INL/EXT-15-35940

# Scoping Study Investigating PWR Instrumentation during a Severe Accident Scenario

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# ABSTRACT

The accidents at the Three Mile Island Unit 2 (TMI-2) and Fukushima Daiichi Units 1, 2, and 3 nuclear power plants demonstrate the critical importance of accurate, relevant, and timely information on the status of reactor systems during a severe accident. These events also highlight the critical importance of understanding and focusing on the key elements of system status information in an environment where operators may be overwhelmed with superfluous and sometimes conflicting data. While progress in these areas has been made since TMI-2, the events at Fukushima suggest there may still be a potential need to ensure critical plant information is available to plant operators. Recognizing the significant technical and economic challenges associated with plant modifications, it is important to focus on instrumentation that can efficiently address these critical information needs.

As part of a program initiated by the Department of Energy, Office of Nuclear Energy, a scoping effort was initiated to assess critical information needs identified for severe accident management and mitigation in commercial light water reactors, to quantify the environment instruments monitoring this data would have to survive, and to identify gaps where predicted environments exceed conditions for instrumentation Environmental Qualification (EQ). Results from the Pressurized Water Reactor (PWR) scoping evaluation are documented in this report. The PWR evaluations were limited in this scoping evaluation to quantifying the environmental conditions for an unmitigated Short-Term Station BlackOut (STSBO) sequence in one unit at the Surry nuclear power station. Quantification was based on results obtained using the MELCOR models developed for the State of the Art Consequence Assessment project sponsored by the US Nuclear Regulatory Commission. Critical instrumentation considered in this scoping evaluation included sensors proposed by the PWR Owners Group (PWROG) in new Severe Accident Management Guidelines (SAMGs) supplemented with alternate generic PWR instrumentation that should be available at the Surry plant. Equipment locations and EQ values were estimated based on input from prior Surry plant evaluations and generic PWR plant information.

There are limitations associated with the information available for this scoping evaluation. The current study was limited to only one sequence, and plant specific instrumentation information was not available. However, results indicate that some instrumentation identified to provide critical information would be exposed to conditions that significantly exceed EQ values for extended time periods in the low frequency STSBO sequence evaluated. It is recognized that the estimated core damage frequency of this STSBO sequence would be considerably lower at some plants if evaluations considered new accident mitigation measures being implemented by industry, including measures to assure survivability of key instrumentation for an extended loss of alternating current power event. Furthermore, it is not clear that degradation of instrumentation systems exposed to conditions that exceed their EQ values would preclude the success of new SAMGs being proposed by industry. The use of alternate methods, such as alternate sensor information and 'trending' of degraded instrumentation, may be able to address critical information needs for implementing actions to mitigate such accidents. Nevertheless, because of uncertainties in instrumentation response when exposed to conditions beyond EQ ranges and challenges associated with different sequences that may present unique challenges to sensor performance, it is recommended that additional evaluations be completed to provide confidence that operators have access to accurate, relevant, and timely information on the status of reactor systems for a broad range of challenges associated with risk important severe accident sequences.

# ACKNOWLEDGEMENTS

Successful preparation of this report required input and support from several individuals and organizations. Financial support for the authors and researchers from Sandia National Laboratory (SNL) was provided by the Reactor Safety Technologies Pathway leader in the US Department of Energy Office of Nuclear Energy Light Water Reactor Sustainability program. In addition, there was substantial input by various industry organizations that provided technical experts to review a draft of this document. These organizations included Dominion, the Electric Power Research Institute (EPRI), the Pressurized Water Reactors Owners Group (PWROG), and the US Nuclear Regulatory Commission (US NRC). In particular, the authors would like to acknowledge the assistance received from William Webster, Dominion Virginia Power; Michael Weiner, Duke Energy Corporation; Joseph Naser, EPRI; Stephen Pierson, Exelon Nuclear; Kyle Ross, SNL; and Russell Sydnor and Pong Chung, US NRC. Their efforts are greatly appreciated.

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*Prepared for the*

*U.S. Department of Energy*

*Office of Nuclear Energy, Science, and Technology*

*Under DOE Idaho Operations Office*

*Contract DE-AC07-05ID14517*

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# ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| AC | Alternating Current |
| ANS | American Nuclear Society |
| BDB | Beyond Design Basis |
| BWR | Boiling Water Reactor |
| BWROG | Boiling Water Reactor Owners Group |
| CA | Calculational Aid |
| CDF | Core Damage Frequency |
| CETCs | Core Exit Thermocouples |
| CHLA | Candidate High-level Actions |
| CNWG | Civil Nuclear Working Group |
| CV | Control Volume |
| DBA | Design Basis Accident |
| DC | Direct Current |
| DEC | Design Extension Condition |
| DOE-NE | Department of Energy Office of Nuclear Energy |
| ECAs | Emergency Contingency Actions |
| ECCS | Emergency Core Cooling Systems |
| ECST | Emergency Condensate Storage Tank |
| EDMGs | Extreme Damage Mitigation Guidelines |
| EOPs | Emergency Operating Procedures |
| EPRI | Electric Power Research Institute |
| EQ | Environmental Qualification |
| FLEX | Accident response strategy developed by U.S. nuclear industry (see Section 2.4). |
| FR | Function Restoration |
| FSG | FLEX Support Guidelines |
| HHSI | High Head Safety Injection |
| IAEA | International Atomic Energy Agency |
| I&C | Instrumentation and Control |

| | |
|---|---|
| IC | Isolation Condenser |
| IEC | International Electrotechnical Commission |
| IEEE | Institute of Electrical and Electronics Engineers |
| INPO | Institute for Nuclear Power Operations |
| IPEs | Individual Plant Examinations |
| ISLOCA | Interfacing Systems Loss-of-Coolant Accident |
| KTA | Kerntechnischer Ausschuss |
| LHS | Lower Head Structure |
| LHSI | Low Head Safety-Injection |
| LOCA | Loss-of-Coolant Accident |
| LTSBO | Long Term Station BlackOut |
| LWR | Light Water Reactor |
| MACCS2 | MELCOR Accident Consequence Code System, Version 2 |
| MELCOR | Methods for Estimation of Leakages and Consequences Of Releases |
| MCCI | Molten Core Concrete Interaction |
| MSIV | Main Steam Isolation Valve |
| NRC | Nuclear Regulatory Commission |
| NSSS | Nuclear Steam Supply System |
| NTTF | Near Term Task Force |
| PORV | Pilot Operated Relief Valve |
| PRAs | Probabilistic Risk Assessments |
| PRT | Pressurizer Relief Tank |
| PWR | Pressurized Water Reactor |
| PWROG | Pressurized Water Reactor Owners Group |
| RCP | Reactor Coolant Pump |
| RCS | Reactor Coolant System |
| RPV | Reactor Pressure Vessel |
| RTD | Resistance Temperature Detector |
| RVLIS | Reactor Vessel Level Indication System |

| | |
|---|---|
| RWST | Refueling Water Storage Tank |
| SA | Severe Accident |
| SA-KEISOU | Severe Accident - Instrumentation & Monitoring Systems |
| SACRG | Severe Accident Control Room Guideline |
| SAMG | Severe Accident Management Guideline |
| SBO | Station BlackOut |
| SFP | Spent Fuel Pool |
| SG | Steam Generator |
| SGTR | Steam Generator Tube Rupture |
| SNL | Sandia National Laboratory |
| SOARCA | State of the Art Consequence Assessment |
| SPAR | Standardized Plant Analysis Risk |
| SRV | Safety Relief Valve |
| STSBO | Short-Term Station BlackOut |
| STUK | Finnish Centre for Radiation and Nuclear Safety |
| TAG | Technical Advisory Group |
| TDAFW | Turbine Driven Auxiliary FeedWater |
| TEPCO | Tokyo Electric Power Company |
| TF1 | TEPCO Fukushima Daiichi |
| TISGTR | Thermally- Induced Steam Generator Tube Rupture |
| TMI-2 | Three Mile Island Unit 2 |
| TSC | Technical Support Center |
| TSGs | Technical Support Guidelines |
| US | United States |
| YVL | Regulatory Guides on Nuclear Safety (Finland) |
| 10CFR | Title 10 of the Code of Federal Regulations |

# 1. INTRODUCTION

The accidents at the Three Mile Island Unit 2 (TMI-2) and Fukushima Daiichi Units 1, 2, and 3 nuclear power plants demonstrate the critical importance of accurate, relevant, and timely information on the status of reactor systems during a severe accident.[1 through 5] These events also highlight the critical importance of understanding and focusing on the key elements of system status information in an environment where operators may be overwhelmed with superfluous and sometimes conflicting data and yet have to make urgent decisions. While progress in these areas has been made since TMI-2, the accident at Fukushima suggests there may still be some potential for further improvement in critical plant instrumentation. In fact, several organizations, including the National Research Council of the National Academies of Science,[6] the Organization for Economic Cooperation and Development Nuclear Energy Agency,[7] and the International Atomic Energy Agency (IAEA),[8] have developed recommendations regarding the need for enhanced instrumentation during accidents with significant core damage.

Recognizing the significant technical and economic challenges associated with plant modifications, it is important to focus on a limited set of instrumentation that can efficiently address these critical needs. As part of a program initiated by the Department of Energy, Office of Nuclear Energy (DOE-NE), a scoping effort was initiated to:

- identify sensors capable of providing critical parameters needed for severe accident management and mitigation in commercial Light Water Reactors (LWRs),
- quantify the environmental conditions that instrumentation monitoring these parameters would have to survive, and
- identify gaps where predicted environments exceed instrumentation qualification levels.

The parameters and associated sensors will vary by reactor design, so these scoping studies evaluated one Boiling Water Reactor (BWR) and one Pressurized Water Reactor (PWR). Because of the availability of severe accident analysis information from recently completed calculations performed in support of the US Nuclear Regulatory Commission (NRC)-sponsored State of the Art Consequence Assessment (SOARCA) program,[9-11] the plants for these scoping studies are the Peach Bottom Atomic Power Station in Pennsylvania and the Surry Power Station in Virginia. Results from the Surry PWR scoping evaluation are documented in this report. Results for the Peach Bottom BWR are presented in Reference 12.

