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Documented Safety Analysis Addendum for the Neutron Radiography Reactor Facility Core Conversion

Safety Analysis Working Group

Boyd D. Christensen

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Boyd D. Christensen
Idaho National Laboratory
P. O. Box 1625, Idaho Falls, Idaho, 83415
208-533-7914 (208-533-7239 fax)
boyd.christensen@.inl.gov

Abstract

The Neutron Radiography Reactor Facility (NRAD) is a Training, Research, Isotope Production, General Atomics (TRIGA) reactor which was installed in the Idaho National Laboratory (INL) Hot Fuels Examination Facility (HFEF) at the Materials and Fuels Complex (MFC) in the mid-1970s. The facility provides researchers the capability to examine both irradiated and non-irradiated materials in support of reactor fuel and components programs through nondestructive neutron radiography examination. The facility has been used in the past as one facet of a suite of reactor fuels and component examination facilities available to researchers at the INL and throughout the DOE complex. The facility has also served various commercial research activities in addition to the DOE research and development support.

The reactor was initially constructed using Fuel Lifetime Improvement Program (FLIP)-type highly enriched uranium (HEU) fuel obtained from the dismantled Puerto Rico Nuclear Center (PRNC) reactor. In accordance with international non-proliferation agreements, the NRAD core will be converted to a low enriched uranium (LEU) fuel and will continue to utilize the PRNC control rods, control rod drives, startup source, and instrument console as was previously used with the HEU core.

The existing NRAD safety analysis report (SAR) was created in 1977. The format combines required sections from both DOE-STD-3009 and Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70. In order to support the NRAD refueling and reactor operations with the LEU core, an addendum to the NRAD SAR was developed. This addendum follows the existing NRAD SAR format, combining required formats from both DOE and NRC. New accident analyses were performed to support the new fuel and characteristics of the new core.

This paper discusses the analysis performed for the core conversion and the process of writing a compliant and approved addendum to the existing safety basis documents.

1. Introduction

The Idaho National Laboratory (INL) is a government-owned reservation located in southeastern Idaho, approximately 25 miles west of Idaho Falls, Idaho, and covers an area of approximately 890 mi^2. The INL is currently operated by Battelle Energy Alliance (BEA). Current missions of the INL include developing nuclear reactor technologies, national security programs, advanced fuel development, spent fuel treatment, and other science and technology programs.

The Materials and Fuels Complex (MFC) is the easternmost facility located on the INL. MFC contains several facilities, including the Hot Fuel Examination Facility (HFEF), an inert atmosphere hot cell facility. The HFEF was constructed to support irradiated fuel and hardware examination programs for EBR-II and other DOE-complex wide projects. The Neutron Radiography Facility (NRAD) is a Training, Research, Isotope Production, General Atomics (TRIGA) reactor which was installed in HFEF in the mid-1970s. The facility provides researchers the capability to examine both irradiated and non-irradiated materials in support of reactor fuel and components programs through nondestructive neutron radiography examination.

Pursuant to the Reduced Enrichment for Research and Test Reactors (RERTR) Program, DOE determined that the NRAD core would be refueled with a low enriched uranium (LEU) fuel, allowing NRAD to continue support for DOE missions with similar performance but with lower proliferation attractiveness fuel. The RERTR program is managed by the Office of Nuclear Material Threat Reduction of the National Nuclear Security Administration (NNSA) and has an international program mission to develop technology necessary to enable civilian reactor programs to convert HEU fuel (fuel containing ≥20% U-235) to LEU fuel (containing <20% U-235) as part of the DOE nonproliferation efforts. To date, more than 40 research reactors have been converted from highly enriched uranium (HEU) to LEU fuel cores. Among the LEU fuel types developed and qualified under the RERTR program, $UZrH_x$ was developed by General Atomics (GA) for use in the TRIGA reactors with power levels up to 14 MW. This fuel type will be used as the replacement fuel for NRAD as well as other TRIGAs previously converted from HEU to LEU fuel.

The existing NRAD SAR was first completed in 1977,[1] and its format combines sections of both DOE-STD-3009[2] and Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70.[3] The documented safety analysis (DSA) addendum discussed in this paper documents the safety basis for converting NRAD from the original PRNC HEU TRIGA- Fuel Lifetime Improvement Program (FLIP)-type fuel to LEU (30/20) and operating NRAD with the LEU (30/20) fuel. After the core conversion, the NRAD reactor will continue to utilize the PRNC control rods, control-rod drives, and instrument console and startup source used with the HEU core.

This paper will provide an overview of the NRAD reactor giving a description of the facility and the reactor operations and capabilities. It will then discuss the core conversion process and the considerations in developing a safety basis for LEU operations.

2. Facility Description

Overview

The NRAD reactor was obtained from the PRNC and installed in the basement of the HFEF in the mid-1970s to compliment the in-cell fuel and hardware examination capabilities of Argonne National Laboratory. With minor modifications, the reactor was used for examinations using neutron radiography for both in-cell specimens and cask transported specimens through the utilization of two separate neutron beam tubes originating in the reactor core and terminating in personnel accessible cells where plates or foils could be irradiated in a process of indirect radiography. The reactor is also utilized for irradiation experiments using a specimen tube directly into the reactor core. The reactor is conducive to frequent startup and shutdown cycles allowing operators to access the experiment tube to insert or remove experiments as needed.

A Manfred Frey Physics Corporation Model A-711 high output neutron generator has been added to the facility to expand capability to perform neutron interrogation experiments on waste containers and fuel subassemblies. The neutron generator is classified as a moderate hazard facility under the DOE guidance document 5480.25, "Guidance for an Accelerator Facility Safety Program,"[4] meaning that the activity has hazards which have the potential for presenting considerable on-site impacts to people or the environment but only minor off-site impacts at most. This classification results from the potential for worker radiation exposures in excess of the limits listed in 10 CFR 835, "Occupational Radiation Protection."[5] This type of exposure could only occur in the event of blatant violation of procedures and training and deliberate circumventing of the safety interlock system on the neutron generator.

The reactor, though located within HFEF, is physically separated from the HFEF hot cells. A gas-tight specimen tube does, however, extend from the inert atmosphere main cell into a radiography cell which also receives one neutron beam tube from the reactor core. The reactor itself is an open tank pool-type, solid uranium fuel, water-moderated, water-cooled reactor which operates at a maximum thermal power of 250 kW in the steady-state power mode. In addition to supporting fuel and material inspection, the reactor may be used for activation analysis, including filtered and tailored neutron spectra and isotope production. Out-of-tank neutron experiments are conducted as needed in the radiography subcells and have no impact on reactor operation or safety. Experiments are subject to their own safety analysis for potential of nuclear criticality, radioactive contamination events, or unplanned radiation exposure to workers.

NRAD operates as a Category B reactor and has been classified as a Hazard Category 2 nuclear facility according to DOE-STD-1027-92[6] based on the estimated inventory of radioactive material in the facility. That categorization has been appropriate for the HEU core and will continue with the LEU core.

NRAD Reactor Control Room

The NRAD reactor control room is located on the operating floor of HFEF and adjacent to the HFEF control room. Access to the control room is restricted by cipher lock. The NRAD control room contains operating and control equipment such as: reactor control console, uninterruptible power supply, reactor operation annunciator panel, reactor operation indicators and controls, cooling system indicators and controls, radiation monitoring instrumentation, and a video monitor from cameras in the reactor room and radiography stations.

NRAD Reactor Room

The NRAD reactor room is located below ground grade level at the service floor level of HFEF under the argon atmosphere main cell and contains the following key equipment and components: reactor tank, reactor top shield, cooling water system pumps and controls, temperature transmitters, control rod drive mechanisms, and radiation monitors.

The reactor tank occupies most of the reactor room. The room is accessible at the top of the reactor. The tank is embedded in a massive high density borated concrete block which provides a working platform at the top of the reactor and also provides support to the tank walls and a thick biological shield for the reactor operations.

Figure 1 shows the service floor of HFEF along with the NRAD reactor room and associated support areas.

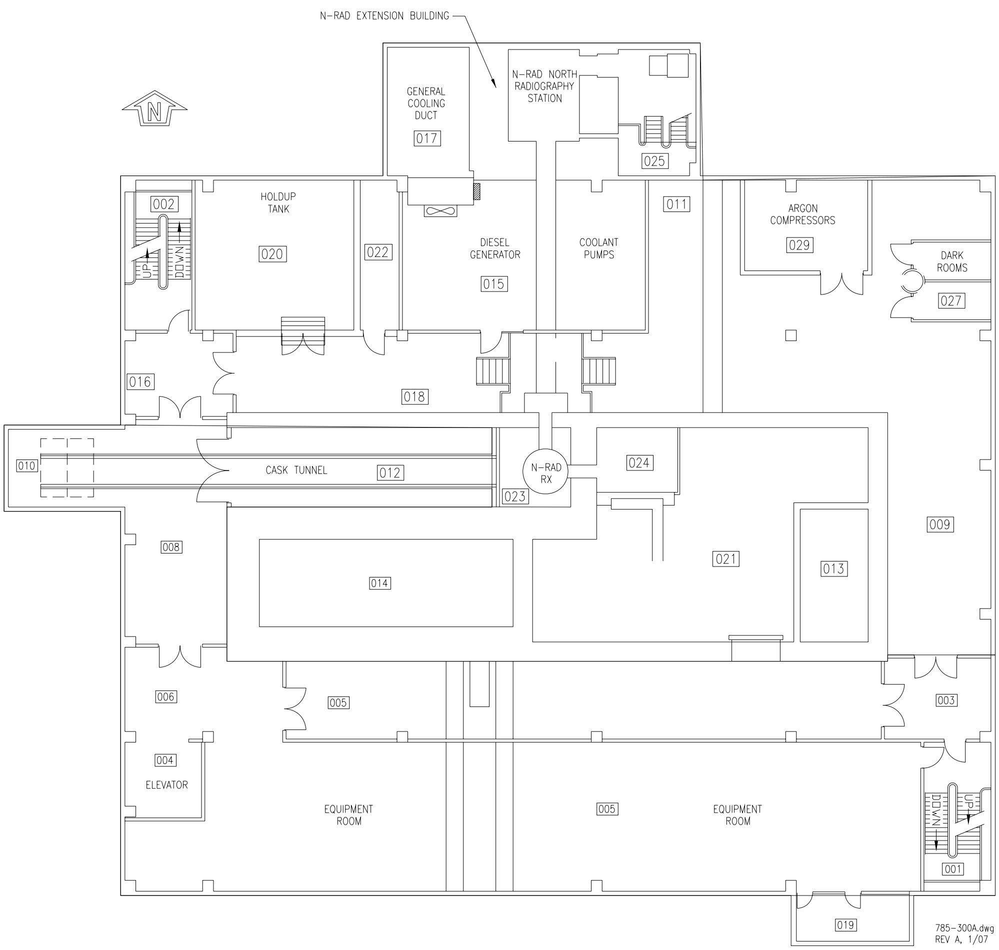


Figure 1. Plan view of HFEF service floor.

