination solutions[f]	8	2,000	16	20,000
Total	88	542,000	592	640,000

[a]Based on gross curies removed and estimated zeolite loading per cubic foot.

[b]Total estimated curie content in source. Zeolites will remove 98+ percent.

[c]Range based on minimum zeolite loading of 1250 Ci/ft^3 and maximum zeolite loading of 15,000 Ci/ft^3.

[d]Based on zeolite loading of 1250 Ci/ft^3.

[e]All versions of zeolite-based systems considered for these liquids have the same zeolite loading of 1250 Ci/ft^3.

[f]CAN DECON alternative only.

8.1.2.3 Organic Resins

Organic resins are used as the primary ion-exchange media in the EPICOR II system. These materials also are used in second- and third-stage ion-exchange vessels for some of the other alternative treatment methods being considered. Generally, these second- and third-stage ion-exchange vessels remove relatively small quantities of radionuclides, and the specific activity of the organic resin wastes is well below that of first-stage ion-exchange vessels, such as the EPICOR II prefilter liners and the SDS zeolite liners.

High-Specific-Activity Organic Resins

The quantities and characteristics of the organic resins generated from EPICOR II system treatment of AFHB accident water are described in Table 8.2. As shown in that table, first-stage prefilter liners contain about 30 ft^3 of organic resin with a specific activity of 10 to 40 Ci/ft^3. This activity level, coupled with the cesium and strontium content of these ion-exchange materials, precludes their being handled and disposed of as routine low-level radioactive waste. Forty-six of these 49 prefilter liners fall into this category of high-specific-activity waste. The characteristics of the 49 prefilter liners and of additional prefilter liners that could be generated if the EPICOR II system is used to treat primary system water are given in Table 8.7. The EPICOR II resins currently are being stored in carbon steel vessels in the dewatered condition in shielded modules at the plant. As indicated in Section 7.1, the EPICOR II system is being considered for treatment of RCS water and RCS flush and drain water. Since the gross curie content of these liners would be comparable to that of liners generated by treatment of AFHB liquids, the liners would be handled and disposed of the same way.

Table 8.7. Generation of High-Specific-Activity Organic Resins--Minimum and Maximum Alternatives

Source of Treated Liquid Waste	Minimum Generation		Maximum Generation	
	Volume (ft^3)	Curies[a]	Volume (ft^3)	Curies[a]
1. AFHB accident water[b]	1,380	54,500	1,380	54,500
2. RCS water	540	19,900	540	19,900
3. RCS flush and drain water	540	19,900	2,690	99,500
Total	2,460	94,000[c]	4,610	174,000[c]

[a]Detailed information on EPICOR II is proprietary. Curies were estimated from actual performance with AFHB liquids extrapolated to other sources.

[b]46 high-specific-activity prefilter liners in storage as shown in footnote c on Table 8.2.

[c]Rounded to nearest thousand.

Because of the relatively high radiation levels, 500 R/hr to 1500 R/hr, there is a potential for breakdown of the organic resin materials. Similar organic resins have been shown to exhibit instability with absorbed doses of 10^8 rads. Some of the EPICOR II prefilter liners have been in storage for 18 months and could remain in storage until waste handling options have been selected and the appropriate facilities become available. These storage times could lead to accumulated doses greater than 10^8 rads. Under these conditions, the organic resins could begin to deteriorate. Gases could be generated and corrosive liquids could be formed in the liners. The licensee currently stores the EPICOR II prefilter liners in the interim waste storage facility modules (see Appendix Q), which are constructed of reinforced concrete and include provisions for sampling

the module sump. In the event the liners develop leaks, the storage module would prevent the migration of the small quantities of liquids which could be present in the dewatered liners. The licensee is planning to perform tests on liquids pumped from a representative prefilter liner to assess the corrosion potential of the liners. The DOE will also perform an investigation of the resins and the corrosion effects of an actual liner to determine if resin degradation is occurring at the rates projected from small-scale laboratory testing. In the event these investigations or actual experience confirm the concerns regarding liner integrity, the licensee will be required to take appropriate corrective actions such as use of overpacks. The staff requested the licensee to develop contingency plans for possible corrective actions. The licensee has informed the NRC staff that future operations of EPICOR II will not use organic resins in the first-stage prefilter liners.[6,7] Only inorganic ion-exchange media will be used in these liners in the future.

Until these high-specific-activity wastes can be shipped offsite, they will be placed in the concrete interim waste storage facility. Because of their characteristics, these materials cannot be disposed of using routine methods at low-level disposal facilities. The most practicable alternative is to transfer these wastes to an existing federal government facility for future processing and eventual disposal. Other less desirable alternatives include processing onsite and/or packaging in a high-integrity container for special disposal at commercial LLW burial sites. The offsite alternative treatment techniques at federal government facilities, which will have to be demonstrated for specific applicability to the condition of the TMI resins, include:

- Storage as a sludge in tanks at government facilities pending eventual immobilization when facilities are available.
- Incineration of the combustible material to reduce volume and mass, followed by immobilization of the residue. A number of incineration facilities being developed for use with transuranic wastes might be capable of modification for incineration of high-specific-activity wastes. These include:
  - Controlled air incinerator developmental units
  - Cyclone incinerator
  - Rotary-kiln production unit
  - Slagging pyrolysis incinerator.
- Acid digestion of combustible waste. This is a potential alternative for treating the organic resins that would result in substantial volume reductions (>60).
- Elution of selected radionuclides (Cs-137, Sr-90) from the resins onto more stable inorganic ion-exchange media (zeolites). The loaded inorganic ion-exchange media could be stored for eventual immobilization. It should be noted that although significant development work has been done at federal government facilities on elution of organic resins, the processes have been developed for highly alkaline systems and would require adaptations for use on TMI resins. This factor and the question of cost effectiveness for such a process requires careful evaluation of the feasibility of this approach.

For each of the alternatives described above, the DOE facilities and staff have experience with similar operations. The various federal government facilities at which these treatment alternatives could be performed and the status of their availability are described in Section 9.1.3.3. Because of the existence of these DOE facilities with experienced staffs, it is the NRC staff's view that these are logical places (as opposed to the TMI site) for further processing and interim storage of the higher specific activity materials, pending ultimate disposal.

Low-Activity Organic Resins

The organic resins that have been or would be generated from second- and third-stage ion-exchange vessels have activity levels and contents that appear to be similar to organic resins from other nuclear facilities and could likely be handled and disposed of as low-level radioactive wastes provided the isotopic concentrations are similar to other resins. Some of these materials, generated by the EPICOR II system, are already in storage at TMI. Additional quantities could be generated through the continued use of EPICOR II for AFHB water and use of the alternative treatment systems, such as the SDS and modified EPICOR II treatment of primary system water. As

low-level radioactive wastes, these resins could be immobilized prior to offsite transport for disposal. Immobilization techniques are discussed in Appendix H, and both cement and vinyl ester styrene binders were considered for immobilization of organic resins.

The quantities and characteristics of organic resins that could be generated depend on the treatment method(s) selected and the binder material used for immobilization. To bound these quantities, best- and worst-case conditions were considered.

- Maximum Package Production. Systems using evaporation followed by resin polishing are assumed to be used for treatment of reactor building sump liquids, and reactor coolant system decontamination liquids. The modified EPICOR II system is assumed to be used for treatment of primary system water and reactor coolant system flush and drain liquids. All resins would be immobilized.

- Minimum Waste Production. Systems using zeolites followed by organic resins polishing are assumed to be used for treatment of reactor building sump water, primary system water, and flush and drain liquid. All resins would be immobilized.

The quantities and characteristics of the resins already generated by EPICOR II and those that could be generated under the assumed bounding conditions are summarized in Table 8.8.

Table 8.8. Generation of Low-Activity Organic Resins--
Minimum and Maximum Alternatives

Source of Treated Liquid Waste	Minimum Generation		Maximum Generation	
	Volume (ft^3)	Ci	Volume (ft^3)	Ci
1. AFHB accident water[a]	1200	260	1200	260
2. RB sump water				
SDS/Modified SDS	200	75	-	-
SDS/EPICOR II	-	-	390	80
3. RCS accident water				
SDS/Modified SDS	200	30	-	-
Modified EPICOR II[b]	-	-	540	100
4. RCS flush and drain[c]				
SDS/Modified SDS	200	60		
Modified EPICOR II	-	-	1970	500
5. Water Based RCS Decontamination	140	1	140	5
Total	1940		4240	

[a]EPICOR II system resins in storage.

[b]Waste volumes based on staff estimate.

[c]Best case removes 20,000 Ci; worst case removes 100,000 Ci.

8.1.2.4 Filters

Most of the treatment alternatives being considered involve use of filters to remove particulates upstream of the main processing system. These filters can be either the cartridge or the precoat type. The volume of waste represented by these filters would be small in comparison with other waste volumes generated by a treatment process. However, the radioactivity level of these filters would be relatively high in comparison with the radioactivity levels of second- and third-stage organic resins, and they may also contain fuel particles and uranium and plutonium.

To date, most of the filter configurations considered have been of the cartridge type, and the staff has based its characterization of these process solid wastes on use of this type of filter. Three different size cartridges have been considered. Cartridge assembiles (6 inches diameter by 26 inches high) with an exterior volume of about 0.5 ft^3 were considered for removal of fuel and core debris from the reactor cooling system water during defueling. When systems include zeolites, a 10-ft^3 filter is used upstream of the zeolite vessel. A limited number of relatively large (96 inches diameter by 36 inches high) core filters could also be generated. The quantity of cartridge-type filters generated depends on the particulate content of the influent and the radioactivity level to which the filters can be loaded. The estimated quantities of filter cartridge assemblies that could be generated are presented in Table 8.9.

The smaller, 0.5-ft^3 filters used for liquid waste pretreatment that contain no fuel particles can be disposed of as low-level radioactive waste in the dewatered condition. They can be readily packaged in 55-gallon drums, each containing seven assemblies. When these same or similar-sized cartridges contain core debris or fuel particles, such cartridges may have to be handled as either transuranic or high-specific-activity waste. In either case, they will have to be stored onsite until suitable facilities are provided by the federal government (Department of Energy). They will have to be packaged in a manner that satisfies U.S. Department of Energy requirements.

The zeolite system prefilters are self contained in that the cartridge assembly also can serve as the disposable container. Repackaging of these filters is not necessary.

In the event the larger core filters are generated, special containers will have to be fabricated to accommodate their unusual configurations.

Table 8.9. Generation of Spent Filter Cartridges

Source of Treated Liquid Waste	Filter Cartridge Type and Loading					
	SDS Type[a]		Inline Type[b]		Core Type[c]	
	Quantity	Curie/Unit	Quantity	Curie/Unit	Quantity	Curie/Unit
1. Reactor building sump water	11	<100	-	-	-	-
2. RCS flush and drain water						
RCS[e]	20	<100	20-200[d]	40	2	<5
Reactor coolant drain tank			6[d]	20	-	-
3. RCS aqueous decontamination solutions	2	<50	-	-	-	-

[a]10-ft^3 stainless steel assemblies used with SDS-type systems.

[b]~ 0.5-ft^3 cartridges used for pre- and postremoval of particulates and ion-exchange media fines.

[c]8-ft diameter × 3-ft high cartridge used to remove particulates during defueling.

[d]Could contain fuel debris and would be handled with spent fuel. Range of units depends on extent of particulates mobilized during defueling.

[e]Range of spent filters depends on activity in system (i.e., 20,000 Ci, or 100,000 Ci).

8.1.2.5 Evaporator Bottoms

Evaporator bottoms could be generated through the use of an evaporator to reduce the volume of AFHB/reactor building and RCS decontamination solutions.

Current practice is to immobilize AFHB and reactor building chemical decontamination solutions with vinyl ester styrene, and the staff is not aware of any reason to discontinue this practice (see Sec. 8.2). In contrast, the relatively large volume of chemical decontamination solutions that could be generated from decontamination of the primary system (500,000 gallons) and the time frame in which these liquids could be generated (after defueling) make the use of evaporation an attractive alternative.