This report is organized into seven sections. Section 2 of this report provides background information related to the Surry reactor and containment design and a description of the accident progression analyses completed for the Surry SOARCA program. Other relevant sources of information, such as a review of prior instrumentation survivability studies and a summary of on-going efforts to enhance nuclear power plant instrumentation, are also found in Section 2. Section 3 describes the method used in this study to identify critical plant parameters required by operators to diagnose the plant status and to evaluate the effects of mitigating actions taken during an accident. These critical parameters, and the sensors for providing this information, are identified in this section with assumed instrumentation locations and Environmental Qualification (EQ) values. Section 4 summarizes the predicted environmental conditions that critical instrumentation systems are exposed to during one of the severe accident sequences evaluated in the SOARCA effort. Section 5 presents results from an instrumentation survivability assessment based on information in Sections 3 and 4. Results and insights from this effort are summarized in Section 6. References associated with this effort are listed in Section 7.

Additional information pertinent to this evaluation is provided in appendices of this report. Appendices A and B provide additional background information related to past efforts on this topic. Appendix C provides additional details related to the MELCOR calculations performed for this scoping evaluation. An initial draft of this document was provided to representatives from several organizations for review. Comments received from these organizations, along with the manner in which they were addressed, are provided in Appendix D.

# 2. BACKGROUND

This section provides background information related to the Surry reactor and containment design and the accident progression analyses completed for the Surry nuclear power plant in the SOARCA project. In addition, this section summarizes other relevant sources of information, such as prior instrumentation survivability studies and on-going efforts to assess the adequacy of nuclear power plant instrumentation during severe accidents. As discussed within this section, considerable effort was expended in prior studies to identify severe accident information needs and instrumentation that could address those needs. In addition, there are other US and international efforts underway to evaluate and, in some cases, enhance LWR instrumentation for severe accident conditions. The current scoping evaluation benefits from insights gained from these prior efforts and other on-going efforts on this topic.

## 2.1. Plant Description

The Surry nuclear power plant, which is located in Surry County, Virginia, adjacent to the James River (Figure 2-1), consists of two Westinghouse-designed PWRs, each with a rated thermal power of 2546 $MW_{th}$. Each reactor core consists of 157 15 x 15 assemblies with an active fuel height of 3.66 m. Each reactor coolant system (RCS) consists of three primary coolant loops. Each loop contains a U-tube steam generator, a reactor coolant pump (RCP), and associated piping. A single pressurizer is attached to hot leg piping in one of the three loops. Two pilot operated relief valves (PORVs) can relieve excess RCS pressure from the top of the pressurizer. One accumulator, containing borated water pressurized by a nitrogen cover gas, is attached to each cold leg. Table 2-1 summarizes important Surry design parameters. The RCS of each unit is housed within a subatmospheric containment building.

Surry Unit 1 began operation in 1972, and Surry Unit 2 began operation in 1973. In 2003, the US NRC extended the operating licenses for both reactors from 40 to 60 years.

**Figure 2-1.** Photo of the Surry plant.[Reference 13].

**Table 2-1.** Important Surry design parameters.[11]

| Parameter | Value, SI Units (British Units) |
|---|---|
| Rated Core Power, $MW_{th}$ | 2546 |
| Number of fuel assemblies in core | 157 |
| Rod array | 15 x 15 |
| Reactor Pressure Vessel (RPV) Inner Diameter, m (ft) | 2.0 (6.5) |
| RPV Height and Closure, m (ft) | 12.3 (40.4) |
| Pressurizer Relief Valves, kg/s ($lb_m$/hr) | 2 x 26.46 (2 x 210,000) |
| Pressurizer Safety Valves, kg/s ($lb_m$/hr) | 3 x 36.96 (3 x 293,300) |
| Pressurizer Relief Tank Liquid Volume, $m^3$ ($ft^3$) | 25.5 (900) |
| Pressurizer Relief Tank Design Pressure, bar (psig) | 6.89 (100) |
| Reactor Inlet / Outlet Temperature, ºC (ºF) | 282 /319 (540/606) |
| RCS Coolant Flow, kg/s ($lb_m$/hr) | 12,700 (100 x $10^6$) |
| Nominal RCS Pressure, MPa (psia) | 15.5 (2,250) |
| Secondary Pressure, MPa (psia) | 6.9 (1,000) |
| Secondary Side Water Mass, kg ($lb_m$) | 41,640 (91,800) |
| Secondary Side Volume, $m^3$ ($ft^3$) | 166 (5,868) |
| Emergency Condensate Storage Tank (ECST) Water Volume, L (gal) | 416,395[a] / 363,400[b] (110,000/96,000) |
| Refueling Water Storage Tank (RWST) Water Volume, L (gal) | 1,511,893 (399,400) |
| Turbine-driven Auxiliary FeedWater (TDAFW) pump, $m^3$/s (gpm) | 1 x 0.442@ 832 m (1 x700@2,730 ft) |
| Motor-driven Auxiliary Feedwater Pump, $m^3$/s (gpm) | 2 x 0.221@ 832 m (2 x 350@2,730 ft) |
| Containment Design Pressure, MPa (psia) | 0.31 (45) |
| Containment Volume, $m^3$ ($ft^3$) | 50,970 (1,800,000) |
| Containment Operating Pressure, MPa (psia) | 0.06 to 0.07 (9 to 10.3) |
| Containment Operating Temperature, ºC (ºF) | 24 to 52 (75 to 125) |
| Containment Failure Pressure, MPa (psia) | 0.7 (100) |
| Accumulator Water Volume, $m^3$ ($ft^3$) | 3 x 27.6 (3 X 975) |
| Accumulator Pressure, bar (psig) | 4.14 to 4.59 (600 to 665) |
| High Head Safety Injection (HHSI), $m^3$/s (gpm) | 3 x 0.0095 @ 1,768 m (3 x 150 @ 5,800 ft) |
| Low Head Safety Injection (LHSI), $m^3$/s (gpm) | 2 x 0.189 @ 69 m (2 x 3,000 @ 225 ft) |

a. Value listed first was assumed for Interfacing Systems Loss of Coolant Accident (ISLOCA) analysis in Reference 11.
b. Minimum amount required by technical specifications.

## 2.2. Surry SOARCA Evaluations

The NRC initiated the SOARCA project[9] to develop best estimates of the offsite radiological health consequences for a set of important severe reactor accidents for two representative nuclear power plants: the Peach Bottom BWR[10] and the Surry PWR.[11] The SOARCA project evaluated plant improvements and modeling changes not reflected in earlier NRC efforts, such as "Technical Guidance for Siting Criteria Development" (NUREG/CR-2239),[14] "Severe Accident Risks: An Assessment for Five U.S. Nuclear

Power Plants," (NUREG-1150),[15] and "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" (WASH-1400). [16]

Improvements and changes not reflected in earlier assessments include enhancements in systems, training and emergency procedures, offsite emergency response, and security-related measures, as well as plant modifications, such as power uprates and operating fuel at higher burnup. SOARCA's more realistic modeling also reduces conservatisms in earlier NRC estimates for offsite consequences. In addition to the improvements in understanding and in calculation capabilities that have resulted from over 25 years of research into severe accident phenomena, numerous changes have occurred in operating personnel training and in plant safety enhancements. These changes include:

- The transition from event-based to symptom-based Emergency Operating Procedures (EOPs) for PWR designs.
- The performance and maintenance of plant-specific probabilistic risk assessments (PRAs) that cover the spectrum of accident scenarios.
- The implementation of plant-specific, full-scope control room simulators to train operators.
- The use of industry-wide owner group severe accident guidance with plant-specific implementation of Severe Accident Management Guidelines (SAMGs).
- The use of additional safety enhancements to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire [i.e., Title 10, Section 50.54(hh) of the Code of Federal Regulations (10CFR50.54(hh))[17]].
- Consideration of improved understanding of severe accident phenomena, such as:
  - in-vessel steam explosions
  - dominant chemical forms for fission products
  - direct containment heating
  - hot leg creep rupture
  - steam generator tube rupture (SGTR)
  - RPV failure, and
  - molten core concrete interactions (MCCIs)

As summarized in Reference 9, SOARCA results indicate that all the modeled accident scenarios, even those cases where it is assumed that operators actions are unsuccessful, progress much slower and release much smaller amounts of radioactive material than calculated in earlier studies.

The SOARCA project sought to focus its resources on more important severe accident scenarios for Peach Bottom and Surry. The project narrowed its approach by using an accident sequence's possibility of damaging reactor fuel, e.g., core damage frequency (CDF), as a surrogate for risk. The SOARCA scenarios were selected from the results of existing PRAs. In general, the SOARCA project only analyzed scenarios with a CDF equal to or greater than $10^{-6}$ per reactor-year. However, the SOARCA project also analyzed scenarios leading to an early failure or bypass of the containment with a CDF equal to or greater than $10^{-7}$ per reactor-year, because these scenarios have a potential for higher consequences and risk. This approach allowed a more detailed analysis of accident consequences for the more likely, although still remote, accident scenarios. Using results from updated standardized plant analysis risk (SPAR) Version 3.31 models and available plant-specific external events information, two major groups of accident scenarios for analysis were identified for Surry plant evaluations.[11]

The first group includes short-term station blackout (STSBO) and long-term station blackout (LTSBO) events. Both types of station blackouts (SBOs) involve a loss of all alternating current (AC) power. The STSBO also involves the loss of turbine-driven systems through loss of direct current (DC) control power or loss of the condensate storage tank and therefore proceeds to core damage more rapidly (hence, the label, "short term"). The STSBO has a lower CDF because it requires a more severe initiating event and more extensive system failures. SBO scenarios can be initiated by external events such as a fire, flood, or earthquake. SOARCA assumed that an SBO is initiated by a seismic event because this is the most extreme case in terms of both the timing and amount of equipment that fails. SBO scenarios are commonly identified as important contributors in PRA because SBOs can lead to common cause failures of reactor safety systems and containment safety systems.