The reactor top shield is a 2-in.-thick steel slab divided into three sections. One section is fixed and provides structural support to the control rod drives. The other two sections are moveable to provide removable shielding to either shield or allow access to in-tank support activities as needed. The steel slabs are all sized such that they cannot fall into the tank pool area damaging the reactor core.

The reactor room ventilation system maintains the reactor room at a lower air pressure relative to the surrounding occupied areas in order to prevent the spread of contamination. As radioactive gases are generated, the effluent air passes from the reactor room through HEPA filters and eventually out the HFEF stack. The effluents are monitored in the stack to prevent unauthorized releases to the environment.

Auxiliary and Support Areas

Auxiliary systems and support areas are located surrounding the reactor room and consist of the NRAD heat exchanger and water treatment room, north and east radiography stations, north and east neutron beam tubes, a cask handling room, a photography darkroom, and a Manfred Frey Model A-711 neutron generator and its associated support equipment. Figure 2 depicts the general layout of the facility and support areas.

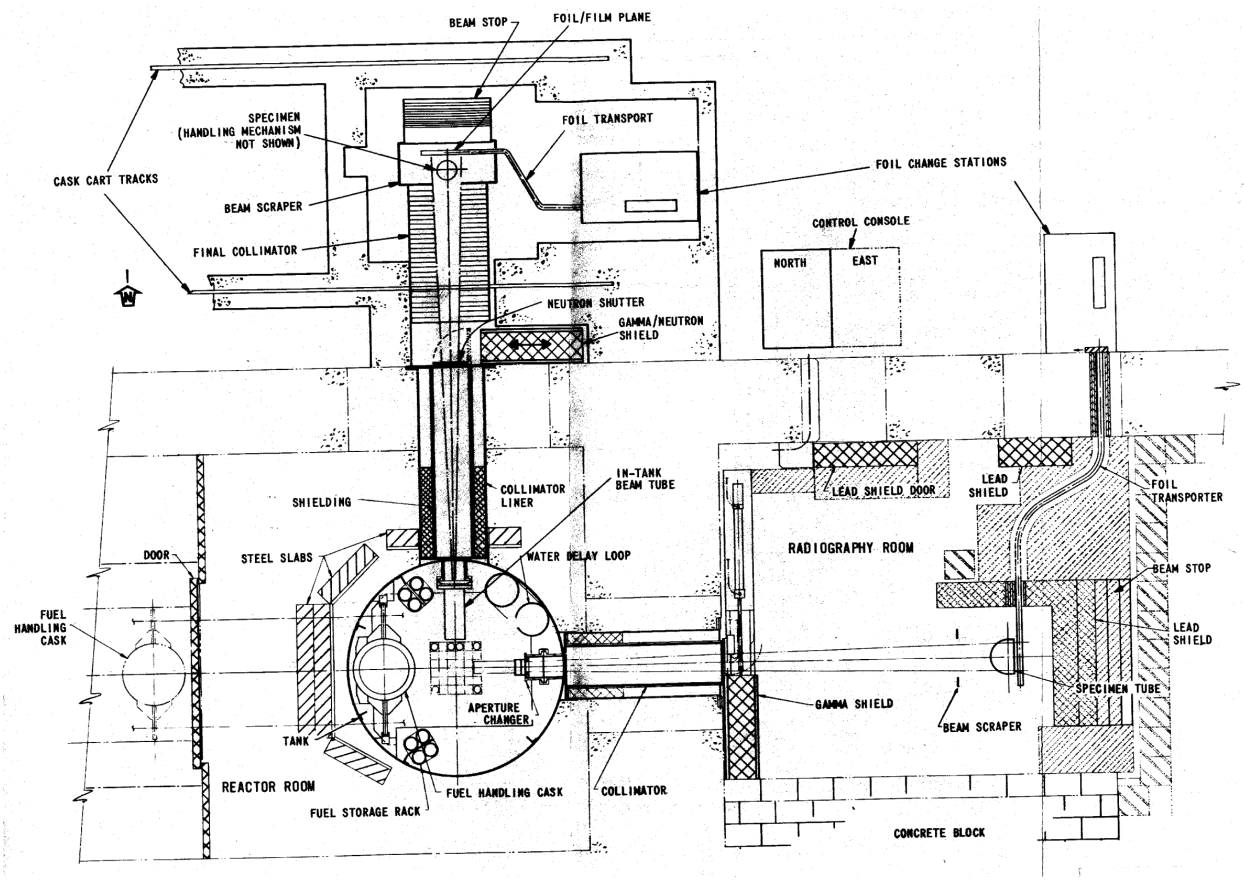


Figure 2. Plan view of NRAD facility.

Figure 3 shows a diagram and key features of the north radiography station where specimens transported by shielded cask can be examined by neutron radiography.

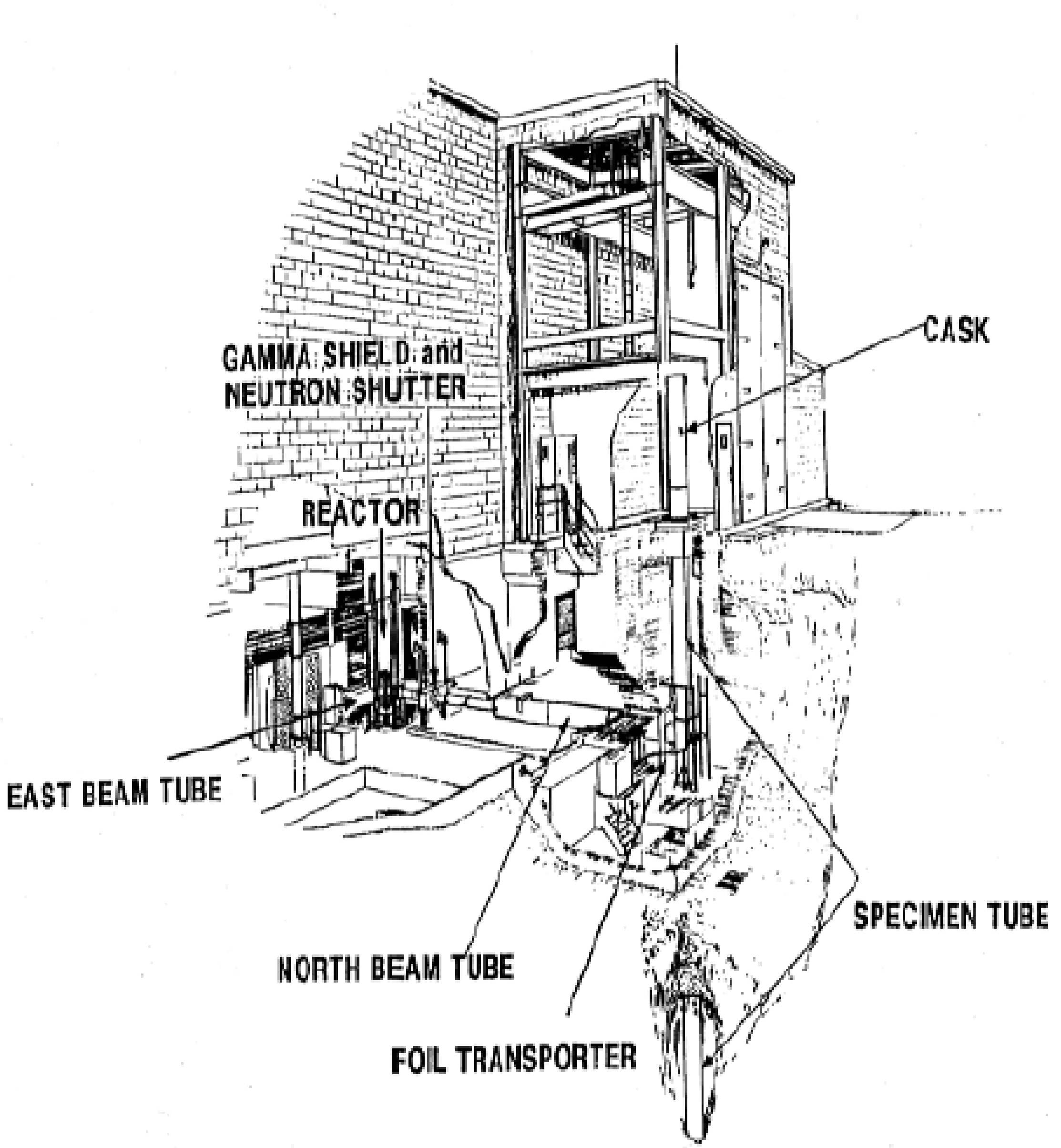


Figure 3. NRAD North radiography station.

NRAD Reactor Operation

The NRAD reactor is operated at a nominal power of 250 kW thermal in steady-state operations. The reactor core is cooled by natural circulation of water within an open reactor tank. Heat is removed from the system by forced circulation of primary water and through a water-to-water heat exchanger. The secondary cooling water is directed to the HFEF process heat cooling tower. Reactor power operating levels are limited by the cooling capacity. The primary cooling water system includes a demineralizer system for maintaining water purity and clarity. The demineralizer system consists of a bypass from primary flow through a series of fiber filters and a mixed resin bed canister bank. Primary coolant water quality is monitored routinely to maintain acceptable conditions.