The characteristics of packaged waste that could be generated if 500,000 gallons of chemical decontamination solutions are processed through an evaporator are presented in Table 8.10. The minimum and maximums shown are based on varying the volume reduction factor from 10 to 30. As shown in the table, it has been assumed that these wastes will be immobilized either with vinyl ester styrene or cement and packaged in 55-gallon drums. The use of 50-ft^3 liners would reduce the number of packages shown by a factor of 6.4 and increase the curie content and radiation levels shown by a factor of 6.8. If the CAN DECON technique is used for reactor coolant system decontamination (treated by ion-exchange techniques) or if these chemical decontamination solutions are not treated by evaporation (see Sec. 8.2), the minimum packaged waste volume would be zero.

Table 8.10. Characteristics of Packaged Evaporator Bottoms--Minimum and Maximum Generation

	Minimum Generation	Maximum Generation
Original volume (gallons)	500,000	500,000
Volume after evaporation (gallons)[a]	16,700	50,000
Package type	55-gallon drum	55-gallon drum
Volume increase factor	1.65	1.65
Number of packages[b]	560	1,670
Average Ci per package[c]	36	12
Average surface radiation level[d]	30 R/hr	10 R/hr

[a]Volume reduction factor of 10 for maximum and 30 for minimum.

[b]Based on 30 gallons per package for vinyl ester styrene or cement.

[c]Based on 20,000 Ci inventory. Could be as low as 2000 Ci which would reduce activity levels and curie content to 10 percent of values shown.

[d]Based on 0.8 R/hr per curie.

8.1.2.6 Bituminized Materials

Bituminized materials could be generated through the use of a bitumen/resin system to reduce the volume and immobilize AFHB/reactor building and RCS decontamination solutions. This one-step volume reduction and immobilization system directly immobilizes the waste in bitumen. The other waste forms generated include organic resins. For the reasons stated above in Section 8.1.2.5, it is doubtful that this system would be used for the AFHB/reactor building solutions; however, it could be used for RCS decontamination solutions.

The characteristics of the waste that could be generated if 500,000 gallons of chemical decontamination liquids are processed through a bitumen/resin system are presented in Table 8.11. The system considered is designed to package bituminized waste in drums, and there is no reason to change this package. Organic resins could be packaged and immobilized with cement or vinyl ester styrene within the demineralizer vessel. Alternatively, the organic resins could be removed from the demineralizer vessel, combined with liquid waste, and processed through the extruder/evaporator for packaging in drums.

Table 8.11. Characteristics of Waste Generated from Bitumen/Resin System

Characteristics	Bituminized Liquids	Organic Resins
Original volume (gallons)	500,000	NA
Volume after evaporation in extruder (gallons)[a]	25,000	NA
Package type	55-gallon drum	50-ft^3 liner
Volume increase factor[b]	2	1.5
Number of packages	1,000	2
Average Ci/package[c]	20	5
Average surface radiation level (R/hr)[d]	16	4

[a]Volume reduction factor of 20.

[b]Based on 50/50 weight percent bitumen/waste salts.

[c]Based on 20,000 Ci inventory. Could be as low as 2,000 Ci, which would reduce activity levels and curie content to 10 percent of values shown.

[d]Based on 0.8 R/hr per curie.

8.1.3 Details of Methods and Facilities

Process solid wastes arise from treatment of liquid wastes, and, as discussed in Section 7.1, alternative techniques for treatment of liquid wastes from all sources except the AFHB are still being considered by the licensee. The characteristics of the packaged process solid waste that bound the alternatives still being considered are described in this section. Packaging and handling of the EPICOR II organic resins which have already been generated also are discussed.

8.1.3.1 Accident Sludge

The bounding conditions for generation of packaged accident sludge were presented in Table 8.5 based on the conditions described in Section 8.1.2.1. The relatively high specific activity of AFHB sludge precludes the use of large, 50-ft^3 containers. No decision has been made on whether these materials would be packaged in a dewatered or an immobilized state. However, so as not to foreclose future treatment of these materials because of their high specific activity, packaging in the dewatered state is preferred. Thus, based on the information currently available on these materials, it is assumed for this analysis that AFHB sludge will be packaged in 55-gallon drums in the dewatered condition.

The radioactivity level of reactor building sludge permits the use of either drums or large containers and disposition as low-level radioactive waste. Under current requirements for

disposal of low-level radioactive waste, sludges can be packaged in the dewatered condition. If these sludges are contaminated with transuranic elements with concentrations in excess of 10 nCi/g, they would have to be handled as transuranic (TRU) wastes. Under these conditions, retention of the sludges in the dewatered form would provide maximum flexibility for subsequent transfer of TRU sludges to a federal government facility. Based on these considerations, reactor building accident sludge also would be packaged in the dewatered condition either in drums or large containers.

The characteristics of the packaged accident sludge that could be generated are shown in Table 8.12. If the AFHB or reactor building sludge is disposed of as low-level radioactive waste and shallow land burial site criteria are changed to require immobilization of these sludges, the number of packages shown in Table 8.12 would be increased by 65 percent and the curie content and radiation levels shown would be reduced by 40 percent.

Table 8.12. Characteristics of Packaged Accident Sludge

		Package Characteristics			
	Total Volume (ft^3)[a]	Type	Number[b]	Curie Content	Surface Radiation Level (R/hr)[c]
AFHB sludge	150	55-gallon drum	22	435	350
Reactor building sludge					
Minimum	100	50-ft^3 liner	2	72	60
Maximum	100	55-gallon drum	15	11	9

[a]Sludge volume; packaged volume greater than this due to freeboard in container. Drums contain 7 ft^3; 50 ft^3 liners contain 45 ft^3.

[b]Rounded to nearest package.

[c]Based on 0.8 R/hr per Ci in package.

8.1.3.2 Zeolites

The bounding conditions for generation of 10-ft^3 stainless-steel liners containing 8 ft^3 of zeolite in the dewatered condition were given in Table 8.6. As discussed in Section 8.1.2.2, these zeolites will be handled as high-specific-activity waste and with the intention of being transferred to federal government facilities for disposition. The alternatives related to treatment at such facilities discussed in Section 8.1.2.2 are still being considered. Therefore, to ensure compatibility with potential federal government facility requirements, zeolites will be stored at the TMI site in the dewatered condition until arrangements are made with the U.S. DOE for their disposition or until a suitable disposal facility becomes available.

In cases when zeolites are selected as the ion-exchange media for treatment of liquid wastes, the liner configurations actually used and the radioactivity loadings of the liners will conform to requirements acceptable to DOE. However, to characterize the zeolites that could be generated, the staff made certain assumptions relative to the liner configuration, zeolite loading per liner, radioactivity loading per liner, and the sources of liquid waste that could readily be treated using zeolites or zeolite/titanate ion-exchange media. These assumptions include:

- Zeolites would be placed in stainless-steel liners of the SDS configuration. Each 10-ft^3 capacity liner would contain 8 ft^3 of zeolite or zeolite/titanate mixture.

- For treatment of reactor building sump liquids, liners would be loaded with a minimum of 10,000 Ci and a maximum of 120,000 Ci.

- For treatment of other sources of liquid wastes, the zeolite system would not be located in the spent fuel pool. Under these conditions, zeolite liners would be loaded to 10,000 Ci to minimize shielding requirements around the relocated equipment.

- Reactor building sump liquids would be treated with a modified SDS system, and RCS liquids and RCS flush and drain liquids would be treated with a modified EPICOR II system.

The characteristics of the zeolite liners that could be generated under these conditions are shown in Table 8.13. As shown, the minimum number of liners that could be generated is 11, while the maximum is 78. The major factor contributing to this range is the loading of the zeolites used for reactor building sump water treatment.

For treatment of reactor building sump water and possibly RCS water, the SDS-type system will be operated submerged in spent fuel pool B. The steps involved in handling expended zeolite and cation vessels are as follows:

- Vessel couplings are disconnected with a monorail lifting device and a special tool. Each vessel requires three disconnects.

- After a vessel is disconnected from the system, it is transferred in the pool using the fuel handling crane. A special yoke and shaft are attached to the crane to perform these transfer operations.

- Transfer operations involve movement of a partially expended zeolite vessel to another position in the system or transfer of a completely expended zeolite or cation vessel to the underwater storage rack. Expended filters are also transferred in this manner.

- For shipment or transfer from the pool, a shield cask is loaded into the spent fuel pool. The cask is loaded and closed except for the vent and drain plug. As the cask is raised above the pool, it is gravity drained.

- The cask is then transferred to a transport vehicle or used to move the expended liner or filter cartridge assembly to another storage location in the plant.

The mixed bed demineralizer is operated within a shield above the surface of the spent fuel pool. When the contents of this vessel are expended, the liner and cask are replaced.

Prior to defueling, the system will have to be removed from the spent fuel pool and relocated within the plant. The handling procedures that would be used in another location have not been defined, but would be similar to those used for handling EPICOR II system liners.

8.1.3.3 Organic Resins

The organic resins that have been generated through use of EPICOR II and those that could be generated from treatment of other liquids with ion-exchange techniques can be divided into three categories that reflect the radioactive characteristics of the wastes: (1) high specific activity waste materials, (2) TRU waste materials, and (3) low-activity waste materials. Each category of organic resin wastes is discussed separately below.

<u>High-Specific-Activity Organic Resins</u>

The characteristics of the organic resins generated by EPICOR II which are currently being stored in the dewatered condition were summarized in Table 8.7. The 46 prefilter liners listed in that table, representing about 1400 ft^3 of resin, must be handled as high-specific-activity waste. Detailed information on these liners is presented in Table 8.14. Because of their characteristics, these materials cannot be disposed of using routine methods at low-level waste disposal facilities. Transfer to a federal government facility represents the most practicable alternative for these materials. The federal facility/treatment alternatives being considered were described in Section 8.1.2.3.

Table 8.13. Characteristics of Packaged Zeolite Liners

Source of Treated Liquid Waste	Packaged Liner Characteristics[a]					
	Minimum Generation			Maximum Generation		
	Number of Liners	Maximum Curies per Liner	Maximum Surface Radiation Level per Liner (R/hr)	Number of Liners	Maximum Curies per Liner	Maximum Surface Radiation Level per Liner (R/hr)
1. Reactor building sump water	6	120,000	100,000	54	10,000	8,000
2. RCS water[b]	2	10,000	8,000	6	5,000	4,000
3. RCS flush and drain water[c]	2	10,000	8,000	12	10,000	8,000
4. Aqueous RCS[d] decontamination solutions	1	2,000	1,600	6	5,000	4,000
Total	11			78		

[a]Each liner contains 8 ft^3 of zeolite.

[b]Minimum based on modified EPICOR II system; maximum based on modified SDS.

[c]Minimum based on modified EPICOR II removal of 20,000 Ci; maximum based on modified SDS removal of 100,000 Ci.

[d]Minimum based on SDS/Modified SDS removal of 2,000 Ci; maximum based on SDS/Modified SDS removal of 20,000 Ci.

Table 8.14. Characteristics of Packaged High-Specific-Activity EPICOR II Resins

Group	Number of Prefilter Liners[a]	Total Curies in Group	Maximum Curies Content per Liner[b]	Maximum Surface Radiation Level of Liners[c] (R/hr)
A	30	38,000	1,300	1,040
B	7	8,000	1,200	960
C	7	6,500	1,000	800
D	2	1,000	650	520
Total	46	53,500		

[a]Liners are 4-ft-diameter × 4-ft-high carbon steel with volume of 50 ft^3 and contain 30 ft^3 of dewatered resin.

[b]Maximum gross curies for any liner in group.

[c]Based on 0.8 R/hr per Ci for the maximum curie content liner in group. Hot spots in liners could lead to local radiation readings 1.5 times the maximum shown.

Low-Activity Organic Resins

The organic resins generated by EPICOR II treatment of AFHB liquids were characterized in Table 8.2. The organic resins in the three low-activity-level prefilter liners and the second- and third-stage liners may be immobilized. The characteristics of these resins after immobilization with cement are presented in Table 8.15.