The second severe accident scenario group identified in Reference 11 is the containment bypass scenario. For Surry, two containment bypass scenarios were analyzed. The first bypass scenario is a variant of the STSBO scenario, involving a thermally-induced steam generator tube rupture (TISGTR). The second bypass scenario involves an ISLOCA caused by an unisolated rupture of low head safety injection piping outside containment. The ISLOCA scenario analyzed in SOARCA is a catastrophic failure of both of the inboard isolation check valve disks within the LHSI piping together with failure to refill the refueling water storage tank (RWST) or to cross-connect to the unaffected unit's RWST. The CDF for the ISLOCA, $3\times10^{-8}$ per reactor-year, falls below the SOARCA screening criterion for bypass events; but it was analyzed for completeness because NUREG-1150 identified ISLOCAs, in addition to SBOs and SGTRs, as principal contributors to mean early and latent cancer fatality risks. Because SOARCA evaluations deemed that it was likely that the operator actions during a SGTR would be successful, this scenario was dropped as a contributor. However, the effects of SGTRs were considered in the TISGTR STSBO scenario.

Using input from the Surry licensee, the SOARCA project developed models of plant systems, defined operator actions, and developed models for simulation of site-specific and scenario-specific emergency planning and response measures. In addition, the Surry licensee provided information on accident scenarios from their PRAs. A human reliability analysis, commonly included in PRAs to represent the reliability of actions by the operator and plant staff, was not performed for SOARCA. Instead tabletop exercises, plant walkdowns, simulator runs and other inputs from licensee staff were employed to model operator actions to mitigate selected scenarios and the ability of plant staff to implement mitigation measures.

SOARCA modeled mitigation measures, including those in EOPs, SAMGs, and 10 CFR 50.54(hh). The 10 CFR 50.54(hh) mitigation measures refer to additional equipment and strategies required by the NRC following the terrorist attacks of September 11, 2001. These measures were required to further improve each plant's capability to mitigate events involving a loss of large areas of the plant caused by fire and explosions. To assess the benefits of 10 CFR 50.54(hh) mitigation measures and to provide a basis for comparison to the past analyses of unmitigated severe accident scenarios, the SOARCA project also analyzed each scenario without 10 CFR 50.54 (hh) equipment and procedures. The analysis that credited successful implementation of the 10 CFR 50.54 (hh) equipment and procedures in addition to actions directed by the EOPs and SAMGs was referred to as the mitigated case. The analysis without 10 CFR 50.54(hh) equipment and procedures was referred to as the unmitigated case (e.g., SAMGs were not implemented in the unmitigated case).

The present work only considered results from the SOARCA evaluation of the unmitigated STSBO scenario. However, as noted above, SOARCA evaluations did consider other sequences, such as SGTR or ISLOCA bypass events or more slowly progressing events with hydrogen burns such as the LTSBO, that could present different challenges to plant instrumentation. SOARCA analyses were performed with two computer codes, the Methods for Estimation of Leakages and Consequences Of Releases (MELCOR), Version 1.8.6,[18] for accident progression and the MELCOR Accident Consequence Code System, Version 2 (MACCS2)[19] for offsite consequences. The present scoping evaluation only used results from the MELCOR analysis and were performed with MELCOR, Version 2.1.[20] The MELCOR model and results obtained from this model are discussed in Section 4. Section 4 also provides details related to assumptions for the STSBO accident sequence.

## 2.3. US NRC Guidance, Evaluations, and Future Actions

The need for better instrumentation was recognized after the TMI-2 event. As discussed in this section, the NRC funded several PWR-specific studies on this topic. However, recommendations from NRC-funded studies were dismissed because the perceived benefits did not appear to offset the anticipated costs. The Fukushima event has again emphasized the importance of having a critical set of reliable post-accident instrumentation. This section summarizes current NRC guidance, relevant past PWR evaluations, and proposed future actions of interest to this topic. Additional details related to NRC regulation and current NRC activities are found in Appendix A of this report.

### 2.3.1. Regulatory Criteria and Guidance

Current regulatory requirements and guidance for operating reactors do not specify that licensees perform a comprehensive evaluation of the instrumentation needed for severe accidents.[21] Accident monitoring equipment in the current fleet of operating reactors must meet qualification criteria based primarily on the reactor and containment response during conditions associated with design basis accidents (DBAs).[22] Initially, post-TMI-2 measures (NUREG-0660)[23] identified the need for licenses to provide instrumentation that provided operators access to timely information about critical parameters, such as water level in the reactor vessel, core temperature, containment hydrogen concentration, and containment radiation level for a range of plant conditions, including "an accident that includes core damage." Subsequent guidance (NUREG-0737)[24] clarified that such instrumentation ranges would be limited to environmental conditions associated with DBAs.

New LWR applicants must perform evaluations of instrumentation survivability during severe accident conditions. The need to consider an accident with core damage for new reactor applications was initially codified as a requirement in 10 CFR50.34(f)(2)(xix).[25] Currently, 10 CFR Part 52[26] applicants must complete analyses that provide assessments of severe accident equipment needs, predicted environments, and equipment survivability. For the instrumentation system to provide information necessary to support operators in responding to severe accident events, the instrumentation components must survive severe conditions and be provided with a functional supply of power.

Regulatory Guide 1.97[27,28] provides guidance for instrumentation needed to comply with regulatory requirements during and following an accident. Revision 3 of Regulatory Guide 1.97,[27] which contains a prescriptive list of the minimum number of variables to monitor in BWR and PWR plants with design and qualification criteria, remains in effect for licensees of operating reactors. Requirements in Regulatory Guide 1.97, Revision 3 (see Appendix A of this report for the list of PWR requirements) are for design-basis events rather than severe accident events. Revision 4 of Regulatory Guide 1.97,[28] which was issued for licensees of new reactor plants, states that licensees should provide instrumentation with expanded ranges capable of surviving the accident environment (with a source term that considers a damaged core) in which it is located for the length of time its function is required.

### 2.3.2. Prior NRC Evaluations

During the 1990s, the NRC funded a program to evaluate instrumentation survivability. As discussed in Appendix A, a method was developed to identify (a) information needed to understand the status of the plant during a broad range of severe accident conditions, (b) the existing plant measurements which could be used to directly or indirectly supply these information needs, (c) the potential limitations on the capability of these measurements to function properly, and (d) the conditions in which information from the measurement systems could mislead plant personnel. As shown in Figure 2-2, steps were established to identify the severe accidents of interest, the information needed by the operator, the capabilities of the instrumentation, and the severe accident conditions imposed on the sensors. Then, an assessment of instrumentation survivability was completed as a final step. Survivability assessments considered the entire instrumentation system, including transducers, cabling, electronics, and other instrumentation components.

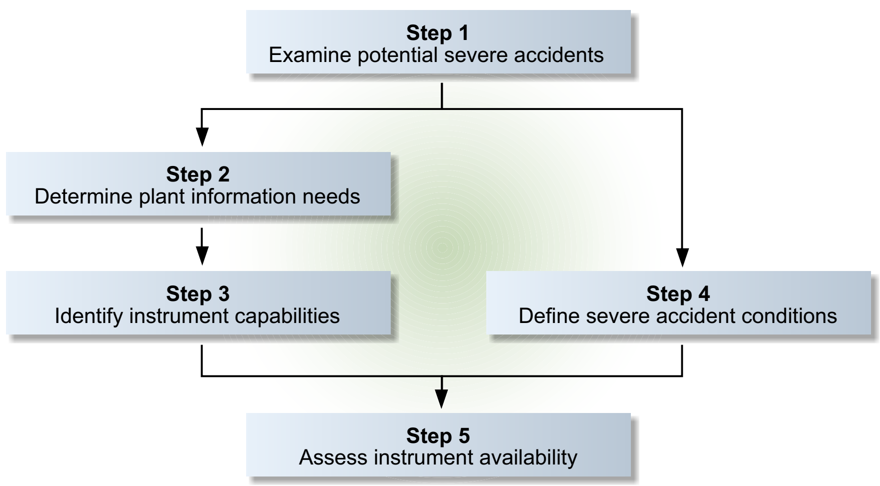



**Figure 2-2.** Reference 34 methodology to assess instrumentation survivability.

The method was applied to representative PWRs and BWRs for risk-important accident sequences identified in NUREG-1150[15] using analysis and information available in the early 1990s. PWR evaluations were completed for the Surry and Zion plants.[29 through 32] These evaluations were completed using analysis results obtained with computer codes, such as MARCH2 and MERGE. At the time that the evaluations were completed, these earlier codes did not consider phenomena, such as natural circulation, that can significantly impact event timing and energy distribution from the core into the upper plenum and regions outside the reactor vessel. Also, these earlier studies did not include activities to determine a critical set of parameters and instrumentation required for accident management.

As an accident progresses, different plant safety functions are challenged; and harsh environmental conditions will develop in different locations within the RCS, containment, and in some sequences, the auxiliary and turbine buildings. Hence, evaluations[33 through 35] were completed for five different phases of an accident: (1) initiation; (2) core uncovery; (3) fuel melting and relocation; (4) relocating core accumulation on the vessel lower head and vessel failure; and (5) ex-vessel interactions in the containment. The studies considered selected instrumentation enhancements, such as using existing instrumentation for different applications, extending the operating range of selected sensors, deploying new instrumentation systems, and the use of analysis aids to guide decision-makers during a severe accident.