The initial fuel loading in the reactor core consists of 16 three- and four-element clusters. Three core locations with three-element clusters utilize the fourth element position as a control rod location. The fourth three-element cluster has an experiment tube at the fourth element position which can be used for in-core irradiations for foil activation, sample irradiation, or isotope production. Graphite reflector blocks surround the core to improve neutron retention efficiency.

Out-of-core neutron radiography is performed in a unique process whereby a collimated beam of neutrons from the reactor pass through a test object. The neutrons will differentially either be scattered, absorbed, or pass through the specimen test object, depending on density and the neutron cross section attenuation of the specimen. Neutrons emerging through the sample activate a large foil which is later placed against industrial x-ray film which is selectively exposed by the photon emissions from the activated foil as it decays. The x-ray image is then preserved as a hard copy radiographic image on the film.

In-tank irradiations and experiments include neutron spectrum research, activation analysis, and isotope generation. An experiment irradiation tube is placed in an unused fuel cluster element position. The in-core tubes are manufactured to the same dimensions as the fuel elements in order to prevent any coolant flow interruptions. A pneumatic transfer system is being designed which will quickly transport a sample into the core for irradiation, then back out and into a shielded examination facility for analyses in support of reactor fuel and material development missions. In-tank removable experiments are limited to a reactivity worth not to exceed 50¢ total cold-critical reactivity to protect the assumptions made in reactivity accident analyses in the safety basis document.[7]

The reactor uses a 5 Ci americium-berylium source for reactor startup which is permanently located within the tank. The source is contained within an aluminum cylinder and is used to induce initial fission events to produce a measureable reactor power level indication during reactor startup.

Figure 4 shows the in-tank positions of neutron beam tubes, instrumentation, fuel clusters and elements, control rods, graphite reflector blocks, start-up source, and experiment positions as explained in this section.

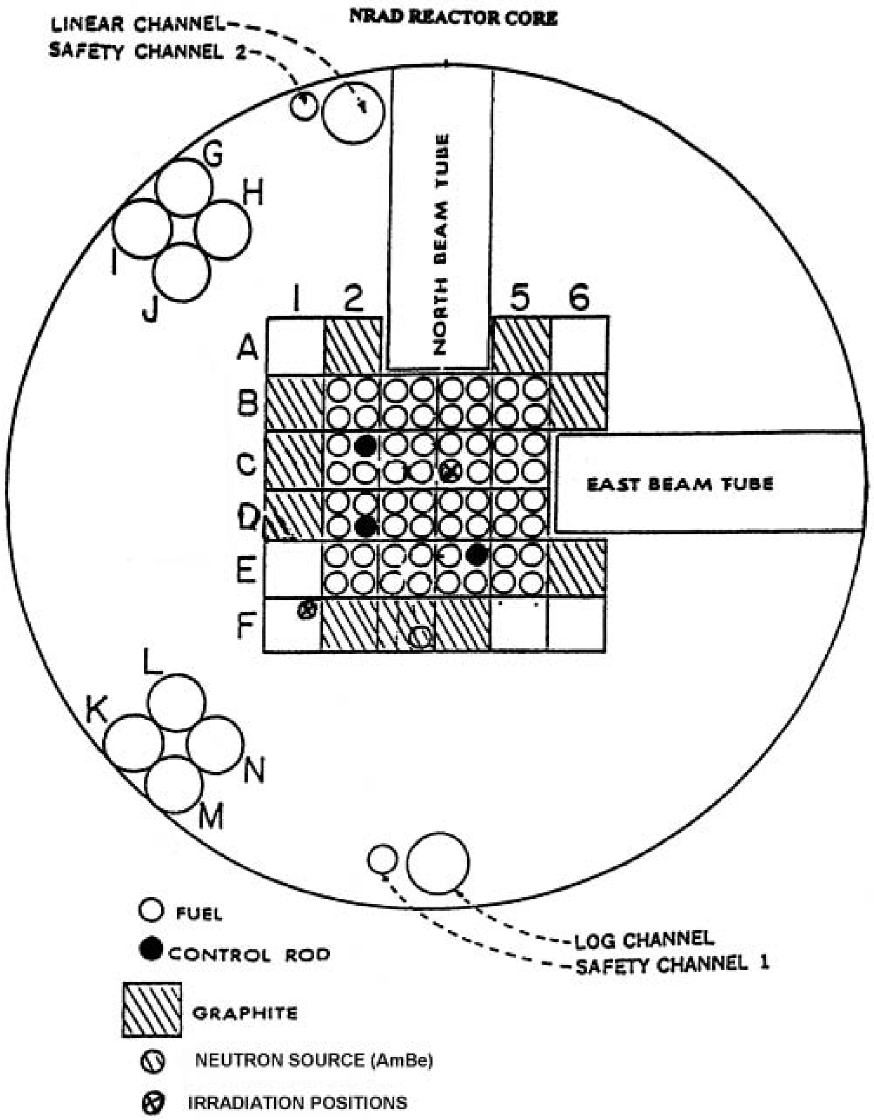


Figure 4. NRAD tank layout.

3. Core Conversion and SAR Addendum

The NRAD reactor fuel is the original fuel used in the PRNC core, which is a TRIGA-FLIP HEU (70% U-235) fuel. The fuel design is a well established, proven, and safe design and is used in reactors at much higher power levels than NRAD. The fuel elements in the NRAD reactor operate at about 25% of the design power density. The U-235 constitutes 8.42 wt% of the fuel element. The elements also contain a uniform dispersion of 1.48 wt % natural erbium, which is used as a burnable poison to maintain consistent performance over the life of the fuel. A hole is drilled through the center of the active fuel section and filled with a zirconium rod subsequent to hydriding. The fuel rods are clad with stainless steel. Figure 5 shows the typical fuel element and cluster assembly design.

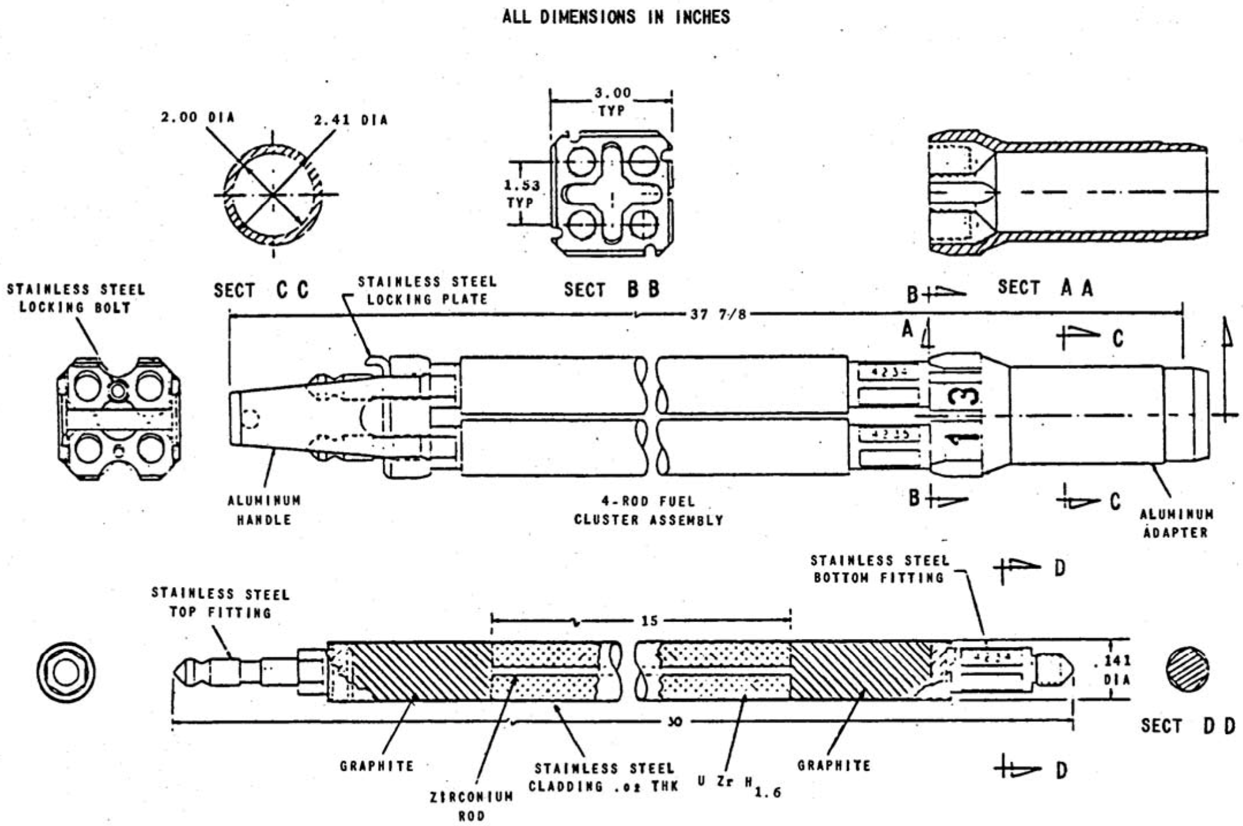


Figure 5. NRAD fuel element and four-element cluster.

As with other civilian research reactors, the RERTR program requires that NRAD be fueled and operated with a low enrichment core to reduce the attractiveness level of the fuel to support non-proliferation initiatives. Several advanced LEU fuels have been developed which will meet the performance needs of these reactors and also meet the goals of the RERTR program. The fuel selected was identified as the best pick to provide comparable performance in low power TRIGA reactors while still maintaining the inherently safe features available to this type of reactor.