Techniques for immobilization of these EPICOR II organic resins with cement are currently being evaluated by the licensee. To minimize package handling and the number of cask shipments required, these evaluations are based on the use of large containers that correspond to the sizes of the demineralizer vessels in the EPICOR II system. Two approaches are being considered:

- Out-of-Container Mixing--This approach involves transfer of the resins from the demineralizer vessels, mixing the resins and immobilization agent with an in-line mixer, and repackaging the mixture in the demineralizer vessel, which would serve as the disposable container. This approach requires the design and installation of a remotely operated solidification facility and removal of the resins from the demineralizer vessels.

- In-Situ Mixing--This approach involves adding the immobilization agent to the demineralizer vessel and mixing the resins and immobilization agent within the vessel. This does not require an extensive, remotely operated solidification facility, but does require the redesign of the demineralizer vessels to accommodate a mixing device. Thus, resins still would have to be transferred from the vessels currently being used into new containers.

The mixing approach selected by the licensee will be based on operational considerations and will not affect the volume of packaged waste. The immobilization agent used, however, will affect the volumes of packaged waste. The licensee's work to date[7] with cement indicates that a mixture of about 33 weight percent resins, 14 weight percent water, 48 weight percent Portland II cement, and 5 weight percent of a chemical additive could yield an acceptable immobilized product. These mixing ratios represent a volume increase factor of about 1.5. The vessels have sufficient void space to accommodate a volume-increase factor of 1.5, so the planned use of cement will not increase the number of containers required. Thus, Table 8.15 is based on a volume increase factor of 1.5. If vinyl ester styrene is used, the estimated volume increase factor also would be about 1.5.

Table 8.15. Characteristics of Packaged Low-Activity EPICOR II Organic Resins[a]

Group	Number of Liners	Total Curies in Group	Maximum Curie Content per Liner[b]	Maximum Surface Radiation Level of Liners[c] (R/hr)
Prefilter[d]	3	440	155	125
Cation A[d]	4	160	60	50
Cation B	3	55	20	16
Cation C	7	25	6	5
Mixed bed[e]	6	20	6	5
Total	23	700		

[a]Organic resins are immobilized in cement with a volume increase factor of 1.5 and packaged in EPICOR II system vessels.

[b]Maximum gross curies for any liner in group.

[c]Based on 0.8 R/hr per Ci for the maximum curie content liner in group. Hot spots could lead to local radiation readings 1.5 times the maximum shown. Mixed bed vessels are 6-ft-diameter × 6-ft-high liners which contain 115 ft^3 of resin plus cement.

[d]Prefilter and cation vessels are 4-ft-diameter × 4-ft-high liners which contain 30 ft^3 of resins plus cement.

[e]Mixed-bed vessels are 6-ft-diameter × 6-ft-high liners which contain 115 ft^3 of resin plus cement.

Source: "EPICOR II Resin Solidification Procurement Specification," Letter (TLL 545) from G.K. Hovey, Metropolitan Edison Co. - TMI, to TMI Program Office, November 17, 1980.

Low-activity resins also could be generated from treatment of other liquids with ion-exchange techniques. The bounding quantities of these materials that could be generated were given in Table 8.8. The characteristics of these packaged materials under minimum and maximum waste generation conditions are described in Table 8.16. The table is based on the assumption that organic resins will be immobilized and repackaged in the ion-exchange vessel removed from the treatment system. The assumed volume increase factor due to immobilization is 1.5 as discussed above. As shown in Table 8.16, under best-case conditions,[11] liners containing about 650 ft^3 of resins could be generated. Under worst-case conditions, 62 liners containing about 3100 ft^3 of resins could be generated.

TRU Organic Resins

The RCS flush and drain liquids could contain dissolved transuranic elements. Treatment of these liquids with ion exchange technology will lead to disposition of these transuranic elements on the organic resins used in the treatment system. The concentration of these transuranic elements could exceed 10 nCi/g, which would require their packaging and handling as TRU wastes. Under these conditions, the organic resins would have to be retained in the dewatered condition and transferred to a DOE facility for disposition as TRU waste.

The characteristics of the TRU organic resins that could be generated from treatment of RCS flush and drain water are presented in item 3 of Table 8.16. These materials would be transferred to DOE in the dewatered condition. They would be packaged in the ion-exchange vessels removed from the treatment system. Since Table 8.16 is based on an immobilization volume increase factor of 1.5, there is no difference between the estimated number of immobilized and dewatered liners.

Table 8.16. Characteristics of Packaged Low-Activity Organic Resins

Source of Treated Liquid Wastes	Minimum Generation			Maximum Generation		
	Number of Liners	Maximum Curies per Liner	Maximum Surface Radiation Level per Liner (R/hr)	Number of Liners	Maximum Curies per Liner	Maximum Surface Radiation Level per Liner (R/hr)
1. Reactor building sump water[a]						
Cation	2	50	40	4	50	40
Mixed bed	1	5	4	2	5	4
2. RCS water[b]						
Cation	2	50	40	7	50	40
Mixed bed	1	5	4	3	5	4
3. RCS flush and drain water[c]						
Cation	2	50	40	35	50	40
Mixed bed	1	5	4	8	5	4
4. RCS decontamination solution[d]						
Cation	1	5	4	2	5	4
Mixed bed	1	5	4	1	5	4
Total	11			62		

[a]Minimum based on SDS/Modified SDS; maximum on SDS/EPICOR II.

[b]Minimum based on SDS/Modified SDS; maximum on modified EPICOR II.

[c]Minimum based on SDS/Modified SDS removal of 20,000 Ci; maximum on modified EPICOR II removal of 100,000 Ci.

[d]Minimum based on use of evaporator/resin or bitumen/resin system; maximum based on use of system with first-stage zeolite liner.

8.1.3.4 Filters

The estimates for spent filter cartridge generation were presented in Table 8.9. As described in Section 8.1.2.4, cartridges to be handled as low-level waste can be packaged in 55-gallon drums or in special large containers. Cartridges that could require handling as TRU or high-level waste would be packaged in containers compatible with DOE requirements. The latter include the filter cartridges generated during RCS and reactor coolant flushing and draining activities. If these filters contain fuel particulates, the radiation level for a cartridge could be 500 R/hr.

8.1.3.5 Evaporator Bottoms

The bounding conditions for evaporator bottoms generation were presented in Table 8.10. The quantities shown in that table are based on the use of an evaporator/resin system to treat RCS chemical decontamination solutions.

Implementation will require the design and construction of a separate evaporator facility that could take up to three to four years to become operational. Design work on such a facility was initiated by the licensee but has been deferred. A 10-gpm-capacity evaporator compatible with processing of chemical decontamination liquids and other liquids was ordered, but the licensee cancelled the order in 1980, pending selection of a technique for reactor coolant system decontamination. If RCS decontamination solutions are generated, this project would have to be reactivated to process these liquids through an evaporator.

The characteristics of packaged waste that could be generated if 500,000 gallons of chemical decontamination solutions are processed through an evaporator were presented in Table 8.10.

No decision has been made concerning the use of an evaporator. The characteristics of the packaged waste arising from direct immobilization of these chemical decontamination solutions are presented in Section 8.2.4.

8.1.3.6 Bituminized Materials

The characteristics of the packaged waste arising from use of a bitumen/resin system to process 500,000 gallons of RCS decontamination solutions were presented in Table 8.11. This system is described in Appendix H.

Space requirements for a system of the type considered are such that the equipment could not be located within existing facilities. Design and construction of facilities to incorporate this equipment plus procurement and installation of the equipment could take three to four years.

Since no decision has been made on selection of a technique for decontamination of the reactor coolant system, a decision cannot be made on the use of a bitumen/resin system at this time.

8.1.4 Effluents and Releases to the Environment

The nature and impacts of releases to the environment that could occur during normal and abnormal conditions or accidents in the packaging and handling of process solid wastes are discussed in this section.

8.1.4.1 Normal Operations

Under normal conditions of operation, airborne releases could arise during immobilization of accident sludge, organic resins, and evaporator bottoms. These releases would be released to the plant vent system. Estimated releases during immobilization are presented in Table 8.17. These values are based on a fractional release of 10^{-5}, or 0.001 percent, of the radionuclides in the waste that will be immobilized.

8.1.4.2 Package-Handling Accidents

Worst-case accidents during packaged-waste-handling operations would arise when a drum or resin liner is dropped from a crane or monorail during remote handling. The consequences of such accidents would depend on the waste type and its condition, the radioactivity and radionuclide content of the breached container, the fractional release of materials in the container, its interaction with other containers, and the area in the plant where the accident occurred.

Table 8.17. Maximum Estimated Airborne Effluents Released to the Environment during Immobilization of Process Solid Waste

Source of Treated Liquid Waste	Amounts Released to Exhaust System (Ci)						
	Sr-89	Sr-90	Ru-106	Sb-125	Cs-134	Cs-137	Ce-144
1. AFHB water[a]							
Prefilters (440)	NEG	NEG	NEG	NEG	7×10^{-7}	3.7×10^{-4}	NEG
Cation liners (240)	NEG	NEG	NEG	NEG	3.4×10^{-7}	2×10^{-6}	NEG
Mixed bed liners (20)	NEG	NEG	NEG	NEG	3.5×10^{-8}	1.6×10^{-7}	NEG
2. Reactor building sump water[b]							
Cation liners (25)	NEG	6.7×10^{-8}	6.7×10^{-8}	-	1.4×10^{-8}	9.7×10^{-8}	3.7×10^{-9}
Mixed bed liners (55)	NEG	1.4×10^{-8}	2.2×10^{-8}	2.3×10^{-7}	NEG	6.6×10^{-9}	NEG
3. RCS primary water[c]							
Cation liners (95)	1.9×10^{-8}	1.6×10^{-7}	6.1×10^{-7}	-	NEG	NEG	1.6×10^{-7}
Mixed bed liners (5)	5×10^{-10}	5.5×10^{-9}	3.2×10^{-8}	5×10^{-9}	NEG	NEG	7×10^{-9}
4. RCS flush and drain water[c]							
Cation liners (475)	9.5×10^{-8}	8×10^{-7}	3×10^{-6}	-	NEG	NEG	8×10^{-7}
Mixed bed liners (25)	2.8×10^{-9}	2.8×10^{-7}	1.6×10^{-7}	2.5×10^{-8}	NEG	NEG	3.5×10^{-8}
5. RCS chemical decontamination solution							
Evaporator bottoms (20,000)	7.6×10^{-6}	7.8×10^{-5}	NEG	NEG	1.5×10^{-5}	1×10^{-4}	NEG

( ) indicates total Ci in waste to be immobilized, worst case. Assumed fractional release is 0.001 percent.

[a]Resins presently in storage.

[b]Treated with modified SDS system or other system which removes Ru, Sb and Ce.

[c]Treated with modified EPICOR II system.

The fractional release rates used to estimate the amounts of radionuclides released in the form of respirable particulates for the postulated accidents are as follows:

- Dewatered zeolites, and resins, accident sludges, and filter cartridges--10^{-4} of the package radionuclide content
- Resins, accident sludges, and evaporator bottoms immobilized with cement--10^{-5} of the package radionuclide content
- Resins, accident sludges, and evaporator bottoms immobilized with vinyl ester styrene--10^{-6} of the package radionuclide content
- Waste materials immobilized with bitumen--10^{-6} of the package radionuclide content.

The worst-case accidents postulated during handling of packaged waste from each alternative are discussed separately below, and the releases that would result are summarized for all the accidents in separate tables.

Accidents Involving Zeolite-Based System Wastes

Different handling procedures are used for each type of waste container or vessel used in the zeolite-based systems. The zeolite and cation vessels and the spent filter cartridges are placed in top-loaded shipping or transfer casks underwater in the spent fuel pool and then removed from the pool using the overhead crane. Mixed-bed demineralizer vessels are not handled in the spent fuel pool. The worst-case accident that could arise from these operations would occur during the transfer of a first-stage zeolite liner in a shipping cask within the reactor building. While the cask is designed to retain its integrity during a 30-ft drop, at one point it could fall a distance of 60 ft. For this 60-ft drop accident it was assumed that the zeolite liner in the cask, as well as the cask seal, would be breached. The estimated releases for a zeolite liner containing 120,000 curies of dewatered zeolites arising from treatment of reactor building sump water are shown in Table 8.18. This curie loading is the maximum expected.