Instrument survivability evaluations were primarily based on the pressure, temperature, and radiation EQ ranges, and the location and source of backup power for each instrument. Pressure and temperature conditions were emphasized because these conditions appeared to have the potential to strongly influence instrument performance, particularly in the early stages of the accident. Exposure of instrument system components to radiation was also found to have the potential to impact availability. In particular, components made from synthetic organic materials were particularly susceptible. However, in many events, this effect was found to be much less important than temperature and pressure effects because it was only influential in the very long term (days). Exceptions are SGTRs and ISLOCAs, where the availability of information from instruments used to monitor secondary side coolant radioactivity levels could degrade because of radiation levels and temperatures that were well beyond instrumentation system EQ ranges. Furthermore, the ability of plant personnel to obtain and analyze samples of reactor coolant, containment air, containment sump water, and other process fluids would be impeded. In these studies, it was concluded that relative humidity would not affect availability, because instruments were generally qualified for operation in an environment with 100% humidity.

These NRC-funded studies[33 through 35] assumed that instrument performance was degraded if pressure and temperature environments exceed instrumentation EQ values. However, the studies recognize that the assumption of degraded instrument performance for all conditions exceeding the EQ values may be conservative, particularly if the environmental conditions exceed the values by only small amounts or for short periods of time. Furthermore, it is possible that some components of the instrument systems are sufficiently protected to withstand the temperature pulse expected during some of these events. Limited testing[36,37] indicates that typical nuclear instrumentation could survive a single hydrogen burn, but failures were observed in transducers and cabling[34] when exposed to multiple hydrogen burns. However, in general, basic instrument system performance is not well known when EQ conditions are exceeded. There is a need to consider specific conditions expected during accident scenarios, failures of instrumentation system components such as cabling and splicing, and plant-specific locations of instrumentation components.

### 2.3.3. On-Going Regulatory Efforts

Section 4.2 of the Near Term Task Force (NTTF) report[1] discusses the significant challenges faced by operators in understanding the condition of the Fukushima Daiichi reactors, containments, and Spent Fuel Pools (SFPs) because existing design-basis instrumentation was either lacking electrical power or providing erroneous readings. A post-Fukushima action item (Identifier SECY-12-0025, Enclosure 2)[38] was established to address this concern and to evaluate the regulatory basis for requiring reactor and containment instrumentation to be enhanced to withstand severe accident conditions. This activity was prioritized as Tier 3 because it requires further staff study and depends on the outcome of other lessons-learned activities. For example, there are opportunities for licensees to enhance PWR instrumentation as they address several post-Fukushima actions, including NTTF recommendations and in orders issued by the staff, such as EA-12-049, "Requirements For Mitigation Strategies For Beyond-design-basis External Events,"[39] and EA-12-051, "Reliable Spent Fuel Pool Instrumentation."[42] As part of their efforts, the NRC staff is reviewing information from previous and ongoing research efforts for severe accident management analysis, and is monitoring results of DOE-NE, industry, and international research activities and reviewing guidance being developed by domestic and international organizations (see Section 2.5). Reference 21 indicates that the NRC is considering several options, such as dedicated independent power sources for critical plant instrumentation for time periods before diverse and flexible coping capability or "FLEX"* equipment could be installed, analyses and environmental testing that demonstrate that critical instrumentation will survive 'well into the accident progression', and operating procedures that incorporate insights from such analyses and testing. Reference 42 indicates the NRC will make a regulatory determination on this topic by December 2015. Appendix A provides additional details about current NRC efforts.

## 2.4. Industry Evaluations and Future Actions

The significant effort that followed the accident at TMI-2 led the U.S. nuclear industry to develop SAMGs for the U.S. nuclear fleet. This section reviews prior and on-going industry efforts related to severe accident guidance development and instrumentation survivability evaluations.

### 2.4.1. Generic SAMGs and CHLAs

Guidance to aid operating crews in responding to a severe core damage accident was first developed as a response to the 1979 accident at TMI-2. This guidance encompasses those actions that should be considered to arrest core damage accident progression or to limit the extent of resulting fission product releases. Early guidance was developed by Electric Power Research Institute (EPRI) in a logical manner, starting with compiling the best information regarding severe accident phenomena available at that time.[43] In turn, this information was used to identify general actions that could be taken to manage a severe accident; these general actions are referred to as candidate high-level actions (CHLAs). The CHLAs formed the basis of generic guidance developed by the various owners groups representing the Nuclear Steam Supply System (NSSS) vendors. This generic guidance is ultimately used to assemble the plant-specific guidance for each operating nuclear power plant. Reference 44 provides updated CHLAs to account for the initial lessons learned from the Fukushima Daiichi accidents that occurred in March 2011. To provide a technical basis for plant-specific guidance development, Reference 44 also identified various damage conditions that may occur during different phases of a severe accident and methods for detecting such conditions with plant instrumentation. Additional information about each of these conditions and when they could be expected during a severe accident is found in Appendix B.

There are several levels of guidance for the operating staff of a commercial nuclear power station (Figure 2-3). The first level, termed operating procedures, focuses on plant operation during the time that plant parameters are within an acceptable range. The second level, termed abnormal operating procedures, focuses on restoring the function of systems that could impact overall plant operating margins. The third level, termed EOPs, is aimed at bringing the plant to a safe, stable state following a reactor trip or safety injection signal. These procedures represent the initial phase of accident management and have been formulated around the essential safety functions such as reactivity control, adequate core cooling, etc. EOPs have been developed for each NSSS design and have continued to evolve as additional information becomes available. Last, SAMGs (with CHLAs) and other guidance and calculational aides (see Section 3.2) are used to address RCS and containment conditions that develop following core damage. Such guid-

* FLEX is a strategy developed by the U.S. nuclear industry in response to the accidents that occurred at Fukushima Daiichi. It includes the use of portable equipment, such as pumps and generators, that are kept on site or delivered from one of two regional FLEX facilities and that are used in a "flexible" way to respond to various potential challenges to core cooling and power restoration.

ance becomes necessary when the accident has progressed beyond the plant state for which detailed EOPs have been developed. As discussed in Section 2.4.3, efforts are underway to develop updated SAMGs for PWRs and BWRs.

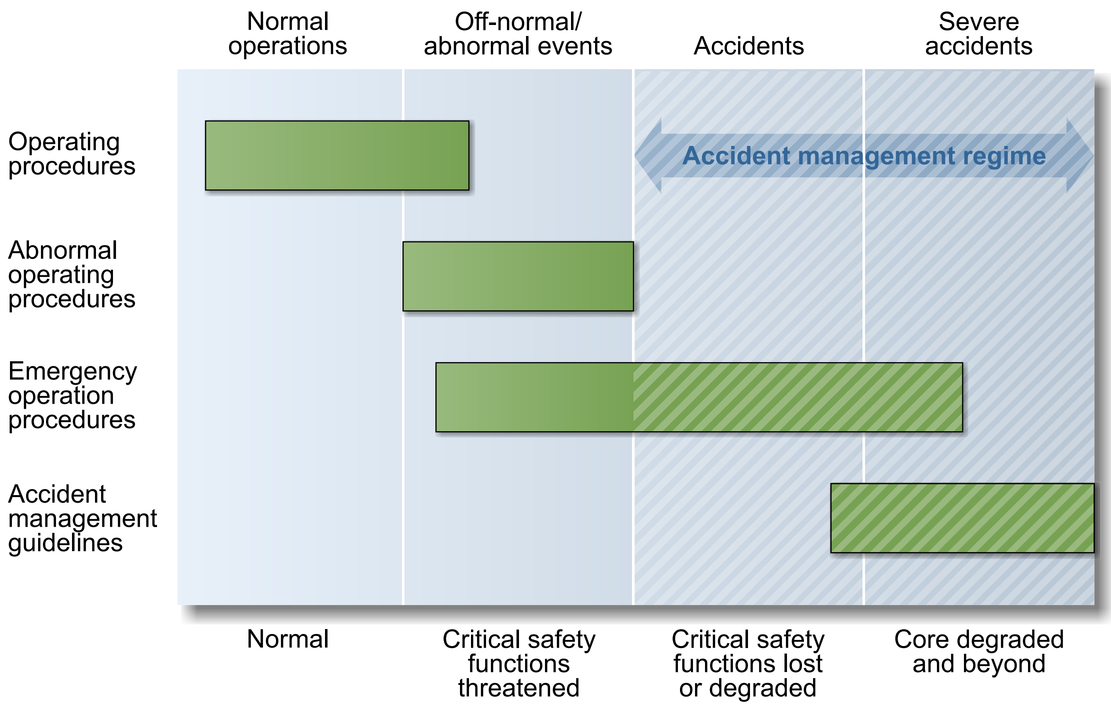



**Figure 2-3.** Typical role of procedures and accident management guidelines.[44]

### 2.4.2. Prior Instrumentation Evaluation Efforts

Reference 45 describes results from a systematic process followed by EPRI to evaluate what types of information might be expected from various types of installed instrumentation during severe accident conditions. Fourteen types of generic instrumentation loops were identified that could measure parameters such as RCS pressure and temperature, containment pressure, temperature, radiation levels, and combustible gas concentration. The study evaluated available information related to instrumentation performance beyond their operating envelope and identified operational aides that could be used by operators to gain confidence in sensor data during a severe accident, such as redundant information from different types of sensors, indirect information from other sensors, portable instruments to measure parameter or related parameters, and methods to evaluate circuit health (e.g., circuit resistance and continuity measurements). Representative checklists are provided to assist owners/operators in developing plant-specific approaches for implementing operational aides. In addition, as discussed in Appendix B, tables are provided in Reference 45 that list ranges of interest for various types of parameters during different severe accident phases or conditions.

EPRI also completed an instrumentation survivability assessment for two pilot plants (a 4-loop Westinghouse PWR and a Mark II BWR), similar to the studies completed by the US NRC (see Section 2.3.2). Results are documented in Reference 46. At a high level, the EPRI approach (see Figure 2-4) is similar to the NRC approach (see Figure 2-2). In both cases, the evaluations identify information needed by the operators to manage a severe accident, select the instrumentation capable of providing such information, and estimate the environmental conditions to which such instrumentation is exposed. Finally, in each case, an evaluation is completed to assess the adequacy of such instrumentation to provide required information.