The new fuel to be used in the NRAD core is an LEU 30/20 fuel (19.75% enriched) uranium-zirconium-hydride (UZrH) in a similar element and cluster configuration as the HEU FLIP fuel. The core will consist of 60 elements placed in three- and four-element clusters for a total of 16 clusters and 60 elements. Four element positions will remain open in the clusters to accept the three control rods and a water hole for in-core experiment placement. Figure 6 shows the configuration of the NRAD LEU core.

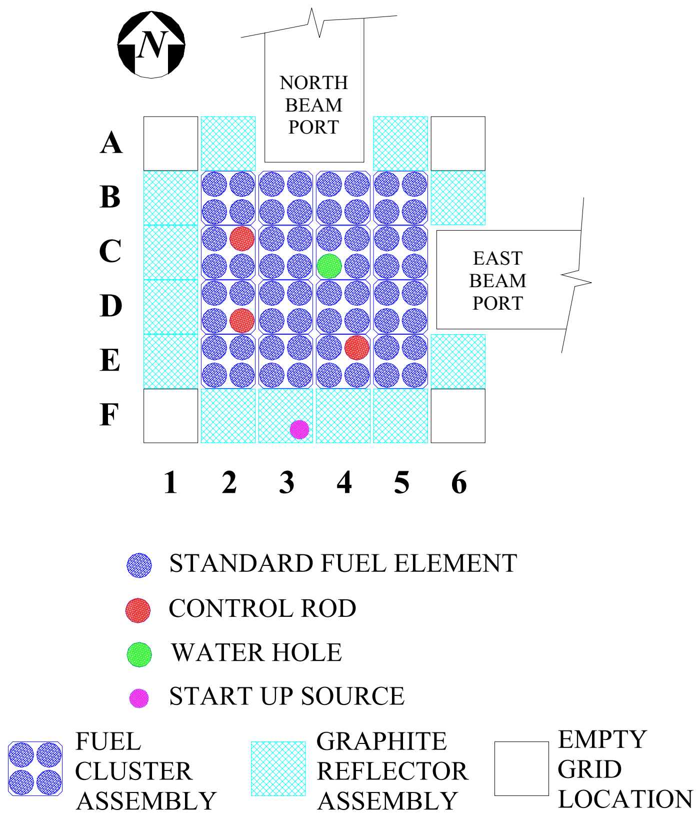


Figure 6. NRAD LEU core configuration.

It was decided that an addendum was the most practical method of addressing the safety of the new fuel in the core since facility operations are unchanged. An addendum can be completed in a timely manner to cover the operation of the facility while a new SAR is prepared. A significant effort will follow with a total SAR upgrade to be completed in 2012. At that time the LEU fuel conversion will be incorporated into the new SAR base document.

LEU Fuel Analysis

Dimensions of the LEU fuel elements are identical to those of the HEU elements. Total uranium mass and density increase with the LEU fuel over the HEU core, and erbium weight percent decreases from 1.48% in the HEU to 0.90% in the LEU, given the decreased need for burnable poison in a lower enrichment fuel matrix.

Based on the LEU core configuration, it is concluded that:

1. The shutdown margin meets the required limit for safety

2. The reactivity coefficients remain essentially the same as for the HEU FLIP core

3. Fuel integrity is maintained under all operating conditions (see detailed fuel conclusions below).

   Fuel Integrity Conclusions:

- The temperature limit for the fuel element is determined by the hydrogen gas over-pressure in the zirconium hydride and the ultimate strength of the cladding, both as functions of temperature. An element would be expected to fail at the temperature where the stress produced by the overpressure equaled the ultimate strength of the cladding. The analyses show that the cladding should not fail during short, transient operations even if fuel temperatures reach 1,150°C, and experimental data are available to confirm this analytical prediction.[8] Figure 7 shows the relationship between stress induced in the cladding from hydrogen pressure and the ultimate strength of the cladding as functions of temperature.

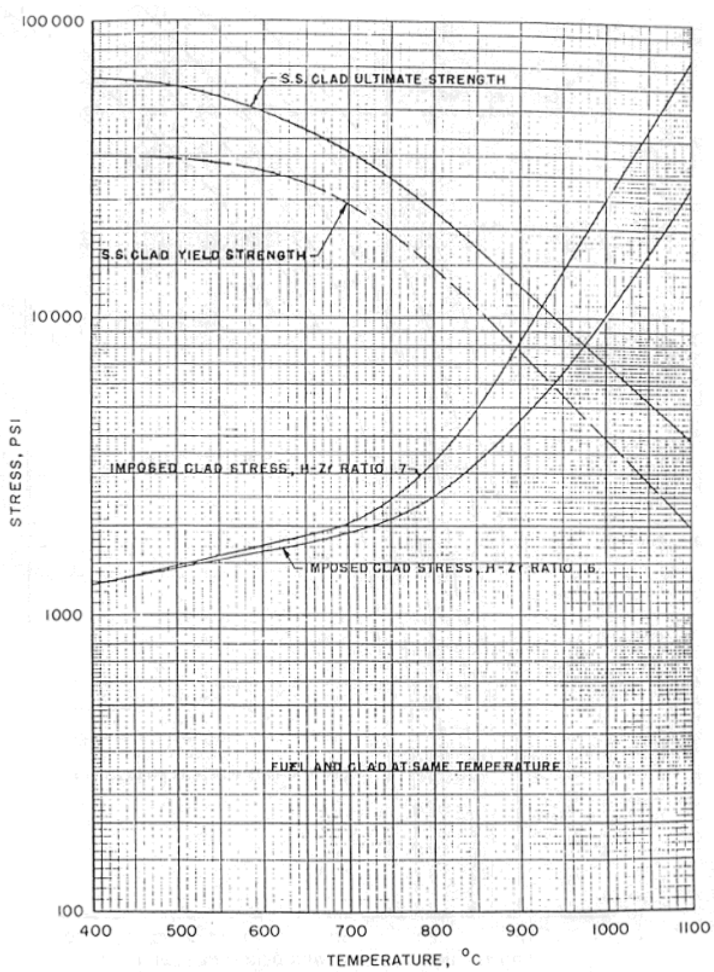


Figure 7. Cladding strength and applied stress as functions of temperature.

- For steady-state operating conditions at very high power densities or during loss-of-coolant conditions, it is conservative to assume that the fuel and cladding temperatures are equal. Based on this assumption and the data presented in Figure 7, the expected failure point for these conditions is about 975°C for a hydrogen to zirconium ratio of 1.6. The manufacturer's fuel design temperature limit when the cladding and fuel temperature are equal is 950°C. For conservatism, the temperature limit established for these conditions in the PRNC technical specifications is lower than the temperature where the overpressure causes the cladding stress to equal the yield strength; thus an equilibrium temperature limit of 900°C was established for the PRNC reactor, and this same limit is retained for the NRAD reactor with either fuel. At the NRAD operating power level of 250 kW_t, maximum fuel temperature is 344°C, well below any design safety limits.

- Experiments have shown that the uranium-zirconium-hydride fuel systems have a relatively low chemical reactivity with respect to water and air.[9] Therefore, a cladding failure that exposes the fuel to coolant will not produce a safety hazard.

- Irradiation data for zirconium hydride fuel at temperatures of 700 to 800°C at atom-burnup fractions comparable to those expected at the end of life (EOL) for FLIP fuel have not shown noticeable effects on the fuel.

The calculated radial power factor in an individual NRAD LEU fuel element is shown in Figure 8. The axial power-distribution curve calculated for the NRAD LEU core is shown in Figure 9. The axial peak-to-average power ratio of 1.28 for NRAD LEU, which was obtained from a two-dimensional calculation, is nearly the same as that for the TRIGA-FLIP system. The radial peak-to-average element power density for NRAD LEU, from a two-dimensional calculation, is 1.593, with the peak power occurring in the elements adjacent to the reactor centerline.

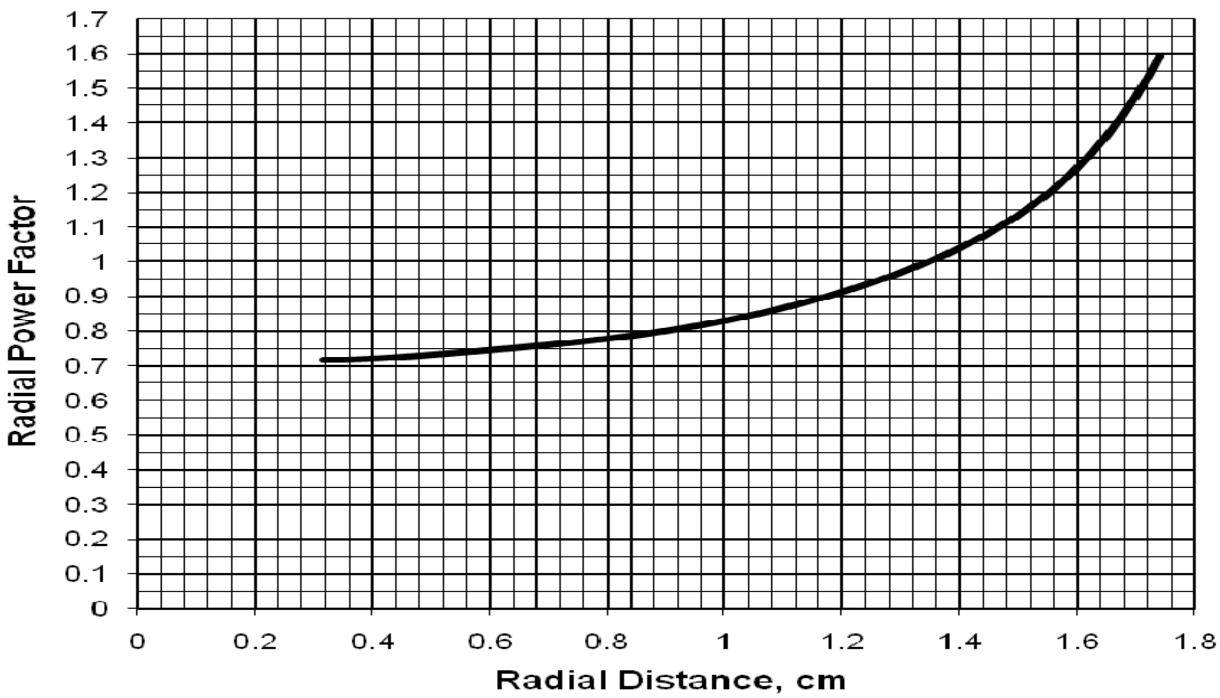


Figure 8. Radial intra-rod power factor versus fuel element radial distance.