Table 8.18. Estimated Releases during SDS-Based System Package Handling Accidents

Radionuclide	Estimated Releases (Ci)			
	Zeolite Liner[a,b]	Filter Assembly[b,c]	Cation Liner[b,d]	Mixed Bed Liner[e]
Sr-89	1×10^{-2}	2×10^{-4}	1×10^{-5}	5×10^{-6}
Sr-90	0.12	3×10^{-3}	8.5×10^{-4}	6×10^{-5}
Ru-106	-[f]	-	3.2×10^{-3}	3.2×10^{-4}
Sb-125	-	4×10^{-4}	-	5×10^{-5}
Cs-134	1.6	1×10^{-4}	-	-
Cs-137	10.3	1.2×10^{-3}	-	-
Ce-144	-	-	8.5×10^{-4}	8.5×10^{-5}

[a]One liner containing 120,000 Ci.

[b]Accidents occur in AFHB; therefore, releases to atmosphere will be 1/1000 the values shown.

[c]Filter loading of 50 Ci after cleanup of reactor building sump water.

[d]Cation liner loading of 50 Ci after cleanup of primary system water with modified SDS system; a worst-case isotopic distribution.

[e]Mixed-bed liner loading of 5 Ci after cleanup of primary system water with modified SDS system; a worst-case isotopic distribution. Released directly to the atmosphere.

[f]"-" indicates that radionuclide is less than one percent of release.

The estimated releases arising from the breach of a spent filter cartridge containing 50 Ci and a cation liner containing 50 Ci under similar accident conditions also are shown in Table 8.18.

In all cases, the estimated releases would be vented to the plant exhaust system.

If the zeolite/resin system were installed in another location where there was no potential for a cask drop of more thant 30 ft, there would be no releases because the cask would retain its integrity under those accident conditions.

Use of a zeolite-based system also generates mixed-bed organic resins. The worst-case accident would involve the breach of one of these liners during transfer to a storage location outside of the reactor building. The estimated releases from a 190-ft^3 mixed-bed liner are also shown in Table 8.18. The worst-case conditions shown are based on use of a modified SDS system for treatment of primary system water.

Accidents Involving EPICOR II and Modified EPICOR II System Wastes

The worst-case scenario for EPICOR II wastes involves an accident with a prefilter liner containing dewatered organic resins with a loading of 1300 Ci. This accident could occur during transfer of the liner from within the chemical cleaning building to a storage module in the EPICOR II waste storage facility. The accident with the maximum release consequences would occur when a dropped liner containing dewatered resins breaches the integrity of a second container already in storage and these releases are directly to the atmosphere. The estimated releases for this postulated accident for EPICOR II prefilter liners with a gross activity of 1300 Ci each are given in Table 8.19.

Use of the EPICOR II system also generates second- and third-stage organic resin liners. For the liners already generated, the worst-case accident would be when these liners are being removed from the storage modules for transfer to an immobilization facility. The resins are in a dewatered condition, and the releases would be directly to the atmosphere. For the smaller 50-ft^3 cation liners, a second liner also could be breached when one liner is dropped on another. The estimated releases from these accidents, based on two 50-ft^3 liners containing 60 Ci each and a 170-ft^3 mixed-bed liner containing 5 Ci, are also shown in Table 8.19.

Table 8.19. Estimated Releases during EPICOR II and Modified EPICOR II System Package Handling Accidents

	Estimated Releases (Ci)					
	EPICOR II System[a]			Modified EPICOR II System[b]		
Radionuclide	Prefilter Liner (2,600 Ci)	Cation Liner (120 Ci)	Mixed Bed Liner (5 Ci)	Zeolite Liner (10,000 Ci)	Cation Liner (120 Ci)	Mixed Bed Liner (5 Ci)
Sr-89	-	-	-	4.1×10^{-2}	2×10^{-4}	-
Sr-90	-	-	-	0.39	2×10^{-3}	9×10^{-5}
Ru-106	-	-	-	-	7.5×10^{-3}	5×10^{-4}
Sb-125	-	-	-	-	-	8×10^{-5}
Cs-134	4×10^{-2}	1.7×10^{-3}	1×10^{-4}	7.5×10^{-2}	-	-
Cs-137	0.22	1×10^{-2}	5×10^{-4}	0.50	-	-
Ce-144	-	-	-	-	2.5×10^{-3}	1.1×10^{-4}

"-" show presence of isotope or less than 1 percent or no data for EPICOR II.

[a]Based on isotopic distribution in EPICOR II liners in storage, prefilter PF-10, cation DR-11, and mixed bed DS-2. ("Scoping Studies of the Alternative Options for Defueling, Packaging, Shipping, and Disposing of the TMI-2 Spent Fuel Core," Allied-General Nuclear Services, AGNS-35900-1.5-79, September 1980.)

[b]Based on cleanup of primary system water.

A modified version of the EPICOR II system also is being considered for treatment of primary system water. If this alternative is selected, cation and mixed-bed liners will be generated. The worst-case accident for these liners is identical to that described above for liners already generated. The estimated releases from these accidents, based on two 50-ft^3 cation liners containing 60 Ci each and a 170-ft^3 mixed-bed liner containing 5 Ci, are also shown in Table 8.1-19.

The modified EPICOR II system that could be used to treat primary system water also generates a 10-ft^3 zeolite liner. This liner could contain up to 10,000 Ci and would also be transferred from the chemical cleaning building to another area for storage. During this transfer operation, the liner could be dropped 15 ft. The estimated releases for this accident are shown in Table 8.19.

Accidents Involving Other Wastes

Accident releases could also result from the breach of packages containing accident sludge, immobilized evaporator bottoms, and bituminized chemical decontamination solutions.

The worst-case accident for accident sludges would be the breach of a 55-gallon drum containing dewatered AFHB sludge during remote placement of the drum in a shipping cask. This accident could occur outdoors, and releases would be directly to the atmosphere. The releases for this accident are shown in Table 8.20 for a drum containing 250 Ci, the maximum estimated curie content of a drum of dewatered AFHB sludge. The estimated releases from the breach of a 50-ft^3 liner containing dewatered reactor building sludge also are shown in the table. This liner was assumed to contain 70 Ci, the maximum curie content estimated for a liner of reactor building accident sludge. Cement immobilization would reduce the releases shown by a factor of 165, and vinyl ester styrene immobilization would reduce them by a factor of 1650.

The worst-case accident for immobilized evaporator bottoms or bituminized decontamination solutions would occur if a drum containing these materials was breached during cask loading. The estimated releases for these accidents also are shown in Table 8.20.

Table 8.20. Estimated Releases during Other Waste Type Package Handling Accidents

	Estimated Releases (Ci)[a]			
Radionuclide	AFHB Accident Sludge (250 Ci)	Reactor Building Accident Sludge (70 Ci)	RCS Evaporator Bottoms[b] (36 Ci)	RCS Bituminized Liquids[b] (20 Ci)
Sr-89	5×10^{-4}	-	1.4×10^{-5}	8×10^{-7}
Sr-90	2.5×10^{-3}	6×10^{-3}	1.4×10^{-4}	7.8×10^{-6}
Ru-106	-	1.3×10^{-4}	-	-
Sb-125	-	1.4×10^{-4}	-	-
Cs-134	3.5×10^{-3}	8×10^{-5}	2.9×10^{-5}	1.6×10^{-6}
Cs-137	1.8×10^{-2}	5×10^{-4}	1.8×10^{-4}	9.8×10^{-6}
Ce-144	-	7×10^{-5}	-	-

[a]Releases are directly to atmosphere. If accident occurs in building with HEPA filter, releases will be a factor of 10^3 less than the values shown.

[b]Waste forms arise from different treatment of same liquid waste source. Therefore releases are mutually exclusive.

8.1.5 Environmental Impacts

8.1.5.1 Occupational Doses

The estimated occupational doses to immobilize organic resins and evaporator bottoms and to package and handle these and other process solid wastes are discussed in this section. These estimates cover the operations involved in handling the packages through placement of the waste in onsite storage facilities to await shipment.

Organic Resins

The EPICOR II system exposure experience for system operation and liner handling was used to estimate occupational exposure during immobilization of organic resins and placement of these packaged wastes into storage to await shipment.

The operations and sampling exposure was 0.006 person-mrem per gallon for a throughput of 2 gpm. This corresponds to about 0.7 person-mrem per hour of system operation. For resin immobilization, liners have to be sluiced to a transfer tank at a rate of about 1 gpm, mixed with a binder material and repackaged. Thus, resins can be transferred at an average rate of about 8 ft^3 per hour. The throughput of a typical immobilization system for large liners is about 40 ft^3 per hour. Combining these two factors results in an average cumulative dose rate of about 0.1 person-mrem/ft^3 of resin. For the maximum volume of resins to be immobilized, about 3300 ft^3, the estimated cumulative dose is about 0.3 person-rem.

The EPICOR II liner handling experience of 85 mrem per liner is also relevant since resin liners must be removed from storage, immobilized, and then returned to storage. A maximum of about 100 liners will have to be immobilized, resulting in cumulative occupational dose of about 8.5 person-rem. For EPICOR II, cumulative maintenance dose was about 1.4 person-mrem per hour of system operation. Since the maintenance needs of an immobilization system will exceed those of EPICOR II, the estimated cumulative maintenance dose for the immobilization system was increased to 4 person-mrem per hour. For 100 hours of operation, the cumulative maintenance dose is 0.4 person-rem.

Thus, the total estimated cumulative dose is about 9 person-rem. These operations will be performed by a three-person crew over a period of 18 months. Under these conditions, the estimated dose to each crew member is 3 rem, and the quarterly dose is about 0.5 rem.

Based on a maximum cumulative dose of 9 person-rem and a work force of three persons, the expected number of additional cancer mortalities in the work force would be less than 0.001. The added probability that the average worker will die of cancer is 1 in 3000. The expected number of additional genetic effects in the offspring of the work force exposed to this cumulative dose would be less than 0.002.

Evaporator Bottoms and Bituminized Decontamination Solutions

The generation of evaporator bottoms or bituminized materials results from treatment of RCS chemical decontamination solutions. Thus, they are mutually exclusive alternatives.

The dose estimates to process these wastes through a bitumen/resin system were presented in Section 7.1.4. This cumulative dose was about 12.5 person-rem. The handling of the 1000 drums that could be generated will lead to additional cumulative dose of about 1 person-rem (see Appendix N).

The dose estimates for treating RCS decontamination liquids with an evaporator/resin system also were presented in Section 7.1.4. The range of cumulative dose estimates was 32 to 63 person-rem. Additional occupational dose will arise from immobilization of evaporator bottoms and drum handling.

The typical throughput of an immobilization system packaging waste in drums is about two drums per hour. At 0.7 person-mrem per hour (EPICOR II experience), the operations cumulative dose incident to immobilizing the maximum of 1670 drums is about 0.6 person-rem. Handling will be completely remote and the estimated dose is less than 1 person-mrem per drum (see Appendix N). For 1670 drums, this is about 1.7 person-rem. At a throughput of two drums per hour, the immobilization system will have to be operated for about 850 hours to package 1670 drums. At

4 person-mrem per hour for cumulative maintenance dose, this results in a cumulative dose of about 3.4 person-rem. Since immobilization and handling of evaporator bottoms drums represents a worst-case exposure condition, the total cumulative dose of about 5.7 person-rem is considered here. These operations will be performed by a single three-person crew over a period of 18 months. Under these conditions, the estimated dose to each crew member would be 1.9 rem, and the quarterly dose would be about 0.3 rem.

Based on a maximum cumculative dose of 5.7 person-rem and a work force of three persons, the expected number of additional cancer mortalities in the work force would be less than 0.0007. The added probability that the average worker will die of cancer is 1 in 4000. The expected number of additional genetic effects in the offspring of the work force exposed to this cumulative radiation dose would be less than 0.0015.

Other Process Solids

The estimated doses for packaging and handling other process solids were based on the information in Appendix N. The cumulative occupational dose estimates for packaged accident sludge and spent filters are shown in Table 8.21.