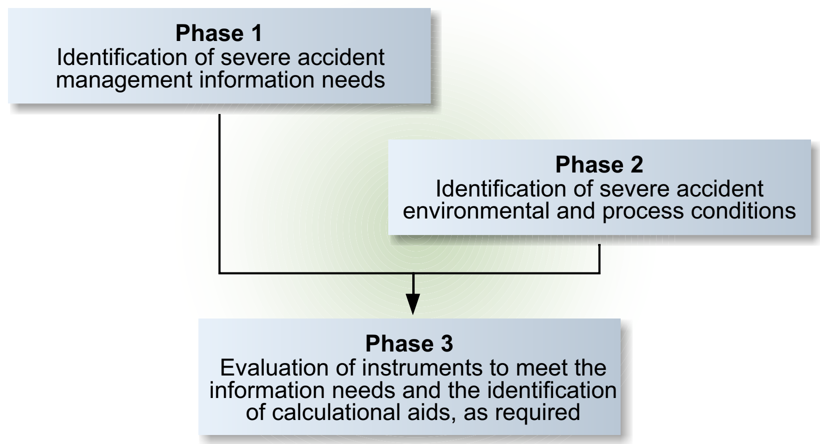



**Figure 2-4.** EPRI instrumentation adequacy evaluation approach.

However, there are significant differences between prior NRC and EPRI approaches. One of the most important differences is the EPRI study emphasis on identifying a *minimum* set of key information needs to support severe accident mitigation. As discussed in Section 2.3.2, the NRC approach considered general information needs based upon phenomenological understanding of severe accidents and possible instrumentation that could provide data to address these information needs. In contrast, the EPRI approach focused on identifying a *minimum* set of key information needs necessary to support severe accident management guidance implementation (e.g., EOPs, SAMGs, and Core/Containment State Assessment). Later NRC study references (e.g., Reference 35) acknowledge that the EPRI approach is a valid method for identifying instrumentation systems capable of providing the required information during severe accidents.

The EPRI study used plant-specific severe accident analysis results, plant-specific design information to identify instrumentation system component location, and equipment qualification data. Plant-specific severe accident analysis results were obtained from Modular Accident Analysis Code (MAAP)[47] computer code calculations performed in the early 1990s. Limitations in MAAP modeling detail required that conservative assumptions and, in some cases, stand-alone calculations were needed to determine the conditions of interest for instrumentation system components. The EPRI study focused on scenarios leading to more harsh consequences to determine the extent to which plant instrumentation may have to operate.

Similar to the NRC study, the EPRI method compared the instrumentation EQ values with conditions predicted to occur for risk-important accidents (see Figure 2-5). EPRI assessment results indicate that existing plant instrumentation can provide the information required during the various phases of severe accidents. Alternative methods were identified that would be available to either directly or indirectly measure the required parameters. Specifically, the study identified 12 information needs that are not satisfied by direct measurements in the PWR and BWR pilot plants. Of these 12, alternative methods were identified for all but two information needs (containment hydrogen concentrations and containment atmosphere temperatures). Of the remaining two, the containment hydrogen concentration can be monitored by post-accident systems that are designed to provide hydrogen concentration information during the early stages of severe accidents. These systems were expected to work up to the time of vessel failure in most cases. After this point, it was suggested that alternate methods could be used to obtain grab samples and that actions could be identified for cases where the containment hydrogen concentration is unknown.

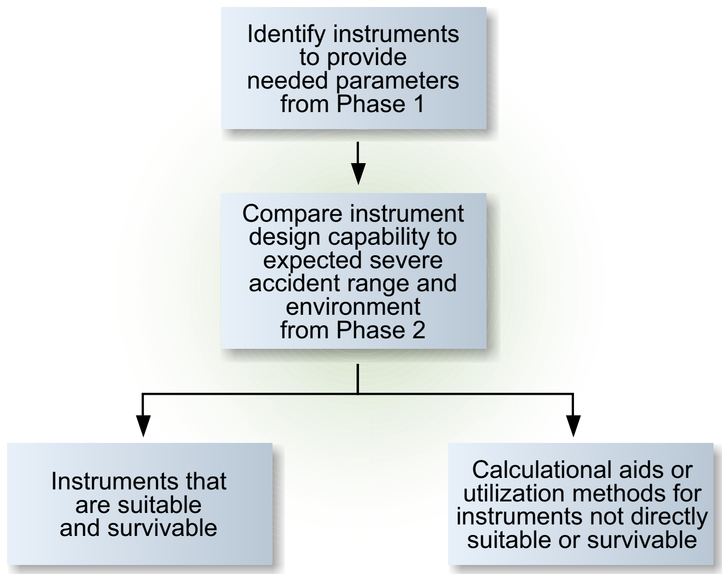



**Figure 2-5.** EPRI evaluation of instrumentation survivability and of required calculational aids.

Rather than identifying sensor enhancements, the EPRI study (see Figure 2-5) proposed development of operating aides for situations in which sensors were not predicted to survive. In addition, the EPRI study proposed (and applied) an approach for extending the methodology to two other PWR plants (one Combustion Engineering unit and one Babcock & Wilcox unit). Results from this proposed extension suggest that the method can be applied generically with few potential plant-specific differences.

### 2.4.3. Current Efforts

In response to NRC Orders EA-12-049[39] and EA-12-051,[40] the industry developed guidance for mitigation of certain beyond design basis accidents similar to the Fukushima accident (See References 48 through 50). This guidance is known as FLEX and includes both additional equipment to assure continued core, containment and spent fuel cooling during an extended loss of AC power as well as FLEX Support Guidelines (FSGs)* for the appropriate use of this equipment. The industry response also included development of guidance for assuring that reliable instrumentation indications were available for key instrumentation that would be used in decision making by the licensed plant operators under these beyond design basis conditions. This guidance includes:

* New FLEX Support Guidelines (FSGs) are a subset of the EOPs for use in certain Beyond Design Basis (BDB) conditions to provide alternate strategies for core, containment, and spent fuel cooling. FSG-7, "Loss of Vital Instrumentation or Control Power,"[51] which provides actions to establish alternate monitoring and control capabilities, has the objective to ensure that operators have access to accurate data for critical parameters.

- Implement spent fuel pool wide-range level instrumentation,
- Provide freeze protection for critical instrumentation,
- Strategies to circulate and cool air in containment compartments to prevent any adverse impact on critical instrumentation,
- Strategies to circulate air in key rooms in the auxiliary building to prevent any adverse impact on power supplies and/or critical instrumentation,
- A strategy to deploy portable generators and cables to directly reestablish power to the power supplies in select cabinets thereby re-powering the instrumentation loops, and
- A strategy to utilize handheld instruments to tap into the instrument loops locally to monitor essential parameters.

While these recent enhancements are directed toward the initial (e.g., "pre-core" damage) phases of an event, they also provide an enhanced instrument availability and an alternate means of obtaining key parameter values if the event progresses to a severe accident.

Both the PWR Owners Group (PWROG) and the BWR Owners Group (BWROG) are developing enhanced post-Fukushima generic SAMGs with Technical Support Guidelines (TSGs) on instrumentation behavior.[4 - 5] In these enhanced SAMGs, instrumentation indications are used to determine challenges to plant fission product boundaries, to identify and prioritize needed actions, and to determine whether implemented actions are successful. Correct interpretation of signals from instrumentation is fundamental to the successful diagnosis, control, and mitigation of a severe accident. Since severe accidents are beyond the design basis of the plant, conditions may be more extreme than ranges for which the instrumentation was designed or calibrated. Several key factors that will be considered in these owner group evaluations include:

- Instrumentation typically relied upon for a DBA may not be available (e.g., power supplies, isolation valves, etc.) during a severe accident,
- The instrumentation range may not be adequate during a severe accident,
- Use of instrumentation may challenge fission product boundaries (e.g., hydrogen analyzer), and
- The magnitude of the environment (pressure, temperature, radiation, etc.), as well as the time at which elevated conditions are present, in comparison to the EQ basis may lead to erroneous readings.

In Reference 4, the PWROG recommends that instrumentation indications be validated by an independent means if possible. The PWROG further recommends that any instrument believed to provide useful information be considered, whether or not it is safety grade or qualified. The PWROG also observes that it is not generally known whether an instrument will fail or continue to function when conditions exceed design basis expectations and/or EQ ranges. Even if an instrument survives testing beyond its EQ values, the tests may not have been completed to the point of instrument failure. Therefore, EQ values do not provide a basis for conclusions on the failure point of an instrument.

The enhanced PWROG SAMGs include TSGs with guidance for determining the validity of the information being provided by the plant instrumentation. The TSGs support the diagnosis and selection of mitigation strategies as well as confirmation of the adequacy of mitigation actions after they are implemented. The instrumentation TSGs provide the SAMG user with additional information that can be used to determine the validity of the instrumentation indications. This guidance is knowledge-based and relies on comparing instrumentation indications with other key information including: alternate instrumentation for the same parameter, assessment of other related or linked parameters (such as pressure and temperature), other indications not directly provided by instrumentation, calculational aids, and expectations for trending of plant parameters based on the accident progression.* Guidance is to be provided for all key parameters needed for effective severe accident management using the new, enhanced PWROG SAMGs. Ultimately, Reference 4 indicates that plant-specific applications will be developed using enhanced generic SAMGs and TSGs.

Validation activities of enhanced PWROG SAMGs with the instrumentation TSGs are scheduled to occur during 2015 at a plant from each of the three PWR reactor vendors (Westinghouse, Combustion Engineering, and Babcock and Wilcox). These validation activities will be performed using simulated severe accident scenarios in a table-top mode. As part of these activities, the PWROG has proposed a list of critical parameters and instrumentation capable of providing data for these parameters during a severe accident. As discussed in Section 3.2, this list serves as a starting point for the scoping evaluation documented in this report.