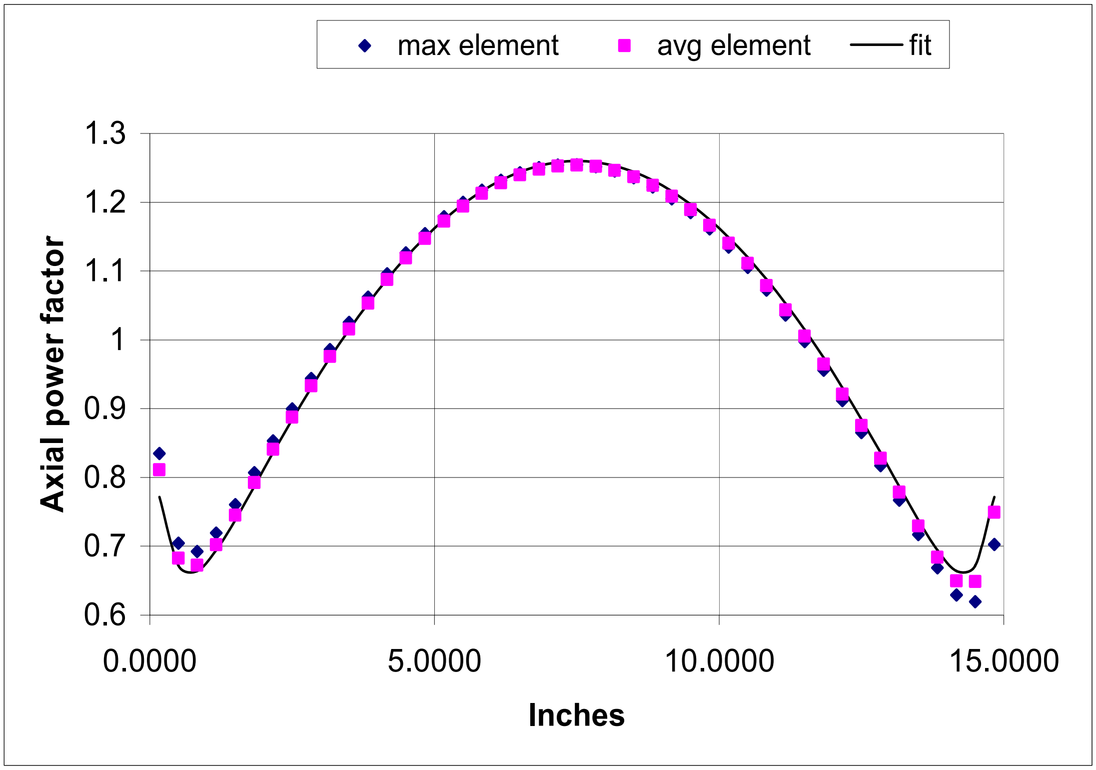


Figure 9. NRAD axial power profile vs. distance from bottom of fueled section (inches).

The change in power-peaking factors with burnup in an NRAD LEU core is discussed in detail in a GA report on core characteristics of the NRAD LEU.[10] The conclusion drawn from these data is that the power distribution within the NRAD LEU core does not change more than a few percent during the lifetime of the core, and this change is too small to have any influence on safety considerations.

The prompt negative temperature coefficient of TRIGA fueled reactors is an important feature in the inherent safe operation of TRIGA reactors. The results for the NRAD LEU for beginning-of-life (BOL) and EOL and NRAD HEU are listed in Table 1.

Table 1. Reactivity change with temperature.

Avg. Core Temperature (°C)	α (Δk/k-°C), PRNC HEU	α (Δk/k-°C), NRAD LEU (BOL)	α (Δk/k-°C), NRAD LEU (EOL)
23-200	4.77×10^{-5}	5.38×10^{-5}	5.15×10^{-5}
200-280	7.41×10^{-5}	7.03×10^{-5}	6.54×10^{-5}
280-400	9.70×10^{-5}	8.46×10^{-5}	7.69×10^{-5}
400-700	14.1×10^{-5}	11.1×10^{-5}	9.71×10^{-5}
700-1000	17.9×10^{-5}	13.4×10^{-5}	11.4×10^{-5}

Figure 10 is a histogram with a best polynomial fit plot of the computed values for α in Table 1 as a function of core temperature for both HEU and LEU (BOL and EOL).

In Figure 10, it can be seen that the prompt negative temperature coefficient (α) for NRAD LEU fuel has only a modest decrease in value at 3,000 effective full-power day (EFPD) burnup (EOL) (e.g., 7.0×10^{-5} to 6.5×10^{-5} Δk/k-°C at 200°C). As illustrated, the corresponding decrease for FLIP fuel is much larger, as an example, ~17.5×10^{-5} to 11.5×10^{-5} Δk/k-°C at 700°C for 2,000 MW burnup. The relatively small change in α for LEU 30/20 fuel is expected due to the 80 wt-% U-238 in NRAD LEU fuel compared to 30 wt-% U-238 in HEU FLIP fuel. It is the 80 wt-% U-238 in standard TRIGA fuel that is responsible for the nearly temperature independent value of ~10×10^{-5} Δk/k-°C.

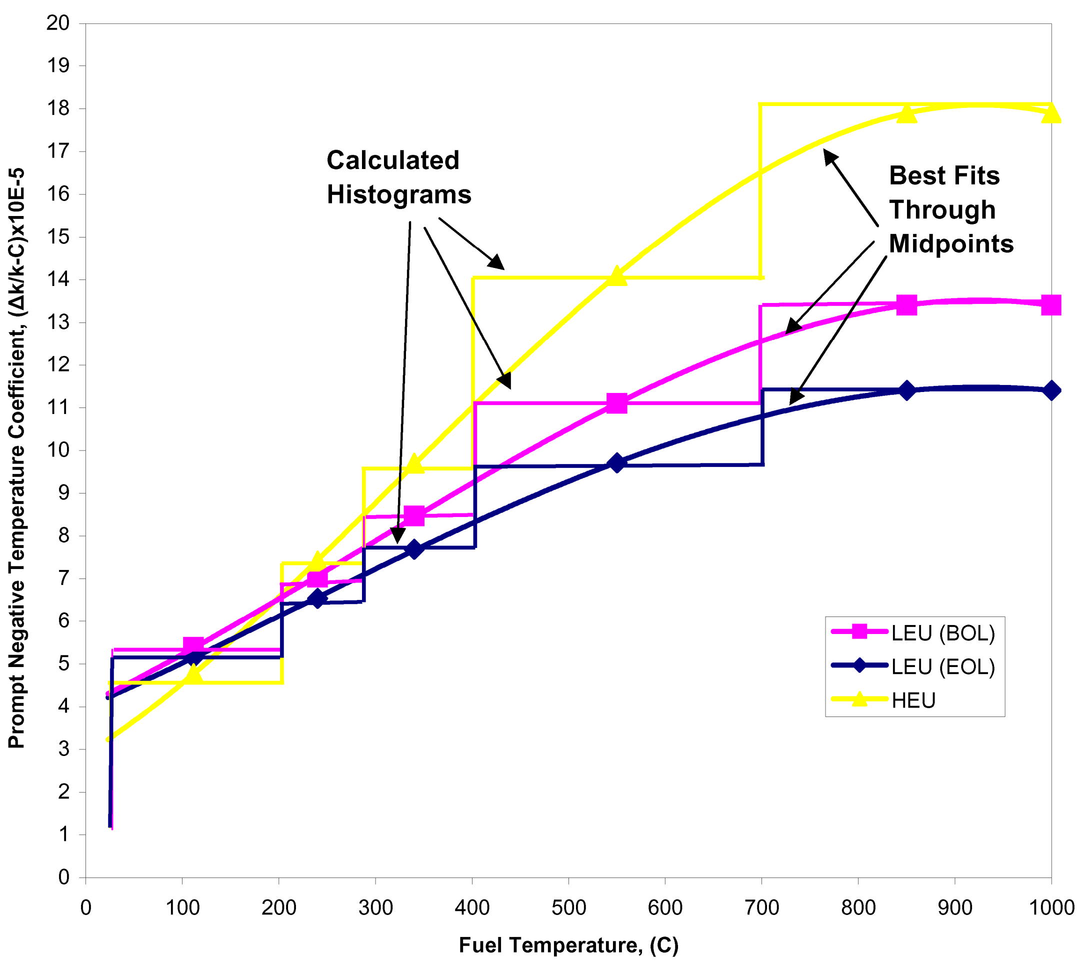


Figure 10. Prompt negative temperature coefficient for LEU (BOL and EOL) and HEU fuel.

Since the water temperature coefficient of TRIGA systems has not been studied extensively either theoretically or experimentally, a new analysis provides help in verifying the assumptions noted in the current NRAD DSA with regard to the overall moderator temperature coefficient. The overall moderator temperature coefficient is the combined effect of both temperature and density reactivity coefficients. This overall coefficient has been found to be similar in effect to previous evaluations of temperature and density. In addition, as stated in the DSA, based on the standard TRIGA isothermal-reactivity-coefficient data from "Experimental Determination of the Total Isothermal Reactivity Feedback Coefficient for the University of Arizona TRIGA Research Reactor,"[11] the overall moderator temperature coefficient changes sign at about 75°C. The current analysis agrees well with this, changing sign at about 72°C. The coefficient results are plotted in Figure 11 in terms of per cent mille (pcm) per degrees Celsius, where pcm equals 10^{-5} of $\Delta k/k$. For comparison, the fuel and isothermal temperature coefficients given in the NRAD DSA and Zagar[12] are utilized to calculate and plot a corresponding data point. In any case, the moderator coefficient is too small to significantly influence the steady-state or transient characteristics of the reactor. There is roughly a 6 pcm/°C (or 0.00006 $\Delta k/k$ per degrees Celsius) change in the curve of Figure 11.