These wastes will be generated and packaged intermittently over about 36 months. For the cumulative occupational dose of 2.3 person-rem, the expected number of additional cancer mortalities in a work force of three people is 0.0003. This means that the added probability that the average individual worker would die of cancer is 1 in 10,000. The expected number of additional genetic effects in the offspring of the work force exposed to this cumulative dose of radiation is about 0.0006.

Table 8.21. Estimated Worker Exposure to Package and Handle Accident Sludge and Filters

Waste Type	Person-mrem per Package	Cumulative Occupational Dose (person-rem)
1. AFHB accident sludge	2	0.01
2. Reactor building accident sludge[a]	2	0.05
3. Core filters[b]	22	0.12
4. In-line filters[c]	11	2.1
Total		2.3

[a]Based on Appendix N estimates for EPICOR II liners.

[b]Based on Appendix N estimates for irradiated hardware.

[c]Based on maximum number of filters--could be a factor of 10 lower.

8.1.5.2 Offsite Doses

The dose estimates presented here for processing of solid wastes are based on the source terms developed in Section 8.1.4. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3. The dose estimates are listed in Table 8.22. The total-body population dose received by the human population residing within 50 miles during these activities is estimated to be about 2×10^{-2} person-rem.

Table 8.22. Dose Estimates for the Maximum Exposed Individual Due to Processing Solid Wastes

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	3.1×10^{-5}	4.5×10^{-4}	2.2×10^{-5}
	Ground Shine	4.2×10^{-5}	4.2×10^{-5}	4.2×10^{-5}
	Vegetable Use	2.6×10^{-3}	1.1×10^{-2}	1.1×10^{-3}
	Total	2.7×10^{-3}	1.1×10^{-2}	1.2×10^{-3}
Nearest milk goat	Inhalation	3.3×10^{-5}	1.7×10^{-4}	1.5×10^{-5}
	Ground Shine	4.0×10^{-5}	4.0×10^{-5}	4.0×10^{-5}
	Goat Milk Use	8.6×10^{-4}	6.8×10^{-3}	6.3×10^{-3}
	Total	9.3×10^{-4}	7.0×10^{-3}	6.4×10^{-3}
Nearest cow and garden	Inhalation	3.4×10^{-5}	5.0×10^{-4}	2.4×10^{-5}
	Ground Shine	6.2×10^{-5}	6.2×10^{-5}	6.2×10^{-5}
	Vegetable Use	3.8×10^{-3}	1.6×10^{-2}	1.6×10^{-3}
	Cow Milk Use	3.6×10^{-4}	2.0×10^{-3}	1.2×10^{-3}
	Total	4.3×10^{-3}	1.9×10^{-2}	2.9×10^{-3}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow and garden locations are for children, and those for the nearest milk goat location are total-body estimates for adults and bone and liver estimates for infants.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

8.1.5.3 Postulated Accident Effects

The type of accident for which dose estimates are presented here is the breach of a package containing waste. The accident scenarios are described in Section 8.1.4.2. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses is discussed in Section 10.4. The estimated doses to the maximum exposed individual and the source for each accident scenario are listed in the following tables:

Scenario	Dose Estimate Table Number	Source Term Table Number
SDS waste (reactor building sump water--zeolite liner)	8.23	8.18
SDS waste (reactor building sump water--filter assembly)	8.24	8.18
SDS waste (primary water--cation liner)	8.25	8.18
SDS waste (primary water--mixed-bed liner)	8.26	8.18
EPICOR II waste (AFHB water--prefilter liner)	8.27	8.19
EPICOR II waste (AFHB water--cation liner)	8.28	8.19
EPICOR II waste (AFHB water--mixed-bed liner)	8.29	8.19
Modified EPICOR II waste (primary water--zeolite liner)	8.30	8.19
Modified EPICOR II waste (primary water--cation liner)	8.31	8.19
Modified EPICOR II waste (primary water--mixed-bed liner)	8.32	8.19

Table 8.23. Dose Estimates for the Maximum Exposed Individual Due to Breaching of a Package Containing SDS Waste-- Reactor Building Sump Water - Zeolite Liner

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	5.7×10^{-6}	2.0×10^{-5}	9.1×10^{-6}
	Ground Shine	6.8×10^{-5}	6.8×10^{-5}	6.8×10^{-5}
	Vegetable Use	4.7×10^{-4}	2.4×10^{-3}	1.5×10^{-3}
	Total	5.4×10^{-4}	2.5×10^{-3}	1.6×10^{-3}
Nearest milk goat	Inhalation	5.7×10^{-6}	1.0×10^{-5}	6.7×10^{-6}
	Ground Shine	6.8×10^{-5}	6.8×10^{-5}	6.8×10^{-5}
	Goat Milk Use	1.1×10^{-3}	7.6×10^{-3}	9.3×10^{-3}
	Total	1.2×10^{-3}	7.7×10^{-3}	9.4×10^{-3}
Nearest cow and garden	Inhalation	3.9×10^{-6}	1.4×10^{-5}	6.3×10^{-6}
	Ground Shine	6.8×10^{-5}	6.8×10^{-5}	6.8×10^{-5}
	Vegetable Use	4.7×10^{-4}	2.4×10^{-3}	1.5×10^{-3}
	Cow Milk Use	2.7×10^{-4}	1.2×10^{-3}	1.2×10^{-3}
	Total	8.1×10^{-4}	3.7×10^{-3}	2.8×10^{-3}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and for the nearest cow and garden locations are for adults for total-body and for children for bone and liver. The dose estimates for the nearest milk goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.24. Dose Estimates for the Maximum Exposed Individual Due to Breaching of a Package Containing SDS Waste--Reactor Building Sump Water - Filter Assembly

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	1.8×10^{-5}	2.8×10^{-4}	1.0×10^{-6}
	Ground Shine	7.0×10^{-6}	7.0×10^{-6}	7.0×10^{-6}
	Vegetable Use	5.4×10^{-3}	2.1×10^{-2}	1.7×10^{-4}
	Total	5.4×10^{-3}	2.1×10^{-2}	1.8×10^{-4}
Nearest milk goat	Inhalation	7.2×10^{-6}	1.1×10^{-4}	7.4×10^{-7}
	Ground Shine	7.0×10^{-6}	7.0×10^{-6}	7.0×10^{-6}
	Goat Milk Use	9.3×10^{-4}	4.3×10^{-3}	1.0×10^{-3}
	Total	9.4×10^{-4}	4.4×10^{-3}	1.0×10^{-3}
Nearest cow and garden	Inhalation	1.2×10^{-5}	1.9×10^{-4}	6.9×10^{-7}
	Ground Shine	7.0×10^{-6}	7.0×10^{-6}	7.0×10^{-6}
	Vegetable Use	5.4×10^{-3}	2.1×10^{-2}	1.7×10^{-4}
	Cow Milk Use	2.8×10^{-4}	1.2×10^{-3}	1.3×10^{-4}
	Total	5.7×10^{-3}	2.2×10^{-2}	3.1×10^{-4}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden locations are for children, and for the nearest milk goat location are for infants.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.25. Dose Estimates for the Maximum Exposed Individual Due to Breaching of a Package Containing SDS Waste--Primary Water - Cation Liner

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	5.3×10^{-6}	8.4×10^{-5}	1.7×10^{-6}
	Ground Shine	2.1×10^{-7}	2.1×10^{-7}	2.1×10^{-7}
	Vegetable Use	1.5×10^{-3}	6.0×10^{-3}	1.9×10^{-7}
	Total	1.5×10^{-3}	6.1×10^{-3}	2.1×10^{-6}
Nearest milk goat	Inhalation	2.2×10^{-6}	3.4×10^{-5}	9.5×10^{-7}
	Ground Shine	2.1×10^{-7}	2.1×10^{-7}	2.1×10^{-7}
	Goat Milk Use	2.4×10^{-4}	9.5×10^{-4}	2.5×10^{-9}
	Total	2.4×10^{-4}	9.8×10^{-4}	1.2×10^{-6}
Nearest cow and garden	Inhalation	3.7×10^{-6}	5.8×10^{-5}	1.1×10^{-6}
	Ground Shine	2.1×10^{-7}	2.1×10^{-7}	2.1×10^{-7}
	Vegetable Use	1.5×10^{-3}	6.0×10^{-3}	1.9×10^{-7}
	Cow Milk Use	7.8×10^{-5}	3.0×10^{-4}	7.9×10^{-9}
	Total	1.6×10^{-3}	6.4×10^{-3}	1.5×10^{-6}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden locations are for children, and for the nearest milk goat location are for infants.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.26. Dose Estimates for the Maximum Exposed Individual due to Breaching of a Package Containing SDS Waste--Primary Water - Mixed-Bed Liner

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	3.8×10^{-4}	6.1×10^{-3}	1.7×10^{-4}
	Ground Shine	2.1×10^{-5}	2.1×10^{-5}	2.1×10^{-5}
	Vegetable Use	1.1×10^{-1}	4.3×10^{-1}	1.9×10^{-5}
	Total	1.1×10^{-1}	4.4×10^{-1}	2.1×10^{-4}
Nearest milk goat	Inhalation	1.6×10^{-4}	2.5×10^{-3}	9.5×10^{-5}
	Ground Shine	2.1×10^{-5}	2.1×10^{-5}	2.1×10^{-5}
	Goat Milk Use	1.7×10^{-2}	6.8×10^{-2}	2.5×10^{-7}
	Total	1.7×10^{-2}	7.1×10^{-2}	1.2×10^{-4}
Nearest cow and garden	Inhalation	2.6×10^{-4}	4.2×10^{-3}	1.1×10^{-4}
	Ground Shine	2.1×10^{-5}	2.1×10^{-5}	2.1×10^{-5}
	Vegetable Use	1.1×10^{-1}	4.3×10^{-1}	1.9×10^{-5}
	Cow Milk Use	5.4×10^{-3}	2.2×10^{-2}	7.9×10^{-7}
	Total	1.2×10^{-1}	4.6×10^{-1}	1.5×10^{-4}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden locations are for children, and for the nearest milk goat location are for infants.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.27. Dose Estimates for the Maximum Exposed Individual due to Breaching of a Package Containing EPICOR II Waste--AFHB Water - Prefilter Liner

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	1.1×10^{-1}	2.1×10^{-1}	2.1×10^{-1}
	Ground Shine	1.5	1.5	1.5
	Vegetable Use	8.7	33	35
	Total	10	35	37
Nearest milk goat	Inhalation	1.1×10^{-1}	1.3×10^{-1}	1.5×10^{-1}
	Ground Shine	1.5	1.5	1.5
	Goat Milk Use	25	170	210
	Total	27	170	210
Nearest cow and garden	Inhalation	7.8×10^{-2}	8.7×10^{-2}	1.0×10^{-1}
	Ground Shine	1.5	1.5	1.5
	Vegetable Use	8.7	33	35
	Cow Milk Use	6.0	41	51
	Total	16	76	88

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden locations are for adults for total-body and for children for bone and liver. The dose estimates for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.28. Dose Estimates for the Maximum Exposed Individual due to Breaching of a Package Containing EPICOR II Waste--AFHB Water - Cation Liner

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	5.1×10^{-3}	9.4×10^{-3}	9.2×10^{-3}
	Ground Shine	6.8×10^{-2}	6.8×10^{-2}	6.8×10^{-2}
	Vegetable Use	4.0×10^{-1}	1.5	1.5
	Total	4.7×10^{-1}	1.6	1.6
Nearest milk goat	Inhalation	5.1×10^{-3}	5.7×10^{-3}	6.7×10^{-3}
	Ground Shine	6.8×10^{-2}	6.8×10^{-2}	6.8×10^{-2}
	Goat Milk Use	1.1	7.6	9.5
	Total	1.2	7.7	9.6
Nearest cow and garden	Inhalation	3.5×10^{-3}	6.5×10^{-3}	6.3×10^{-3}
	Ground Shine	6.8×10^{-2}	6.8×10^{-2}	6.8×10^{-2}
	Vegetable Use	4.0×10^{-1}	1.5	1.5
	Cow Milk Use	1.1	4.7	4.9
	Total	1.6	6.3	6.5