BWROG activities to develop TSGs are currently focused on obtaining insights from detailed evaluations of available TEPCO instrumentation data from Daiichi Units 1, 2, and 3[5] and include an assessment of how differences between indicated and actual values may have influenced actions taken at Fukushima. Results are being used to develop principles for validating instrument indications received during an accident. These principles were demonstrated on validating RPV water level indications from Daiichi Units 1, 2, and 3, on identifying the presence of metal water reactions using alternate indications (no hydrogen monitors) for Units 2 and 3, and on conflicting indications of RPV pressure on Unit 1 and containment pressures from Unit 2. Results allow the BWROG to validate that the SAMGs revised to reflect lessons learned from Fukushima could be implemented and, with proper training, utilized with the limited information the operators had at Fukushima.

As part of their post-Fukushima activities, EPRI formed a Technical Advisory Group (TAG) to address Instrumentation and Control (I&C) for BDB events and severe accidents.[54,55] The purpose of the TAG, which consists of representatives from the Institute for Nuclear Power Operations (INPO), EPRI, PWROG, BWROG, NRC, and DOE-NE is to provide a collaborative and coordinated response in:

- Addressing the lessons learned from the events in Japan about the required durability and capabilities of I&C systems during severe accident events.
- Identifying the required parameters and ability of reactor and containment I&C systems to withstand severe accident conditions.
- Performing research to determine if the availability of the I&C can be improved so that plant data are not lost during severe accidents.

Overall, the ultimate objectives of the TAG are to:

- Improve the knowledge and understanding of I&C's role in monitoring, responding, and mitigating severe accidents.
- Provide research that results in identification of equipment and strategies that foster the capability and survivability of critical I&C during severe accidents.

* Reference 4 emphasizes the use of trending in proposed new SAMGs, noting that differences of 10% or more are acceptable if trending information data for parameters are available. However, in some cases, additional experimental data are needed to support this assertion.

The TAG role is primarily one of communication, meaning that it facilitates exchange of information, rather than directing work or assignments. Each of the listed stakeholders has their own established role and initiatives in response to this topic area. The approach of the TAG is to ensure that each organization knows what others are working on; facilitating collaborations and communication.

## 2.5. Other Relevant Information

There are several other U.S. and international activities related to instrumentation survivability during severe accidents that are relevant to this scoping evaluation. This section summarizes these activities.

### 2.5.1. DOE Reactor Safety Technology Activities

The Reactor Safety Technology Research and Development (R&D) effort was established following the Fukushima Daiichi accident. On October 1, 2014, this effort became a pathway, which is referred to as the Reactor Safety Technologies (RST) Pathway within the LWR Sustainability Program.[56] This pathway seeks to improve the basic understanding of BDB events and reduce the associated uncertainty in severe accident progression, associated phenomenology, and key outcome. The RST pathway accomplishes these goals using existing analytical codes and information that has been obtained (or will be obtained) from severe accidents, in particular the Fukushima Daiichi events. The insights gained from these models and analyses and the forensics information are used with the advice and collaboration of the U.S. nuclear industry to better inform nuclear power plant owner/operators in developing mitigating strategies for accidents that may go beyond the design basis and to aid in the formulation of SAMGs or training on those guidelines for the current LWR operating fleet.

RST Pathway accomplishments include the scoping evaluations documented in this report and in Reference 12. In addition, the following RST activities were completed in 2015 that are relevant to these scoping evaluations and possible future activities related to instrumentation performance during severe accidents:

- *Gap Analysis:* Post-event analyses of the events at Fukushima Daiichi identified several areas that may warrant additional research and development to reduce modeling uncertainties and to assist the industry in development of mitigating strategies and refinement of industry guidance to prevent significant core damage given a beyond design basis event and to mitigate source term release if core damage event does occur. On these bases, a technology gap evaluation on accident tolerant components and severe accident analysis methodologies was completed with the goal of identifying any data and/or knowledge gaps that may exist, given the current state of LWR severe accident research and augmented by insights gained from recent analyses for the Fukushima Daiichi accident. Results from this effort[57] provide a basis for refining DOE research plans to address key knowledge gaps in severe accident phenomenology that affect reactor safety and that are not being directly addressed by the nuclear industry or by the US NRC.
- *Fukushima Forensics and Examinations:* This effort is focused on providing insights into the actual severe accident progression at Fukushima through planning and interpretation of visual examinations and data collection of in-situ conditions of the damaged units as well as collection of samples within the reactor systems and structural components from the damaged reactors as well as associated analyses. As documented in Reference 58, this effort could provide substantial lessons-learned on severe accident progression, similar to those that were learned from TMI-2 acci-

dent examinations. In particular, examinations of instrumentation within the affected units is of interest to the efforts documented in this report and in Reference 12.

### 2.5.2. IAEA Study

The IAEA established an Action Plan on Nuclear Safety in response to the Fukushima Daiichi event. One of the action items of this plan was to provide guidance on “Post-accident and severe accident monitoring systems.” Reference 8 was prepared in response to this action item to reflect current knowledge, experience, and best practices in this area and is based on the results of a series of meetings. It provides a common international technical basis to be considered when establishing new criteria for accident monitoring instrumentation to support operation under DBAs and Design Extension Conditions (DECs) in new plant designs and in existing nuclear power plants. Reference 8 considers monitoring instrumentation and the associated instrumentation support systems for accident prevention and mitigation. Reference 8 addresses instrumentation that is directly used to implement accident management strategies and instrumentation that may be used to validate or backup the directly used instrumentation. This may include permanently installed instruments that are designated for use in accident monitoring, portable instruments, instruments that are installed but not normally in service, and instruments provided to monitor temporary equipment.

Reference 8 recommends that a process, similar to the processes described in Sections 2.3 and 2.4, be implemented to ensure that instrumentation with adequate reliability is available for use during a severe accident. At the end of the process, a reasonable assessment of existing or contemplated plant capabilities should be available, and used in a decision making process. Examples of such decisions are:

- Whether the instrumentation that is already available is adequate for the purpose;
- Whether there are some gaps in information available to the operators, but those gaps can be compensated for, in part or in total, through the use of alternate existing components or instrumentation;
- Whether additional testing or analysis of instrument performance is needed to obtain a better understanding of component or instrument channel capabilities; and
- Whether upgrades in instrumentation systems are needed.

Reference 8 emphasizes that instrumentation survivability analyses must be plant-specific; consequently, conclusions as to what actions are appropriate could differ from one plant to another. The IAEA studies emphasizes the importance of considering the following key aspects of instrumentation:

- *Range* - When determining the accident monitoring instrumentation range, consideration should be given to all analyzed events, including events managed by both EOPs and SAMGs, for which the instrumentation is expected to function.
- *Accuracy* - The accuracy requirements for instrumentation need to consider their intended functions, and how the information provided by the instrumentation is to be used.
- *Response Time* - When determining response time for analogue and digital instrumentation, the instrument’s intended function and potential for any time lags need to be considered.
- *Duration of Operation* - Accident monitoring instrumentation needs to be capable of performing their functions over the duration that they are needed to enable plant operators to appropriately respond to such accidents according to guidelines and procedures.

The IAEA study recommends that accident monitoring instrumentation be developed and maintained in accordance with a nuclear quality assurance program that complies with appropriate guides and to the extent possible, that instrumentation systems be protected and separated from harsh environments (e.g., temperature, pressure, moisture, radiation, shock and vibrations, chemical exposure, electromagnetic fields, voltage surges, etc.).

### 2.5.3. SA-Keisou (Severe Accident - Instrumentation & Monitoring Systems)

The SA-Keisou program was established to develop instrumentation and monitoring systems that could prevent an accident similar to the one that occurred at Fukushima Daiichi.[59,60] The SA-Keisou emphasizes the need to monitor ‘important’ variables, such as reactor water level, reactor pressure, and hydrogen concentration. With this information, operators can prevent an event escalating into a severe accident, mitigate the consequences of a severe accident, achieve a safe state for the plant, and confirm the plant continues to be in a safe state over the long term. The SA-Keisou program addresses BWR and PWR plant instrumentation needs and includes representatives from electric power companies, vendors, and instrumentation manufacturers. The program also has an advisory panel.

The purpose of SA-Keisou is to develop the instrumentation systems needed to provide plant operators with the information they need to mitigate the progression of a severe accident. As shown in Figure 2-6, selection of important parameters or ‘variables’ to be measured is somewhat different than the processes described in Sections 2.3 and 2.4. Candidate variables are determined through an evaluation process that considers: (a) required accident management safety functions; (b) international guidance; and (c) consideration of a sequence similar to the TEPCO Fukushima Daiichi accident (a TF1 accident).

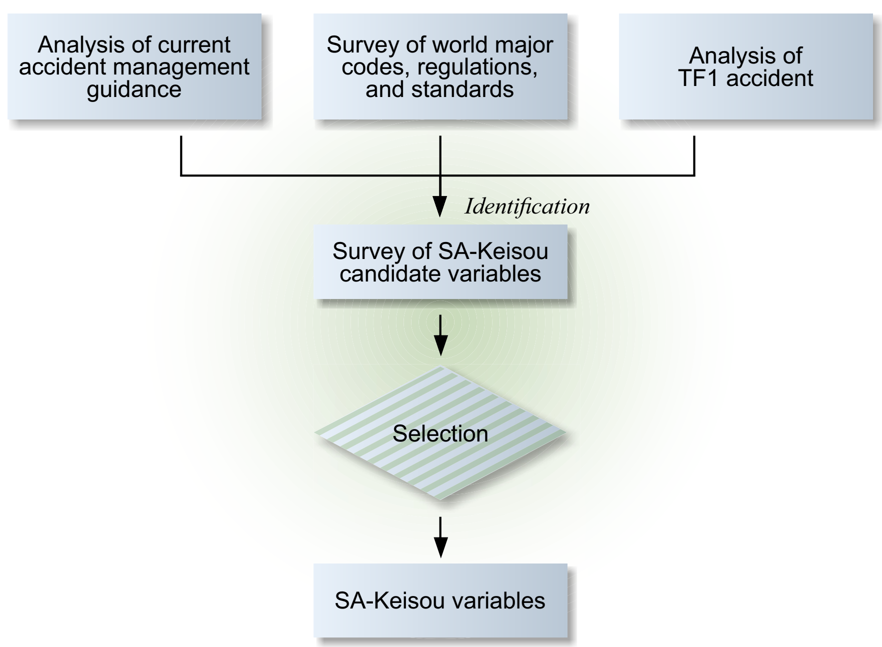



**Figure 2-6.** SA-Keisou important variable selection process for a TF1 accident.

In addition, the SA Keisou program includes research to provide new instrumentation systems for high priority measurements, and it is expected that new sensors will be ready for installation in FY2015. Figure 2-7 shows four new measuring parameters planned for PWR monitoring. New techniques being investigated for measuring water level include techniques based on differential thermocouple methods, heated thermocouple methods, ultrasound-based methods, and gamma ray methods. Hydrogen monitoring will rely on an electrolyte type system that generates a voltage based on concentration differences to provide a real-time signal for hydrogen concentration.