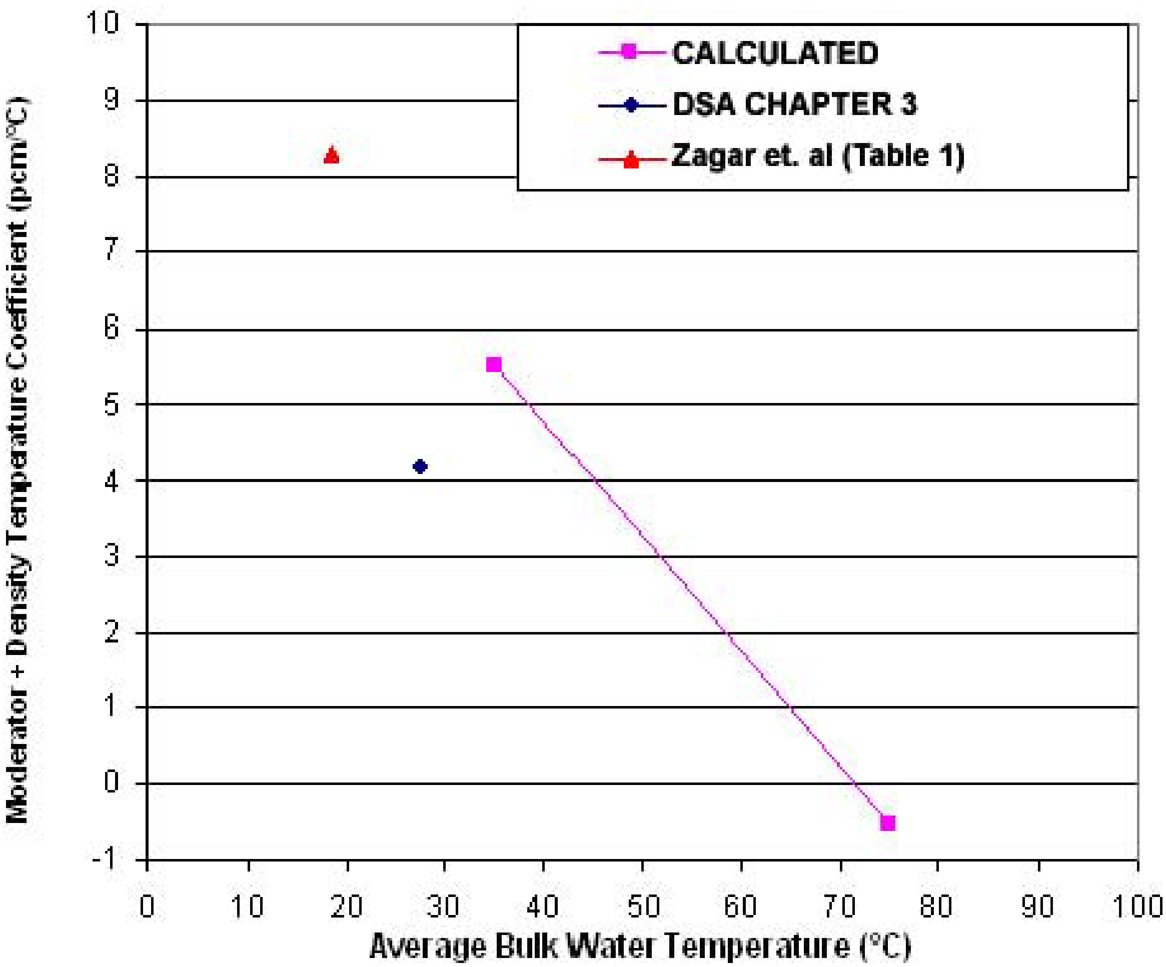


Figure 11. Overall moderator temperature coefficient comparisons.

In addition to temperature and density reactivity factors, reactivity can also be affected by void in the cooling water and also a reactivity feedback produced by the loss of coolant. In the case of void coefficient, it is found that the change in reactivity is negative as the percent void increases.

These changes in reactivity are relatively minor and not a nuclear safety concern since they only enhance the previously established inherent safety aspect of the fuel. In the case of loss of coolant, the reactivity change is strongly negative which ensures that the reactor would shut down without operator intervention in that type of scenario.

Build up of xenon poison in the fuel was calculated and found to not pose operational or safety consequences. The NRAD reactor is a single shift operation and calculations show that xenon equilibrium is not achieved before 60 hours of continuous operation. Operations not exceeding eight hours per day and five days per week will not result in reactivity issues from xenon poison even at core end of life.

The thermal and hydraulic parameters for the NRAD reactor are similar to those of other reactors containing FLIP fuel. Those reactors have routinely operated at steady power levels of 1 MW and with transients initiated by reactivity insertions greater than $2.00. The PRNC reactor has operated at power levels up to 2 MW. The primary differences between the NRAD reactor and the PRNC reactor are in tank size, number of fuel elements, and normal operating power. From this premise, it is apparent that the fuel in the NRAD reactor will operate under much less severe steady-state thermal conditions than HEU fuel in the PRNC and other reactors. The NRAD reactor will operate at much lower power levels and will not be operated in the transient mode.

Updated Hazard and Accident Analysis

Since TRIGA-type reactors have been widely used in the U.S. under NRC license, the hazards associated with its operation have been studied in depth, are well defined, and are documented in Chapter 6 of the NRAD DSA addendum for refueling. That chapter identifies and evaluates potential hazards associated with the conversion of the NRAD reactor fuel from HEU to LEU and subsequent operation of the reactor as described above. That chapter also contains the quantitative consequence analyses of postulated representative, bounding, and/or unique accidents selected from the hazard analysis. Reactor operations after the fuel conversion involve no new hazards; the hazards associated with the refueling effort were qualitatively analyzed and included in the hazard and accident chapter of the addendum. The following paragraphs describe the methodology used to quantify consequences of accidents considered in the core conversion process.

Hazards, Accidents, and Likelihood

The likelihood (e.g., anticipated, unlikely, extremely unlikely, or beyond extremely unlikely) of each hazardous event without controls was qualitatively estimated using the definitions in Table 2. No credit was taken for controls (design or administrative features) that prevent the event. The likelihood category is based on available data, operating experience, and/or engineering judgment. Hazardous events caused by human errors are generally assigned to the anticipated category in the absence of controls (assuming no procedures or training). Unless there is specific failure rate data or operating equipment history that justifies a different category, hazards events caused by equipment failure are also generally considered to be anticipated. For hazardous events involving a sequence of human errors or equipment failures that are unrelated (such as no common cause), a lower category, such as unlikely, is generally selected. If there is uncertainty in the likelihood category, a higher frequency category is conservatively assumed.

Table 2. Qualitative likelihood category.

Likelihood Category	Description	Frequency of Occurrence (per year)
Anticipated (A)	Events that have occurred or are expected to occur during the lifetime of the facility (frequency between once in 10 and once in 100 years)	10^{-2} to 10^{-1}
Unlikely (U)	Events that may occur but are not anticipated in the lifetime of the facility (frequency between once in 100 and once in 10,000 years)	10^{-4} to 10^{-2}
Extremely unlikely (EU)	Events that, while possible, will probably not occur in the lifetime of the facility (frequency between once in 10,000 and once in a million years)	10^{-6} to 10^{-4}
Beyond extremely unlikely (BEU)	Events that are considered too improbable to warrant further consideration (frequency less than once in a million years)	$<10^{-6}$

Potential unmitigated consequences to the environment are qualitatively estimated using the guidelines in Table 3. A qualitative estimate of the potential unmitigated consequences to the off-site public, collocated workers, and facility workers for each hazardous event is made using the guidelines in Table 4. The word unmitigated means that material quantity, form, location, dispersibility, and interaction with available energy sources are considered, but no credit is taken for safety features (e.g., ventilation system, fire suppression) that could prevent or mitigate a hazard. This does not, however, require ignoring passive design features that confine radioactive or hazardous material if their failure is not postulated by design (evaluation) basis events. If there is uncertainty in the consequence category, a more severe consequence category is conservatively assumed.

Table 3. Qualitative consequence category for potential environmental effects.

Consequence Category	Potential Environmental Effects
High (H)	Off-site contamination or major liquid release to the groundwater
Moderate (M)	INL site contamination
Low (L)	MFC site contamination outside the facility
Negligible (N)	No contamination outside the facility

Table 4. Qualitative consequence category.

Consequence Category	Off-Site Public	On-Site (Collocated) Workers	On-Site Facility Workers
High (H)	>25 rem or >ERPG-2*	>100 rem or >ERPG-3 or >Δ10 psi**	>100 rem or >ERPG-3 or >Δ10 psi**
Moderate (M)	5 rem to 25 rem Or ERPG-1 to ERPG-2	25 rem to 100 rem or ERPG-2 to ERPG-3	25 rem to 100 rem or ERPG-2 to ERPG-3
Low (L)	0.5 rem to 5 rem Or TLV-TWA to ERPG-1	5 rem to 25 rem or ERPG-1 to ERPG-2	5 rem to 25 rem or ERPG-1 to ERPG-2
Negligible (N)	<0.5 rem or <TLV-TWA	<5 rem or <ERPG-1	<5 rem or <ERPG-1

* Emergency Response Planning Guideline (ERPG)-Level 2

Threshold Limit Value-Time Weighted Average (TLV-TWA)

**Explosion overpressure is expressed as the differential pressure (Δ psi) of the shock wave.

It is assumed that consequences to the public occur at the INL Site boundary which is approximately 5,000 m from MFC. The collocated worker consequence applies to on-site workers located outside the facility at least 100 m from the release. The facility worker is located inside the facility, i.e., in the immediate vicinity of the release. Radiological consequence EGs are expressed as total effective dose equivalents (TEDEs).