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden locations are for adults for total-body and for children for bone and liver. The dose estimates for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.29. Dose Estimates for the Maximum Exposed Individual due to Breaching of a Package Containing EPICOR II Waste--AFHB Water - Mixed Bed Liner

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	2.6×10^{-4}	2.9×10^{-4}	3.5×10^{-4}
	Ground Shine	3.6×10^{-3}	3.6×10^{-3}	3.6×10^{-3}
	Vegetable Use	2.1×10^{-2}	7.6×10^{-2}	7.9×10^{-2}
	Total	2.5×10^{-2}	8.0×10^{-2}	8.3×10^{-2}
Nearest milk goat	Inhalation	2.6×10^{-4}	2.9×10^{-4}	3.5×10^{-4}
	Ground Shine	3.6×10^{-3}	3.6×10^{-3}	3.6×10^{-3}
	Goat Milk Use	5.9×10^{-2}	3.8×10^{-1}	4.9×10^{-1}
	Total	6.3×10^{-2}	3.8×10^{-1}	4.9×10^{-1}
Nearest cow and garden	Inhalation	1.8×10^{-4}	2.0×10^{-4}	2.4×10^{-4}
	Ground Shine	3.6×10^{-3}	3.6×10^{-3}	3.6×10^{-3}
	Vegetable Use	2.1×10^{-2}	7.6×10^{-2}	7.9×10^{-2}
	Cow Milk Use	1.4×10^{-2}	9.3×10^{-2}	1.2×10^{-1}
	Total	3.9×10^{-2}	1.7×10^{-1}	2.0×10^{-1}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden locations are for adults for total-body and for children for bone and liver. The dose estimates for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.30. Dose Estimates for the Maximum Exposed Individual due to Breaching of a Package Containing Modified EPICOR II Waste--Primary Water - Zeolite Liner

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	2.3	36	4.6×10^{-1}
	Ground Shine	3.3	3.3	3.3
	Vegetable Use	650	2700	730
	Total	650	2800	730
Nearest milk goat	Inhalation	9.6×10^{-1}	15	3.3×10^{-1}
	Ground Shine	3.3	3.3	3.3
	Goat Milk Use	150	810	460
	Total	150	820	460
Nearest cow and garden	Inhalation	1.6	2.5	3.1×10^{-1}
	Ground Shine	3.3	3.3	3.3
	Vegetable Use	650	2700	730
	Cow Milk Use	42	200	58
	Total	700	2900	790

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden locations are for adults for total-body and for children for bone and liver. The dose estimates for the nearest goat location are for adults for total-body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.31. Dose Estimates for the Maximum Exposed Individual due to Breaching of a Package Containing Modified EPICOR II Waste--Primary Water - Cation Liner

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	1.3×10^{-2}	2.0×10^{-1}	4.9×10^{-3}
	Ground Shine	6.2×10^{-4}	6.2×10^{-4}	6.2×10^{-4}
	Vegetable Use	3.5	14	5.5×10^{-4}
	Total	3.5	14	6.1×10^{-3}
Nearest milk goat	Inhalation	5.2×10^{-3}	8.3×10^{-2}	2.8×10^{-3}
	Ground Shine	6.2×10^{-4}	6.2×10^{-4}	6.2×10^{-4}
	Goat Milk Use	5.7×10^{-1}	2.2	7.6×10^{-6}
	Total	5.8×10^{-1}	2.3	3.4×10^{-3}
Nearest cow and garden	Inhalation	8.7×10^{-3}	1.4×10^{-1}	3.4×10^{-3}
	Ground Shine	6.2×10^{-4}	6.2×10^{-4}	6.2×10^{-4}
	Vegetable Use	3.5	14	5.5×10^{-4}
	Cow Milk Use	1.7×10^{-1}	7.1×10^{-1}	2.4×10^{-5}
	Total	3.7	15	4.6×10^{-3}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden locations are for children, the nearest goat locations are for infants.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

Table 8.32. Dose Estimates for the Maximum Exposed Individual due to Breaching of a Package Containing Modified EPICOR II Waste--Primary Water - Mixed Bed Liner

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	5.7×10^{-4}	9.0×10^{-3}	2.1×10^{-4}
	Ground Shine	2.7×10^{-5}	2.7×10^{-5}	2.7×10^{-5}
	Vegetable Use	1.6×10^{-1}	6.3×10^{-1}	2.4×10^{-5}
	Total	1.6×10^{-1}	6.4×10^{-1}	2.6×10^{-4}
Nearest milk goat	Inhalation	2.3×10^{-4}	3.7×10^{-3}	1.2×10^{-4}
	Ground Shine	2.7×10^{-5}	2.7×10^{-5}	2.7×10^{-5}
	Goat Milk Use	2.5×10^{-2}	1.0×10^{-1}	3.3×10^{-7}
	Total	2.5×10^{-2}	1.0×10^{-1}	1.5×10^{-4}
Nearest cow and garden	Inhalation	3.9×10^{-4}	6.2×10^{-3}	1.5×10^{-4}
	Ground Shine	2.7×10^{-5}	2.7×10^{-5}	2.7×10^{-5}
	Vegetable Use	1.6×10^{-1}	6.3×10^{-1}	2.4×10^{-5}
	Cow Milk Use	8.2×10^{-3}	3.2×10^{-2}	1.0×10^{-6}
	Total	1.7×10^{-1}	6.7×10^{-1}	2.0×10^{-4}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and the nearest cow and garden locations are for children, the nearest goat locations are for infants.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

8.1.5.4 Psychological-Socioeconomic Effects

See Section 8.3.5.4.

8.1.6 Economic Costs

Solid waste processing is broken down into two basic activities--(1) conditioning and (2) waste packaging and handling. Conditioning refers to those operations that transform the concentrates produced during processing of untreated materials into forms suitable for transportation and disposal. Packaging refers to placement of the waste material into disposable containers. Package handling refers to those operations that involve movement of containers in preparation for storage and shipment.

Because of the numerous alternatives available for this activity, minimum and maximum costs have been developed based on the waste volumes presented in Tables 8.3 and 8.4. Details supporting these cost estimates are presented in Appendix K. The staff believes that the costs for solid waste processing lie somewhere between $11,600,000 and $16,000,000 regardless of the alternatives selected.

8.2 CHEMICAL DECONTAMINATION SOLUTIONS

Chemical decontamination solutions are liquid wastes with high chemical and detergent content that are generated during "hands-on" decontamination of building and equipment surfaces and, depending on the techniques used, possibly during decontamination of the reactor coolant system. Section 8.1 discussed evaporation and bituminization of these liquids. This section discusses direct immobilization without volume reduction. Characteristics of these wastes; the alternatives considered for their immobilization, packaging, and handling; and the impacts of these operations are discussed.

8.2.1 Status and Specific Considerations

The status of the chemical decontamination solutions generated through September 22, 1980, and the practices used to package, handle, and solidify these wastes are summarized in this subsection; the types and amounts of chemical decontamination solutions that could be generated by continued decontamination activities also are projected.

8.2.1.1 Efforts to Date

The characteristics and current disposition of chemical decontamination solutions generated during decontamination of the AFHB through September 22, 1980, are shown in Table 8.33. These wastes were generated by the use of a commercially available chemical decontamination solution and have been immobilized in 55-gallon drums using the vinyl ester styrene (VES) process (see Appendix H for details).

8.2.1.2 Projected Requirements

Since decontamination of the AFHB is not yet complete, more chemical decontamination solution waste will be generated there, and similar solutions also will be generated during decontamination of reactor building surfaces and equipment (see Appendix G for details). Decontamination of the reactor coolant system also could result in generation of chemical decontamination solutions if a technique other than the CAN DECON process is used. Estimates of the volumes and characteristics of the chemical decontamination solution wastes that could be generated from these sources are presented in Table 8.34. As shown, decontamination of the reactor coolant system represents the largest potential source of these liquid wastes.

8.2.2 Alternative Methods Considered

Immobilization is the only feasible treatment alternative for AFHB and reactor building chemical decontamination solutions. The options within this alternative involve the selection of a binder material and the type of disposable container used for packaging. The binder materials considered include bitumen, cement, and vinyl ester styrene. A discussion of these immobilization techniques is presented in Appendix H. The types of containers considered include 55-gallon drums and larger containers.

Table 8.33. Characteristics and Disposition of Chemical Decontamination Solution Wastes Generated through September 22, 1980

Factor	Value
Waste Volume (ft^3)	290[a]
Waste treatment process	Solidification with vinyl ester styrene[b]
Packaging Characteristics	
Type	Drum
Volume (ft^3)	7.35
Number	118
Maximum curies	0.15
Maximum surface radiation level	180 mR/hr
Disposition of Containers	
Number stored onsite	19
Number shipped offsite	72

[a]Original liquid volume prior to solidification.

[b]Volume increase factor of 2.5 (20 gallons).

Source: Memorandum from T.L. Gilbert to W.K. Lehto, Argonne National Laboratory, Subject: AFHB Sludge Volumes and Activities, December 9, 1980.

Table 8.34. Projected Volumes and Characteristics of Chemical Decontamination Solutions

Source	Estimated Volume (gallons)	Curie Content	Range of Specific Activity (μCi/mL)
AFHB decontamination	7,000	60	1 to 2
Reactor building decontamination	40,000	10	0.1 to 1
Reactor coolant system decontamination[a]	500,000	20,000[b]	1 to 10

[a]If CAN-DECON technique is used, this source of waste will not occur.

[b]Maximum estimate of activity that could be removed. Maximum estimate is a factor of 10 lower.

Because of the relatively large volume of waste that could be generated during reactor coolant system decontamination, the use of evaporation was considered for treatment of these liquids (see Section 7.1). Evaporator bottoms then would be immobilized and packaged as discussed in Section 8.1.2. If these liquids are not processed through an evaporator, their characteristics would be similar to AFHB and reactor building chemical decontamination liquids. Under these conditions, the alternatives considered in this section for the immobilization and packaging of these chemical decontamination solution wastes from reactor coolant system decontamination are the same as those alternatives considerd for AFHB and reactor building chemical decontamination liquids--immobilization with either bitumen, cement, or vinyl ester styrene and packaging in either drums or large containers.

Consideration of the alternative immobilization techniques and disposable containers in conjunction with the projected volumes bounds the quantities of packaged waste that could be generated. The assumed bounding conditions are:

- Maximum Waste Production. AFHB and reactor building waste volumes plus use of a technique other than the CAN DECON process to decontaminate the reactor coolant system. All liquids would be immobilized with vinyl ester styrene or cement using maximum volume increase factors.

- Minimum Waste Production. AFHB and reactor building waste volumes plus use of the CAN DECON technique to decontaminate the reactor coolant system. All liquids would be immobilized with vinyl ester styrene or cement using minimum volume increase factors.

The characteristics of the packaged waste generated under these bounding conditions are summarized in Table 8.35.

Table 8.35. Chemical Decontamination Solution Waste Generation--Minimum and Maximum Cases

	Source			
	AFHB and Reactor Building		Reactor Coolant System	
Factor	55-Gallon Drum[b]	50-ft^3 Liner[c]	55-Gallon Drum	50-ft^3 Liner
Minimum Generation				
Original volume (gallons)	47,000	47,000	zero	zero
Volume increase factor[d]	1.67	1.67		
Number of packages	1600	220		
Average curies/package	.04	0.3		
Maximum Generation				
Original volume (gallons)	47.000	47,000	500,000	500,000
Volume increase factor[d]	2.5	2.5	2.5	1.67
Number of packages	2,400	350	25,000	2,380
Average curies/package	.03	0.2	0.8[e]	8.4[e]

[a]Where CAN DECON technique is used, this source of waste will not occur.

[b]Useable drum volume is 50 gallons.

[c]Useable liner volume is 350 gallons.

[d]Volume increase factors shown are bounding conditions for immobilization of these liquids with either cement or vinyl ester styrene.

[e]Averages shown are based on estimated maximum inventory of 20,000 Ci in reactor coolant system--could be a factor of ten lower.