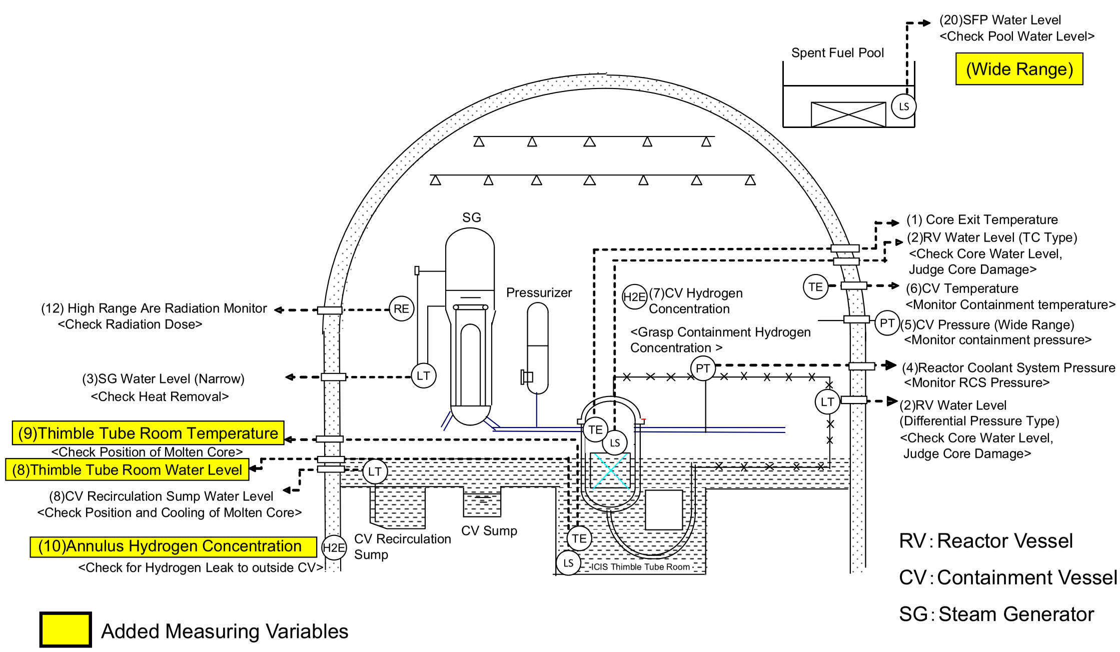



**Figure 2-7.** SA-Keisou selected four new PWR measuring variables.[60]

### 2.5.4. US and International Standards

Several US professional organizations have issued standards related to severe accident instrumentation. After the accident at TMI-2, the American Nuclear Society (ANS) issued ANS Standard 4.5-1980, “Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors,”[61] to provide a functionally-based methodology for categorizing various types of accident monitoring instruments based on the functions served and type of information provided. The Institute of Electrical and Electronics Engineers (IEEE) Standard 497-2002, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,”[62] is also of interest. This standard is currently endorsed (with some clarifying position) by US NRC Regulatory Guide 1.97 Revision 4 (see Section 2.3). The current version of IEEE Standard 497 was released in November 2010, just 4 months before the accident at the Daiichi plants at Fukushima occurred. However, this accident prompted the Nuclear Power Engineering Committee of the IEEE to initiate the next update of IEEE Standard 497, and it is expected that the US NRC will revise Regulatory Guide 1.97 when the updated IEEE Standard 497 is released.[63]

Historically, the International Electrotechnical Commission (IEC) has not had a standard for accident monitoring instrumentation design criteria. However, IEC 61226[64] identifies the instrument functions, classifies them, and defines the applicable requirements; and IEC 60964[65] defines requirements applicable to human-machine interface in the control room. IEC also has standards dealing with specific functions that have a role in accident monitoring such as monitoring radiation release, containment conditions, and core cooling. In addition, IEC has standards for qualifying instrumentation to withstand harsh environments,[64 through 68] including anticipated accident conditions, electromagnetic effects, and seismic events. It has been proposed that IEC should join with the IEEE for issuing a dual-logo standard on accident monitoring systems for nuclear power plants based on the IEEE Standard 497 that is now being revised.

The German Kerntechnischer Ausschuss (KTA) has issued KTA 3502[69] to address accident monitoring instrumentation. The current version of the German KTA 3502 was released in 2012. This standard establishes requirements for equipment that monitors DBAs at LWRs. KTA has also published several standards dealing with monitoring of radioactive releases.

Another perspective is provided by the regulatory guides on nuclear safety (YVL Guides) issued by the Finnish Centre for Radiation and Nuclear Safety (STUK). In the guide YVL 1.0, "Safety criteria for design of nuclear power plants,"[70] severe accidents are specified as DBAs. YVL 1.0 requires that license applicants for new plants assume 100% oxidation of materials in the reactor area be considered in containment design. YVL 1.0 also requires information be provided on re-criticality, pressure vessel melt-through, debris location, and containment threats.

## 2.6. Summary

This section provides background information related to the Surry plant reactor and containment design, accident progression analyses completed in the Surry nuclear power plant SOARCA evaluations, and other relevant sources of information considered in this effort. In particular, this section summarizes prior instrumentation survivability studies and on-going efforts to assess the adequacy of nuclear power plant instrumentation during severe accidents. As discussed within this section, considerable US effort was expended in prior studies and continues today to identify severe accident information needs and instrumentation that could address those needs. In addition, there are significant international efforts underway to evaluate and enhance LWR instrumentation for severe accident conditions.

The scoping efforts documented in this report benefit from this background information. As discussed in Sections 3 through 5, the current scoping evaluation uses approaches previously applied in US NRC and industry studies to evaluate the ability of instrumentation proposed by industry to provide critical information when subjected to conditions predicted during a risk-important severe accident sequence. These scoping efforts rely on state-of-the-art methods for assessing plant response and consider improvements and changes not reflected in earlier assessments, such as enhancements in systems, training and emergency procedures, offsite emergency response, security-related measures, and plant modifications. It is anticipated that results from the current scoping evaluation will inform other on-going US and international efforts on this topic.

# 3. INSTRUMENTATION EVALUATION PROCESS

This section describes the approach used in this scoping evaluation to select critical plant parameters required by operators to diagnose the plant status and to evaluate the effects of mitigating actions taken during an accident. Critical instrumentation systems for providing this information are identified. The locations and EQ values for this instrumentation are estimated based on input from prior Surry plant evaluations and generic PWR plant information.

## 3.1. Approach

The approach for the current evaluation was selected after reviewing prior US and current international efforts on this subject (see Section 2). The selected approach (see Figure 3-1) draws most heavily from the methods developed and deployed by NRC and US industry described in Sections 2.3 and 2.4. In addition, recent efforts by the PWROG to develop enhanced severe accident management guidelines and identify critical information needs and instrumentation for use during a severe accident (Section 2.4.3) influenced the approach adopted for this scoping evaluation. Insights from prior evaluations and on-going international instrumentation survivability evaluations discussed in Section 2 were also considered.

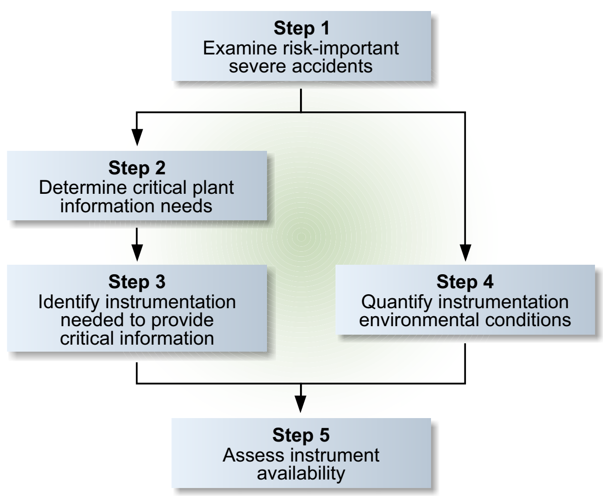



**Figure 3-1.** Approach adopted for current PWR instrumentation scoping evaluation.

As shown in Figure 3-1, the selected approach emphasizes identifying critical plant information needs (Step 2) and instrumentation required to provide this critical information (Step 3). The methods used to completed these steps are described in Section 3.2. The methods used to complete Steps 4 and 5 are discussed in Section 4 and 5.

## 3.2. Critical Plant Information Needs and Instrumentation

As long as the accident progression is within the plant's design basis, the information provided by instrumentation is generally considered to be highly reliable based on design, qualification testing, and redundancy. Once the accident progression causes one or more conditions to exceed the plant's design basis, the information provided by plant instrumentation needs to be scrutinized to increase confidence that an appropriate mitigation strategy is selected.