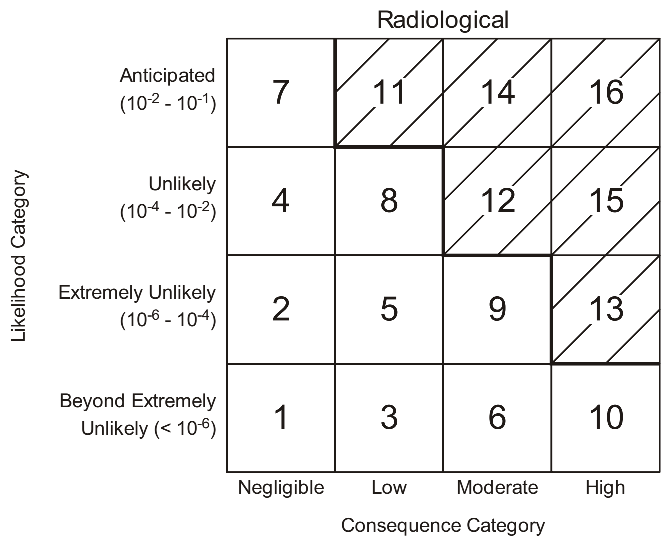
Consequences due to chemical hazards are evaluated in terms of emergency response planning guideline (ERPG) and threshold limit value time weighted average (TLV-TWA) values. TLV-TWA is the concentration for a normal eight-hour workday and a 40-hour workweek to which nearly all workers may be repeatedly exposed, day after day, without adverse effect. ERPG values are intended to provide estimates of concentration ranges where one reasonably might anticipate observing adverse effects (as described in the definitions of ERPG-1, ERPG-2, and ERPG-3) as a consequence of exposure to the specific substance.

- ERPG-1 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined, objectionable odor.

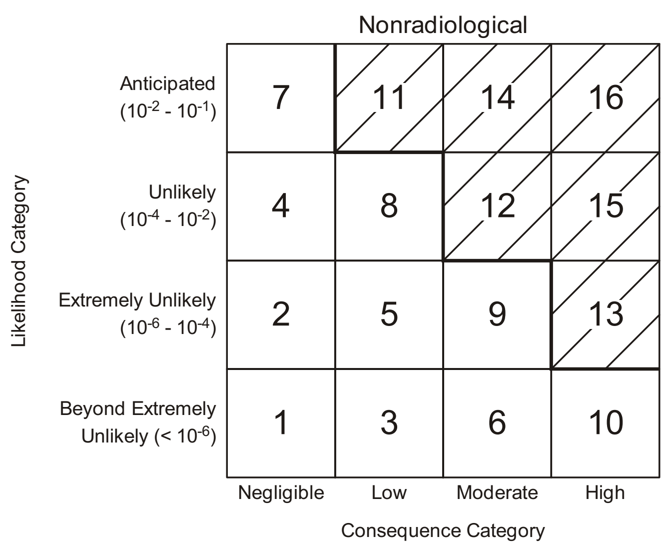
- ERPG-2 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair an individual's ability to take protective actions.

- ERPG-3 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiences or developing life threatening health effects.

Based on the likelihood and consequence categories, a risk bin number is assigned using the qualitative risk matrixes in Figures 12, 13, and 14. There is no risk bin for environmental effects. The risk bin numbers in the risk matrices indicate whether safety SSCs, TSRs, and/or safety analysis commitments (safety requirements) should be identified to manage the risk.



Consequence Category	Off-Site Public
High (H)	greater than 25 rem or greater than ERPG-2
Moderate (M)	5 rem to 25 rem or ERPG-1 to ERPG-2
Low (L)	0.5 rem to 5 rem or TLV-TWA to ERPG-1
Negligible (N)	less than 0.5 rem or less than TLV-TWA



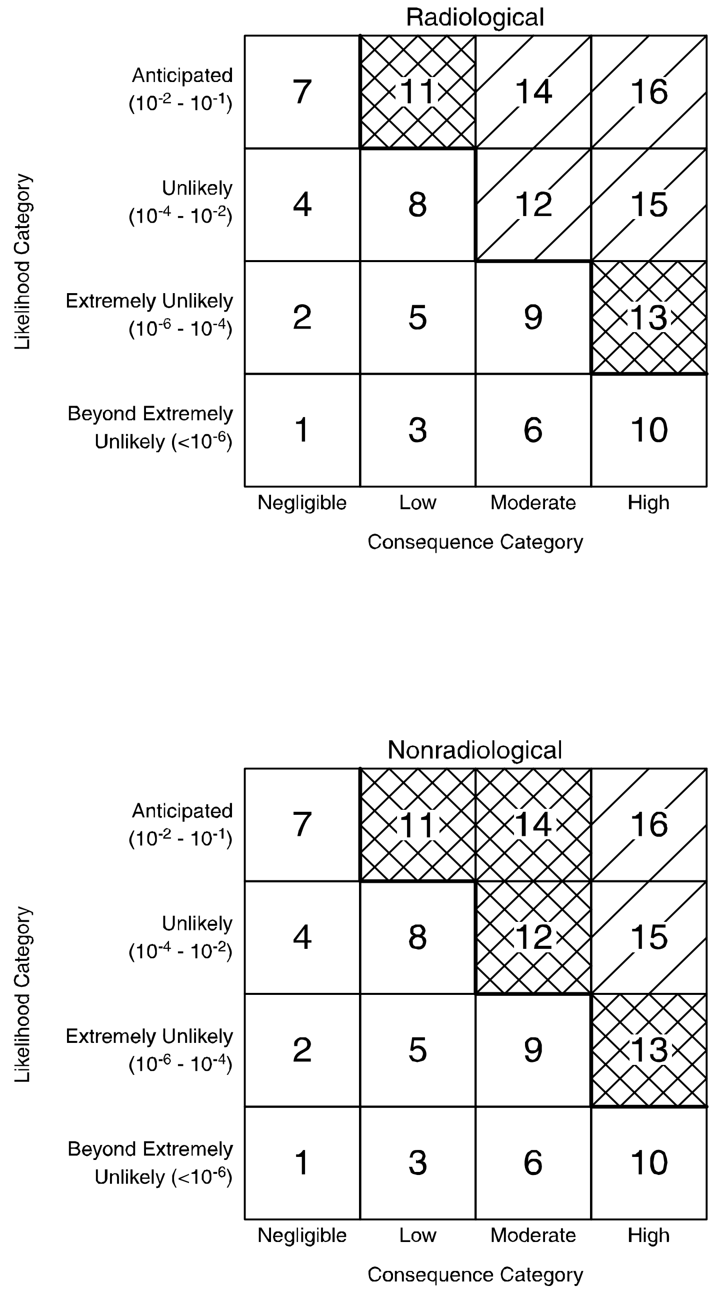
KEY

Safety-class SSCs and/or TSRs should be identified to manage off-site public risk; accident analysis may be needed.

Safety-class SSCs or TSRs are generally not required to manage off-site public risk.

02-GA51330-01

Figure 12. Qualitative risk matrix for the off-site public



Consequence Category	On-Site (Co-located) Workers
High (H)	greater than 100 rem or greater than ERPG-3 or greater than ▲10 psi
Moderate (M)	25 rem to 100 rem or ERPG-2 to ERPG-3
Low (L)	5 rem to 25 rem or ERPG-1 to ERPG-2
Negligible (N)	less than 5 rem or less than ERPG-1

KEY

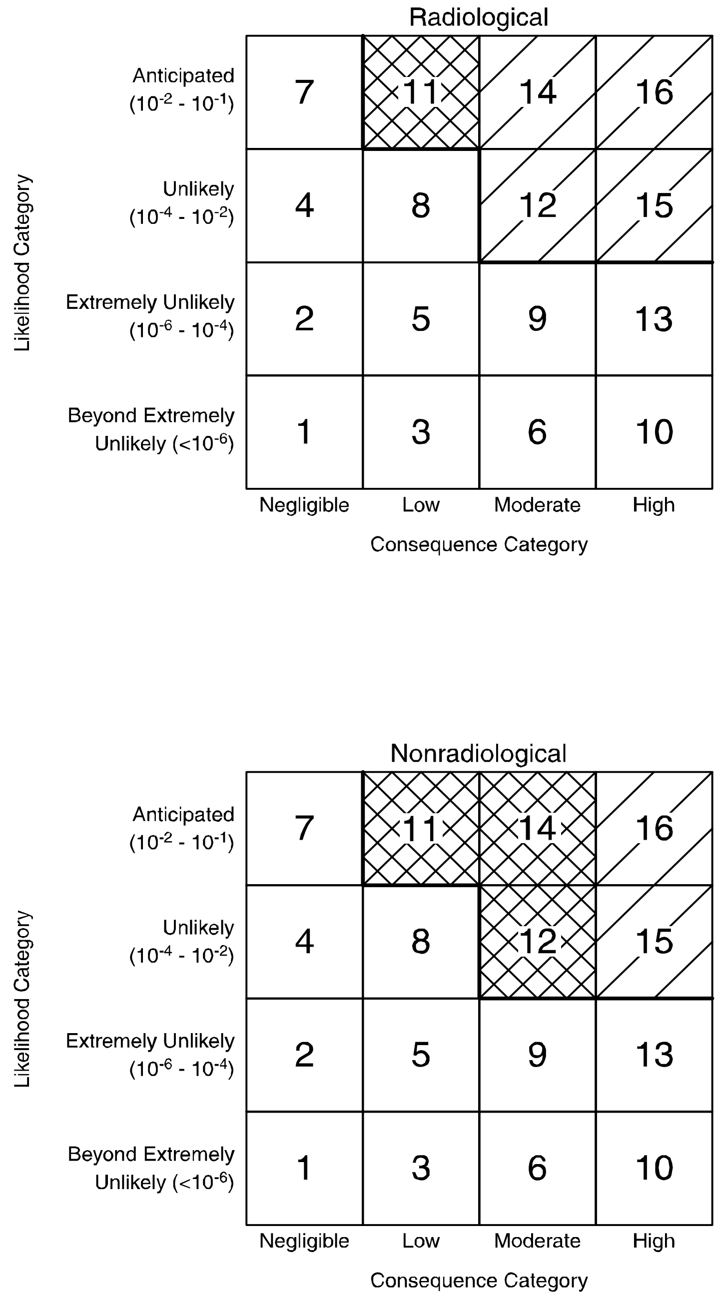
Safety-significant SSCs and/or TSRs should be identified to manage co-located worker risk; accident analysis may be needed.