The characteristics of the projected AFHB and reactor building chemical decontamination solutions are similar to those generated to date and described in Table 8.32. Immobilization with vinyl ester styrene has been successful and the staff knows of no reason to depart from this technique. The projected volume for these wastes is about ten times greater than the AFHB solutions packaged thus far, and the container size warrants consideration. Use of drums will result in relatively low-radiation-level packages that can be readily handled and disposed of with contaminated trash. The use of the larger liners will substantially reduce the number of packages to be handled, but the radiation levels will be higher and will necessitate the use of remote handling techniques and shielded transport.

If chemical decontamination solutions are generated from reactor coolant system decontamination and are not processed through an evaporator, as discussed in Section 8.1, these liquids will be the largest source of liquid waste to be packaged. As shown in Table 8.35, the use of drums and large liners will result in radiation levels that could require shielded shipment. Under these conditions, the use of larger liners is preferred since the number of packages would be minimized.

8.2.3 Details of Methods and Facilities

The facilities currently being used to immobilize AFHB decontamination solutions with vinyl ester styrene and package these wastes in drums will continue to be used. Minor modifications would have to be made to handle large containers, but this can be readily accomplished.

These facilities are not adequate to handle the relatively large volume of primary system decontamination solutions that could be generated. These wastes could, however, be immobilized and packaged in the same facility that is being considered for immobilization of process solid wastes.

The characteristics of the packaged waste that would be generated are summarized in Table 8.36. As shown, AFHB and reactor building decontamination solutions would be immobilized with vinyl ester styrene and packaged in drums or large containers. If chemical decontamination solutions are generated from reactor coolant system decontamination, large containers are the preferred package. These wastes could be immobilized with either vinyl ester styrene, cement, or bitumen. The selected binder material will depend on the immobilization technique chosen.

8.2.4 Effluents and Releases to the Environment

The nature and impacts of releases to the environment that could occur during packaging and handling of chemical decontamination solution wastes under normal conditions and under abnormal or accident conditions are discussed below.

8.2.4.1 Normal Conditions

Generally, airborne effluents arising from waste-packaging operations are vented to the plant exhaust system. The radionuclides released during immobilization of chemical decontamination solutions were estimated using an assumed fractional release rate of 10^{-5}, or 0.001 percent of the package radionuclide content. These estimates are presented in Table 8.37.

8.2.4.2 Accident Conditions

The accidents postulated during handling and storage consist of a breach of container integrity as a result of the container's being punctured or dropped during waste-handling operations. Worst-case conditions arise when waste is packaged in a large liner with maximum curie content and the breach occurs on the loading dock or some other outdoor location. The fractional release rates used to estimate the amounts of radionuclides released in the form of respirable particulates for postulated accidents are as follows:

- Liquids immobilized with cement--10^{-5} of the packaged waste volume
- Liquids immobilized with vinyl ester styrene--10^{-6} of the packaged waste volume.

The estimated releases for an accident in which the liquids were immobilized with vinyl ester styrene are shown in Table 8.38.

Table 8.36. Packaged Decontamination Solution Waste--Minimum And Maximum Cases

Factor	AFHB and Reactor Building	Reactor Coolant System[a]
Minimum Generation		
Process used	Immobilize with vinyl ester styrene	NA
Volume increase factor	1.67	
Package type	50-ft^3 liner	
Number of package	220	
Average curie/package	0.3	
Average package radiation level	250 mR/hr[d]	
Maximum Generation		
Process used	Immobilize with vinyl ester styrene	Immobilize[b]
Volume increase factor	2.5	2.0
Package type	55-gallon drum	50-ft^3 liner
Number of packages	2400	2860
Average curies/package	0.03	7[c]
Average package radiation level	10 mR/hr	6 R/hr[d]

[a]Where CAN DECON technique used, this source of waste will not occur.

[b]Either cement or vinyl ester styrene.

[c]Based on maximum inventory of 20,000 Ci in reactor coolant system--could be factor of ten lower.

[d]At this radiation level shielded shipment is required.

Table 8.37. Estimated Releases to Environment under Normal Conditions during Immobilization of AFHB and Reactor Building Decontamination Solutions

Radionuclide	Releases (curies)[b]
Cs-137	5.5×10^{-7}
Cs-134	8.4×10^{-8}
Sr-90	6.5×10^{-8}
Sr-89	-

[a]Immobilization of the curies in reactor coolant system already covered in Table 8.17.

[b]Based on immobilization of liquids with total radionuclide inventory of 70 Ci.

Table 8.38. Estimated Accident Releases to Environment from Breach of Packaged Immobilized AFHB and Reactor Building Decontamination Solutions[a]

Radionuclide	Releases[b] (Ci/package)
Cs-137	8.5×10^{-7}
Cs-134	1.3×10^{-7}
Sr-90	1×10^{-8}
Sr-89	-

[a]Accidents assumed to occur outside buildings; releases are directly to atmosphere.

[b]50-ft^3 liner containing 1 Ci immobilized with vinyl ester styrene.

8.2.5 Environmental Impacts

8.2.5.1 Occupational Doses

Occupational radiation doses incurred by workers involved in packaging and handling of chemical decontamination solution wastes (through the stage of placing the packaged wastes in onsite storage facilities to await shipment) are summarized in this section. The basis for the estimates are given in Appendix N.

Estimated occupational doses are presented in Table 8.39. As shown, the cumulative dose from AFHB and reactor building solutions varies from 2.9 to 4.2 person-rem, depending on package type. These wastes would be packaged and handled by two-person crews, and three crews would be used over the 18-month period required to perform this activity. The average dose to each crew member would range from about 0.5 rem to 0.7 rem. The average quarterly dose to each crew member would be about 0.1 rem. The expected number of additional cancer mortalities in the work force of six persons exposed to the cumulative dose of radiation would be between 0.0004 and 0.0006. This means that the added probability that the average individual worker would die of cancer would be between 1 in 20,000 and 1 in 10,000. The expected number of additional genetic effects in the offspring of the work force exposed to the cumulative dose of radiation would be between 0.0008 and 0.001.

If reactor coolant system chemical decontamination solutions are generated, these wastes would be packaged and handled over a 12-month period. Two crews of two persons each would be used, leading to an average dose per crew member of 1.5 and a cumulative dose to the group of 6 person-rem. The average quarterly dose to each crew member would be about 0.4 rem. The expected number of additional cancer mortalities in the work force of four persons exposed to the cumulative dose of radiation would be 0.0008. This means that the added probability that the average individual worker would die of cancer would be 1 in 1200. The expected number of additional genetic effects in the offspring of the work force exposed to the cumulative dose of radiation would be 0.002.

8.2.5.2 Offsite Doses

The dose estimates presented here for processing of chemical decontamination solutions are based on the source terms developed in Section 8.2.4. The calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses and their human health and environmental consequences are discussed in Section 10.3. The estimates are listed in Table 8.40. The 50-mile total body population dose received by the human population during these activities is estimated to be about 2×10^{-5} person-rem.

Table 8.39. Estimated Occupational Doses from Handling and Packaging Chemical Decontamination Solution Wastes[a]

Source and Package Type	Unit Dose (person-mrem/package)	Cumulative Occupational Dose (person-rem)	
		Best Case	Worst Case
AFHB and reactor building			
55-gallon drum	3.5	NA	4.2
Large liner	21	2.9	NA
Reactor coolant system			
Large liner	21	NA	6.0
Totals		2.9	10[b]

[a]Quantities from Table 8.35.

[b]Totals rounded to two significant figures.

Table 8.40. Dose Estimates for the Maximum Exposed Individual due to Immobilizing Chemical Decontamination Solutions

Location	Pathway	Dose (mrem)[a]		
		Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	2.7×10^{-8}	3.8×10^{-7}	2.9×10^{-8}
	Ground Shine	7.0×10^{-8}	7.0×10^{-8}	7.0×10^{-8}
	Vegetable Use	2.2×10^{-6}	9.0×10^{-6}	1.4×10^{-6}
	Total	2.3×10^{-6}	1.0×10^{-5}	1.5×10^{-6}
Nearest milk goat	Inhalation	3.4×10^{-8}	1.5×10^{-7}	1.9×10^{-8}
	Ground Shine	6.0×10^{-8}	6.0×10^{-8}	6.0×10^{-8}
	Goat Milk Use	1.1×10^{-6}	8.0×10^{-6}	9.0×10^{-6}
	Total	1.2×10^{-6}	8.0×10^{-6}	9.0×10^{-6}
Nearest cow and garden	Inhalation	3.0×10^{-8}	4.2×10^{-7}	3.2×10^{-8}
	Ground shine	9.0×10^{-8}	9.0×10^{-8}	9.0×10^{-8}
	Vegetable Use	3.2×10^{-6}	1.4×10^{-5}	2.1×10^{-6}
	Cow Milk Use	4.0×10^{-7}	2.2×10^{-6}	1.6×10^{-6}
	Total	3.7×10^{-6}	1.6×10^{-5}	3.8×10^{-6}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group are listed. The dose estimates for the nearest garden and nearest cow/garden locations are for children, and for the nearest goat location are for adults for total body and for infants for bone and liver.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

8.2.5.3 Postulated Accident Effects

The type of accident for which dose estimates are made here is the breach of a package containing waste from AFHB and reactor building decontamination. The accident is described in Section 8.2.4.2 and the source term is listed in Table 8.38. Calculational models used to make these estimates and the interpretation of their results are described in Appendix W. The significance of these doses is discussed in Section 10.4. The dose estimates for the maximum exposed individual due to breaching a package containing AFHB or reactor building wastes are listed in Table 8.41.

Table 8.41. Estimates of Offsite Doses to the Maximum Exposed Individual Caused by Breach of Package Containing AFHB and Reactor Building Immobilized Decontamination Solution Waste

		Dose (mrem)[a]		
Location	Pathway	Total-Body	Bone	Liver
Nearest garden[b]	Inhalation	4.8×10^{-7}	1.7×10^{-6}	7.7×10^{-7}
	Ground Shine	5.7×10^{-6}	5.7×10^{-6}	5.7×10^{-6}
	Vegetable Use	4.0×10^{-5}	1.9×10^{-4}	1.3×10^{-4}
	Total	4.6×10^{-5}	2.0×10^{-4}	1.4×10^{-4}
Nearest milk goat	Inhalation	4.8×10^{-7}	8.5×10^{-7}	5.6×10^{-7}
	Ground Shine	5.7×10^{-6}	5.7×10^{-6}	5.7×10^{-6}
	Goat Milk Use	9.3×10^{-5}	6.5×10^{-4}	7.9×10^{-4}
	Total	9.9×10^{-5}	6.6×10^{-4}	8.0×10^{-4}
Nearest cow and garden	Inhalation	3.3×10^{-7}	1.2×10^{-6}	5.3×10^{-7}
	Ground Shine	5.7×10^{-6}	5.7×10^{-6}	5.7×10^{-6}
	Vegetable Use	4.0×10^{-5}	1.9×10^{-4}	1.3×10^{-4}
	Cow Milk Use	2.2×10^{-5}	1.0×10^{-4}	9.8×10^{-5}
	Total	6.8×10^{-5}	3.0×10^{-4}	2.3×10^{-4}

[a]Doses were calculated for total-body, GI-tract, bone, liver, kidney, thyroid, lung, and skin. The maximum three organ doses are listed in this table. Doses were calcuated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each group are listed. The dose estimates for total body exposure are for adults for all locations. For the bone and liver doses the estimates are for children for the nearest garden and nearest cow and garden locations and for infants for the nearest goat location.

[b]The basis for selecting the special locations is described in Appendix W. The actual locations are: nearest garden = 1.05 miles east-northeast, nearest milk goat = 1.02 miles north, and nearest cow and garden = 1.05 miles east.

8.2.5.4 Psychological-Socioeconomic Effects

See Section 8.3.5.4.

8.2.6 Economic Costs

Chemical-based decontamination solutions are broken down into two categories--treated and untreated. Treated solutions are those which are processed through an evaporator and then immobilized. Untreated solutions are those which are immoblized directly in vinyl ester styrene or cement and

require appropriate immobilizing facilities. The costs for packaging and handling these solutions are highly variable because of the uncertainty in the quantities to be handled. The quantities may vary from 47,000 gallons to as much as 550,000 gallons.