Safety analysis commitments should be identified to manage co-located worker risk.

Safety SSCs, TSRs, or safety analysis commitments are generally not required to manage co-located worker risk.

06-GA50123-02

Figure 13. Qualitative risk matrix for on site collocated workers



Consequence Category	Facility Workers
High (H)	greater than 100 rem or greater than ERPG-3 or greater than ▲10 psi
Moderate (M)	25 rem to 100 rem or ERPG-2 to ERPG-3
Low (L)	5 rem to 25 rem or ERPG-1 to ERPG-2
Negligible (N)	less than 5 rem or less than ERPG-1

KEY

Safety-significant SSCs and/or TSRs should be identified to manage facility worker risk.

Safety analysis commitments should be identified to manage facility worker risk.

Safety SSCs, TSRs, or safety analysis commitments are generally not required to manage facility worker risk.

06-GA50123-03

Figure 14. Qualitative risk matrix for facility workers

Natural phenomena hazards (NPH) that could impact NRAD, other than the evaluation basis earthquake, were determined to be beyond extremely unlikely based on the location and climate history of the facility. Relative to NPH, volcanic activity has been dismissed as an applicable hazard for the NRAD facility. There are no design criteria for volcanic activity and the potential for future volcanic eruptions is considered extremely unlikely with a recurrence interval of 51,000 years.

Hazards associated with the refueling of the NRAD reactor core include: fissionable material inventory, hazardous material inventory, and standard industrial hazards. The facility hazards identified which could result in an uncontrolled release of radioactive materials or pose direct radiation exposure hazards are evaluated using industry standard methodology and provide the results of the hazard evaluation. The following are accident types and scenarios postulated that potentially could lead to a release of hazardous material:

1. Hazards during reactor operations

2. Hazards during the fuel conversion process

3. Nuclear criticality events

4. Natural phenomena hazards (earthquake)

5. External events.

Discussions of the accidents listed above provided an evaluation of the hazards or hazardous events and derived case-specific conclusions, including planned design and operational safety improvements, defense-in-depth, worker safety, and environmental protection. An accident selection summary was performed which identifies potential hazardous events for further quantitative analysis.

Since the NRAD reactor type has been widely used in the U.S. under NRC license, the types of hazards associated with its operation have been studied in depth and are well defined. Process hazard analyses (PHAs) have established the activities and consequences for the range of off-normal events that have been postulated for reactor operations. The results of the hazard analyses were documented with frequencies and consequences revised to reflect reactor operations with LEU fuel.

Off-normal events associated with TRIGA reactors and NRAD specifically are summarized in the following bullets:

- Abnormal reactivity insertion events

- Loss of coolant events

- Loss of heat removal system events

- Fuel cladding failure events

- In-tank and out-of-tank experiment events
- Radiography systems events
- NGEN event with tritium release.

The DSA Addendum for the LEU fuel summarizes the initiating events, their expected frequency of occurrence, and their expected consequences. For initiating events that do not result in the release of radioactive materials from the fuel elements, only a qualitative evaluation of the event was performed to show this to be the case. These accident and consequence discussions were based on the accident analysis provided by General Atomics.[10] Events leading to the release of radioactive material from the fuel elements were analyzed to the point required to conclude whether a particular event is the limiting-case event for each category.

Dose Calculations

The airborne source term (ST) was estimated by considering the MAR (curies or grams), damage ratio (DR), airborne release fraction (ARF), respirable fraction (RF), and leak path factor (LPF), as applicable. These factors and the values of the factors are explained and presented in DOE-STD-1027-92[6] and in DOE-HDBK-3010-94.[13] The five-factor formula from DOE-HDBK-3010-94 for selecting the ST is:

$$ST = MAR \times DR \times ARF \times RF \times LPF$$

Where:

ST = Source term

MAR = Material at risk

DR = Damage ratio

ARF = Airborne release fraction

RF = Respirable fraction

LPF = Leak path factor.

The committed effective dose (CED) is estimated from:

$$CED = ST \times \chi/Q \times BR \times DCF$$

where:

CED = Committed effective dose, rem

ST = Airborne source term, Ci

χ/Q = Dispersion coefficient, sec/m^3

BR = Breathing rate, 3.5×10^{-4} m^3/sec assumed in DOE-STD-1027-92

DCF = Dose conversion factor, rem/Ci

Dose consequences were estimated based on DCFs from International Commission on Radiation Protection Publication 68 (ICRP-68)[14] for the facility and collocated worker, and from ICRP-72[15] for the off-site public. The ICRP-68/72 methodology results in a CED determination, whereas the methodology used to derive the EGs presented in Table 4 results in a TEDE determination. If pathways other than internal are added to the CED, then the result is the total effective dose (TED). Although it is not appropriate to apply the committed effective dose equivalent (CEDE) and TEDE terminology when CED and TED results are actually calculated, the results are similar enough to allow the TED to be compared to the TEDE guidelines. Because of the dominance of the inhalation dose, the CED results are roughly equivalent to the TED determination.

Bounding events were analyzed and dose consequences calculated as described above and found that doses were well below evaluation guidelines for the public and adjacent workers. Furthermore, the maximum dose to a facility worker due to radioactive material release is negligible. The dose due to direct radiation exposure can be high in the immediate vicinity of a loaded cask drop during fuel movement. However if evacuation times are considered, the consequences and dose rates decrease rapidly from a drop accident. The maximum doses to the collocated worker and public are negligible; therefore, no controls are required to protect these receptors.

Safety Controls

Reactor fuel and reactor fuel-element cladding are inherent design features that limit radioactive material release during most accident events; these features are designated as safety-significant SSCs requiring TSR controls. The cask lifting and handling system is required to reduce the likelihood of drop events to extremely unlikely; therefore, this system is designated as a safety-significant SSC.

TSR controls for the fuel conversion project are related to fuel element cladding, the fuel and moderator design, and the cask lifting and handling system. Controls were identified as safety limits, limiting control settings, limiting condition for operations, and administrative controls.

The cask lifting and handling system was not previously designated as a safety-significant SSC and is therefore identified in this addendum as a new safety-significant SSC. The remaining safety-significant SSCs, namely the reactor protection system, reactor primary coolant system, reactor-room ventilation system, and reactor radiation monitoring system, have no changes due to the NRAD fuel conversion. Based on the hazard and accident analysis in the DSA addendum, these systems retain the designation of safety-significant because of their importance to “defense-in-depth” and worker safety. Their safety functions, safety descriptions, and functional requirements, system evaluations, and controls are provided in the existing NRAD DSA.

Accidental Criticality Considerations

Sufficient quantities of fissionable materials are associated with the NRAD reactor to necessitate the evaluation of the risk of an inadvertent criticality. The location of the material in the reactor room varies depending on the operations and activities on-going at the time. Within the NRAD reactor room, fissionable material is either handled in the reactor, in a fuel cask, or managed by hand out of storage.

With respect to determining criticality concerns, the criticality safety requirements applicable to NRAD are stated in the original DSA. The fundamental requirement is that criticality safety analysis be performed to document that a process will be subcritical under both normal and credible abnormal conditions. In addition, the purpose of the analysis is to identify the controlled parameters and the limits on these parameters. In the case of the fuel conversion in the NRAD reactor, not only are there sufficient quantities of fissionable materials present that criticality is possible, the criticality safety analysis has also defined specific critical configurations and situations. These form the basis for criticality accident scenarios evaluated. A criticality safety analysis was documented in a criticality safety evaluation (CSE)[16] performed in accordance with procedures that implement the INL Criticality Safety Program.

The CSE for the fuel conversion in the NRAD reactor addresses the fuel handling configurations for normal and abnormal conditions and provides the technical basis for determining the controlled parameters and applicable limits for fissionable material handling in the NRAD reactor. The scope of the CSE is as follows:

- Flooded NRAD fuel transfer cask
- Out-of-storage fuel clusters or elements
- NRAD cask-to-core collision or cask impact on fuel out of approved storage (including fuel in the tank pool storage racks).

A combination of a single engineered control and five administrative controls were established to ensure that an accidental criticality would not occur:

- The design of the NRAD cask lifting and handling system (monorail system, ceiling anchoring, and chain hoists) prevents cask drop and impact to the core or fuel out of storage (including fuel in the pool storage racks).
- No more than 13 fuel elements and a total of 150 g of U-235 equivalent mass in a reactor tank experiment are allowed out of approved storage in the NRAD reactor pool at any given time.
- Monorail system stops, guide rails, and guide rail extensions shall be in place/used during the handling of the NRAD fuel transfer cask in the reactor pool.
- No more than six NRAD reactor fuel clusters may be present in the NRAD reactor core unless at least one control rod is fully inserted in the core.

- No more than 12 NRAD reactor fuel clusters may be present in the NRAD reactor core unless at least two control rods are fully inserted in the core.

- No more than 16 NRAD reactor fuel clusters may be present in the NRAD reactor core unless at least three control rods are fully inserted in the core.

Through the implementation of these controls, an adequate margin of safety is achieved against the occurrence of accidental criticality.

4. Conclusion

The safety analysis for operation of the NRAD LEU core together with the core conversion project fuel handling, as documented in the addendum to the NRAD SAR, demonstrates that the public and worker health and safety and the environment are adequately protected. Protection is provided by the design features and inherent reactor core design and by administrative controls that govern NRAD reactor operation and procedures for fuel handling during the core conversion.

5. References

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6. DOE-STD-1027-92, "Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports," December 1992, (including Change 1, September 1997) U.S. Department of Energy.

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