The staff believes that the costs for chemical decontamination solution immobilization lie somewhere between $2,000,000 and $13,000,000. Details supporting these cost estimates are presented in Appendix K.

8.3 SOLID MATERIALS

Decontamination of the auxiliary and fuel handling buildings (AFHB) began shortly after the accident, and decontamination of the reactor building is scheduled to begin in early 1983. These activities and defueling of the reactor, followed by primary system decontamination, will generate solid waste materials in the form of trash, contaminated equipment, and irradiated hardware. The characteristics of these wastes, the alternatives considered for their treatment, packaging, and handling, and the environmental impacts of these operations are discused in this section.

8.3.1 Status and Specific Considerations

The work performed to date in decontamination of the AFHB has not resulted in the generation of solid materials other than trash. The characteristics and disposition of the trash generated by AFHB decontamination activities through September 22, 1980, are given in Table 8.42.

This trash consisted of compactible and noncompactible solid material, some of which also is combustible. The compactible and combustible solids consist of disposable clothing, rags, plastic covers, laydown pads, and miscellaneous trash. The noncompactible solids consist of tools, hoses, safety goggles, miscellaneous construction materials, and other small items of equipment used by decontamination personnel. The compactible trash has been processed through a compactor and packaged in 55-gallon drums. This method of treatment reduces trash volume by about a factor of 5. Noncompactible trash has been packaged in 3-ft × 4-ft × 6.5-ft wooden low-specific-activity (LSA) boxes with a capacity of 80 ft^3 each.

8.3.1.2 Projected Requirements

Additional work in the AFHB and the other decontamination and defueling activities will lead to the generation of additional trash plus other solid waste materials. The estimated amounts of these other solid wastes that will be generated, their sources, and projected radioactivity levels are summarized in Table 8.43.

As shown, relatively large volumes of trash are projected to be generated during decontamination of the reactor building and during defueling of the core. Prior to August and September 1980 entries into the reactor building, it was assumed that this trash would have radioactivity levels about a factor of 10 higher than the material generated during AFHB decontamination (characterized in Table 8.42). However, the swipe samples taken during these entries did not substantiate this assumption, and it now appears that this additional trash will have gross radioactivity levels comparable to AFHB trash.

The gross activities shown in Table 8.43 have been revised from those presented in the draft PEIS to reflect containment entry data. The volume of trash that could be generated depends on the number of personnel involved in decontamination operations and on the time required to decontaminate areas to acceptable levels. The minimum and maximum volumes projected in Table 8.43 are bounding conditions for trash generation.

The condition of equipment in the reactor building which could be contaminated cannot be defined until decontamination begins. The estimated volumes shown in Table 8.43 were derived by identifying the equipment that could have been contaminated during the accident and then estimating best- and worst-case conditions for the portion of this equipment that would have to be disposed of as waste. Since the surface contamination level data obtained during reactor building entry shows levels below those assumed for the draft PEIS, the projected radioactivity levels shown in Table 8.43 have been reduced from earlier estimates. The contaminated equipment generated during defueling will consist of tooling and equipment placed in the spent fuel pool to assist with defueling; the volumes shown in the table are based on the staff's estimate of needs.

Table 8.42. Solid Waste Generated through September 22, 1980

Waste Form[a]	Waste Volume (ft^3)	Waste Treatment Process	Packaging					Disposition	
			Type	Volume (ft^3)	Number	Maximum Curies	Maximum Surface Radiation Level	Onsite Storage	Offsite Disposal
Compactible	72,300	Compaction[b]	Drum	7.35	1961	0.02	20 mR/hr	338	1623
Noncompactible	21,300	None	LSA Box	91	234	Neg.	15 mR/hr	62	172

[a]All solid waste generated through September 22, 1980, consisted of trash and rubbish.

[b]Volume reduction factor of 5.

Table 8.43. Solid Waste Projections

Waste Type	AFHB Volume (ft^3)	Reactor Building Cleanup Volume (ft^3)	Defueling Volume (ft^3)	Total Volume (ft^3)	Average Radioactivity (Ci/ft^3)	Estimated Inventory (Ci)
Minimums						
Compactible trash	115,000	60,000	50,000	225,000	0.006	1,350
Noncompactible trash	49,000	20,000	13,000	82,000	0.001	80
Contaminated equipment	NA	3,000	450	3,450	0.01	35
Mirror insulation	NA	15,000	NA	15,000	0.004	60
Irradiated hardware	NA	NA	3,200	3,200	0.01[a]	32
Total	164,000	98,000	66,650	328,650		1,550[b]
Maximums						
Compactible trash	115,000	200,000	140,000	455,000	0.006	2,730
Noncompactible trash	49,000	54,000	40,000	143,000	0.001	140
Contaminated equipment	NA	24,000	12,000	36,000	0.003	110
Mirror insulation	NA	15,000	NA	15,000	0.004	60
Irradiated hardware	NA	NA	3,500	3,500	0.01[a]	35
Total	164,000	293,000	195,500	652,500		3,080[b]

[a]The estimated curie content of irradiated hardware was assumed to be the same as the highest level contaminated equipment.

[b]Rounded to three significant figures.

The reactor vessel is surrounded by mirror insulation that was wetted during the accident. This contaminated solid material, which is neither trash or equipment, will have to be removed and packaged as solid waste. The estimated volume is based on the volume installed, and the radioactivity level is based on the residual radioactivity in this material after it has been dried.

The condition of the reactor vessel internals, which when removed may be handled as irradiated hardware, will not be known until the reactor vessel head is removed and the core internals are inspected. Therefore, the values given in this section for radiation levels for this hardware are estimates and could vary by an order of magnitude. It has been assumed that the upper plenum assembly and the core support structure will have to be removed and disposed of as radioactive waste.

8.3.2 Alternative Methods Considered

The alternatives considered for treatment, packaging, and handling of the solid wastes projected in Table 8.43 are discussed below.

8.3.2.1 Trash

Currently, trash is processed through a compactor to achieve a volume reduction factor of about 5. Volume reduction also could be achieved by burning combustible trash in an incinerator. A number of incinerators for radioactive waste have been operated. This technique would reduce the trash volume by a factor of 80 to 100; however, immobilization of the resultant ash will result in an effective volume reduction factor of 40 to 50. Consideration of these two alternatives and the range of volumes shown in Table 8.43 bound the volumes of trash that could be generated. These best- and worst-case conditions for packaged trash are as follows:

- Maximum Waste Package Production. The maximum volumes from Table 8.43, compaction of compactible trash to reduce volumes by a factor of 5, and packaging of noncompactible trash in 80-ft^3 LSA boxes.

- Minimum Waste Package Production. The minimum waste volumes from Table 8.43, incineration of 75 percent of the compactible trash to reduce effective volume by a factor of 50, compaction of the remaining 25 percent of the compactible trash to reduce volumes by a factor of 5, and packaging of noncompactible trash in 80-ft^3 LSA boxes.

The amounts of packaged trash that would be generated under these two conditions are summarized in Table 8.44.

Table 8.44. Minimum and Maximum Alternatives for Trash Generation

Factor	Compactible, Combustible Trash	Compactible,[a] Noncombustible Trash	Noncompactible, Noncombustible Trash
Minimum Generation			
Original volume (ft^3)	168,000	56,000	82,000
Process used	Incineration	Compaction	None
Volume reduction factor	50[b]	5	None
Package type	Drum	Drum	LSA box
Number of packages	480	1,600	1,025
Maximum Generation			
Original volume (ft^3)	341,000	114,000	143,000
Process used	Compaction	Compaction	None
Volume reduction factor	5	5	None
Package type	Drum	Drum	LSA box
Number of packages	9,740	3,260	1,790

[a]Assumes 25 percent of compactible trash is noncombustible.

[b]Incinerator ash is immobilized with volume increase factor of 2.

8.3.2.2 Contaminated Equipment

The decontamination of the reactor building, as discussed in Section 5.2, will require the removal and disposal of materials and equipment that cannot be decontaminated and refurbished to an acceptable operational condition. A partial listing of materials and equipment that may be removed for packaging and disposal is given in Table 8.45. Defueling and decontamination of the primary system will generate similar waste materials. A partial listing of this equipment and materials is given in Table 8.46.

The alternatives considered to package and handle these wastes involve volume reduction, but the method of volume reduction depends on the physical characteristics of the waste. The methods considered include:

- Disassembly - Large equipment items, such as pumps and motors, will be disassembled to reduce bulk and packaged volume.
- Sectioning - Large metal structures, such as the polar crane cab and ventilation fans, could be dismantled and cut up, or sectioned.
- Baling - Sheet metal and the mirror insulation could be baled in a press after the heavy structural fittings are removed. Baling will reduce the volume by a factor of 2 to 3.

In general, these techniques will reduce volumes by a factor of 2 but are not suitable for all equipment. For example, the reactor coolant pumps present some unique problems if they cannot be decontaminated in place for return to service. There are four pumps and four motors. Each pump is 8 ft in diameter, 15 ft long, and weighs about 75,000 pounds. The volume of each, excluding the casing, is about 750 ft^3, and the maximum expected radiation level is 5 R/hr. Each pump motor is 13 ft in diamater, 17 ft long, and weighs about 103,000 pounds. Each has a volume of about 2300 ft^3; the maximum expected radiation level is 10 mR/hr.

The packaging methods considered depend on the radioactivity level of the waste. Most of the material will be of low radioactivity and can be packaged in 80-ft^3-capacity LSA wooden boxes Other material, with radioactivity levels which require shielded transport, would be packaged in steel liners that can be placed in shielded overpacks. These liners would have a waste capacity of 70 ft^3.

When these alternatives are considered relative to the volumes shown in Table 8.43, bounding conditions for packaged contaminated equipment can be defined. The bounding conditions, which represent minimum and maximum volumes of packaged waste, are as follows:

- Maximum Waste Package Production. The maximum waste volumes and activity levels in Table 8.43, packaging in LSA boxes or liners, and no volume reduction for equipment or mirror insulation.
- Minimum Waste Package Production. The minimum waste volumes and activity levels in Table 8.43, decontamination and reuse of the reactor coolant pumps, volume reduction of 50 percent of the remaining equipment by a factor of 2, and volume reduction of mirror insulation by a factor of 2.5.

The quantities and characteristics of waste packages arising from these bounding conditions are shown in Table 8.47.

8.3.2.3 Irradiated Hardware

The material that would be removed from within and around the reactor vessel as irradiated hardware is characterized in Table 8.48. The alternatives for packaging and handling these wastes depend on their physical characteristics and radiation levels. Disassembly and sectioning could reduce volumes by a factor of 2. Low-activity material could be packaged in 80-ft^3-capacity LSA boxes, while higher activity material could be packaged in 70-ft^3-capacity steel liners compatible with shipment in a shielded overpack.

Table 8.45. Partial Listing of Reactor Building Contaminated Equipment for Disposal

Air Supply System
Purge system:
Vertical mounted fans
Samplers
Coolers
Air cooler portion of the main system
Main system ductwork on the 305-ft level
Main system ductwork above the 305-ft level but not including that within the secondary shield system
Main system ductwork below the 305-ft level but not including that within the secondary shield system
Registers, dampers, relief valves, etc.
Chiller plenum
Fans and coolers
Ductwork within the secondary shield system

Polar Crane
Motors
Wiring
Hoist rope
Contacts and controllers

Steam Generator
Mirror insulation
Cables, wire trays, conduit
Instrumentation
Main coolant pump motors
Pump coolers
Lubrication pump systems
Electrical cabinet

Ancillary Equipment
Instrument packs
Control stations
Drain pumps
Letdown coolers
Crane
Valves
Wire trays
Cables
Small piping
Instrument piping and connectors
Light fixtures
Mirror insulation
Small tools and fixtures

Elevator
Shaft block walls
Drive motors
Electric cables
Switches, relays
Hoist cables
Elevator car

Fuel Handling Bridges
Bridge assembly
Electricals
Mechanisms
Missile shields