During a severe accident, plant status parameters are monitored for several purposes:

- Implementing EOPs to prevent extensive fuel cladding damage;
- Implementing SAMGs to mitigate accident progression and bring the plant into and maintain it in a controlled state;
- Assessing the potential magnitude of fission product releases and monitoring such releases together with meteorological conditions for emergency planning actions;
- Assessing environmental conditions for monitoring control room and Technical Support Center (TSC) habitability or for access to selected plant areas in order to perform local actions; and
- Resolving ambiguities in displayed information.

Available information (see Section 2) indicates that the same 'critical' information needs are widely used throughout various guidance documents and aids, such as: EOPs, which include Emergency Contingency Actions (ECAs), Function Restorations (FRs), and the new FSGs; SAMGs, which include Severe Accident Control Room Guideline (SACRGs), and Calculational Aids (CAs); and Extreme Damage Mitigation Guidelines (EDMGs), These information needs provide the basis for plant personnel to select appropriate CHLAs identified in plant guidance. Although it is recognized that the instrumentation EQ ranges may not correspond to their survivability limits, it is useful to identify these critical information needs and compare the EQ ranges for sensors that provide data for these critical needs with conditions predicted in a risk-dominant SOARCA sequence using the process shown in Figure 3-2.

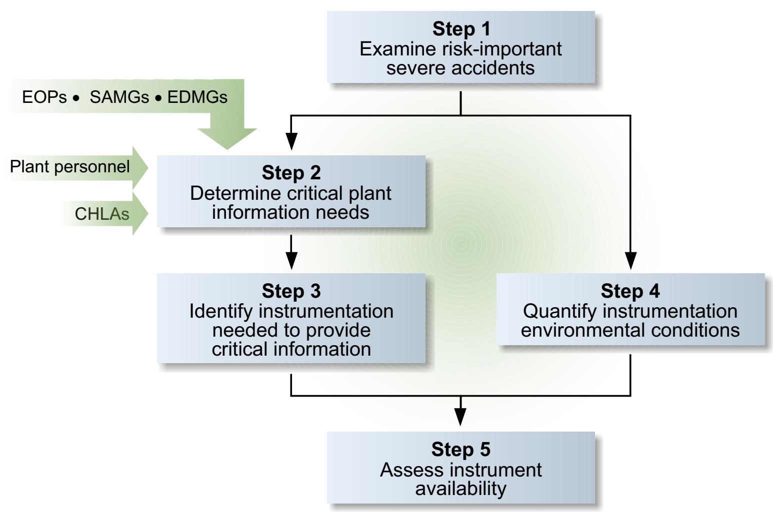



**Figure 3-2.** Identification of critical Surry information needs.

Table 3-1 identifies the critical parameters selected for the current study, the purpose of each of these parameters, and methods for measuring each parameter. This list was developed by starting with the critical parameters and instrumentation proposed by the PWROG in Reference 4 (items NOT in italics). Then, additional items were added (in italics) using insights from prior NRC and industry studies,[33-35, 43-45] Surry plant-specific information,[71-73] and other authoritative PWR instrumentation references.[74-75] It should be noted that the Surry plant currently uses the 'generic' Westinghouse Owner's Group SAMG, so instrumentation listed in this table differs from that specified in current SAMGs at the plant.

**Table 3-1.** Proposed SAMG critical parameters and instrumentation[a]

| Parameter | Primary Purpose/Information Provided | Measurement Method | Alternate Method |
|---|---|---|---|
| Steam Generator (SG) Water Level | • RCS heat sink available<br>• Creep rupture of SG tubes possible<br>• Fission product scrubbing for faulty or leaking SG tubes | • Wide range SG level | • Narrow range SG level[b] |
| SG Pressure | • Creep rupture of the SG tubes possible<br>• Ability to inject into the SGs | • SG secondary pressure[c] | • TDAFW pump header pressure (only for select SGs) |
| RCS Pressure | • Ability to inject into the RCS<br>• High Pressure Melt Ejection possible<br>• Uncontrolled opening in the RCS | • Wide Range RCS pressure | • Pressurizer pressure[d]<br>• Accumulator pressure[e]<br>• Charging pump or LHSI pump discharge pressure |
| *Core Temperature*[f] | • *Transition from EOPs to SAMG*<br>• *In-vessel recovery of core cooling* | • *Core Exit Thermocouples (CETCs)* | |
| *RCS Temperature*[f] | • *Understanding earlier stages of accident progression* | • *CETCs* | • *Resistance Temperature Detector (RTDs) [Hot Leg or Cold Leg]*[g] |
| *RCS Water Level*[h] | • *Understanding earlier stages of accident progression* | • *Reactor Vessel Level Indication System (RVLIS)* | • *Ex-core neutron detectors*[i] |
| RCS Injection Flow[h] | • Water provided to cool the core | • Charging pump flow rate<br>• LHSI flow rate | • *Change in suction source* level[j] |
| Containment Water Level[k] | • Flooding of equipment and instruments<br>• Safety injection or spray recirculation possible<br>• Spillover to the reactor cavity<br>• Ability to quench dispersed core debris | • Wide range containment water level | • RWST level[l]<br>• Narrow range sump level[m] |
| Containment Flammable Gas Concentration | • Containment gas flammability | • Containment hydrogen monitor[n,o] | • Sampling<br>• Calculational Aides |
| Containment Pressure | • Containment over-pressurization | • Wide range containment pressure | • Containment temperature[p] |
| SFP Level[q] | • Ability to maintain spent fuel cooling | • Spent fuel pool level[r] | • Visual[s]<br>• Radiation levels[t] |

a. Non-italics text corresponds to instrumentation proposed by PWROG (References 4 and 53). Italics indicate typical instrumentation and alternate indications suggested by R. Lutz, Lutz Nuclear Safety Consultant (Lutz-NSC), in email dated July 16, 2015,[74] that should be generally applicable to Surry.
b. Off-scale low is in the U-bend region; provides information for adequate water for heat sink and covering tubes for fission product scrubbing; off-scale high is same as wide range SG level so overfill can also be diagnosed. Off-scale low is an indication that more inventory is needed, but it does not indicate how much and cannot diagnose dryout / creep failure potential.
c. There is no direct secondary pressure measurement in the SG. Rather, main steam line pressure is measured in the line upstream of the main steam isolation valve.
d. Only when RCS pressure > 11.7 MPa (1700 psig).
e. Only when RCS pressure < 5.5 MPa (800 psig).
f. Core temperature is a control parameter for existing SAMGs. For the new, enhanced, PWROG SAMGs (see Section 2.4), RCS injection flow is the new control parameter to indicate core cooling. Currently, core temperature is the only parameter for transition from EOPs to SAMGs for both existing and new proposed PWROG SAMGs.
g. Can be confusing after hot leg or cold leg has voiding, two-phase flow, and/or counter current flow.
h. Not a SAMG control parameter.
i. Similar to TMI-2 indications.
j. Although not listed (because it requires plant-specific knowledge), alternate indications at some plants include: narrow range pressure, temperature, or level instrumentation (wide range is typically used in SAMGs because of the extended range); pump discharge pressure; and other mechanical gauges (e.g., steam line pressure).
k. NUREG/CR-5513 (Reference 33) notes the sump level is not a reliable indicator of reactor vessel cavity water level in some PWRs. Plant-specific evaluations will need to consider alternate sensors in such cases.
l. Not applicable for ISLOCA or SGTR events where injected RWST inventory may be discharged from RCS and bypass containment.
m. Can only indicate if sufficient water for recirculation.
n. Typically outside containment and requires AC power
o. Cannot measure carbon monoxide from MCCI.
p. Approximation based on saturation line if no MCCI.
q. SFP instrumentation is not considered in the current study.
r. SFP level indication relies on the instrumentation implemented in response to NRC Order EA-12-051.[40]
s. If not low enough to increase radiation levels but temperature (due to steaming) may be too high for observation.
t. Possible indication of loss of significant inventory (shielding) or spent fuel uncovery.

As discussed in Section 2.4.3, the PWROG is developing SAMGs with TSGs.[4 through 5] The instrumentation TSG is used to support the diagnosis and selection of mitigation strategies as well as to confirm the adequacy of mitigation actions after they are implemented. This TSG is knowledge-based and relies on comparing instrumentation indications with other key information including: alternate instrumentation for the same parameter, assessment of other related or linked parameters (such as pressure and temperature), other indications not directly provided by instrumentation, calculational aids, and expectations for trending of plant parameters based on the accident progression. Guidance is provided for all key parameters needed for effective severe accident management using the new, enhanced PWROG SAMGs. The critical parameters proposed by the PWROG are being evaluated to assess the ability of the plant to respond to challenges, including its physical state (e.g., pressure and temperature) and physical and chemical phenomena (e.g., rapid steam generation and hydrogen burns) that can impact the integrity of the final fission product barrier (i.e., the containment and the SG tubes). Ultimately, Reference 52 indicates that plant-specific applications are being developed using these enhanced generic SAMGs and TSGs.

As discussed in Section 2, information needs and instrumentation availability differ during the phases of a severe accident. As an accident progresses, different plant safety functions are challenged; and harsh environmental conditions will develop in different locations within the RCS, containment, and in some sequences, the auxiliary and turbine buildings. For example, during early stages of a severe accident, operators can use RCS temperature instrumentation methods to determine when they should transition from EOPs to SAMGs. Instrumentation survivability assessments must consider this time dependence.

## 3.3. Location and Qualification of Critical Instrumentation

Table 3-2 provides location, operating range, and EQ information for critical sensors identified in Table 3-1. Much of this information was obtained from prior NRC-sponsored instrumentation survivability studies, such as NUREG/CR-5691.[34] NUREG/CR-5691 indicates that its information was based on a Regulatory Guide 1.97 evaluation for the Calvert Cliffs Nuclear Power Station. However, NUREG/CR-5961 values were checked by comparing them with available Surry plant-specific values[71-73] and authoritative PWR references.[74-75]

In some cases, instrumentation locations were based on expert opinion when available information only referenced a general building or area location. In other cases, two locations were given to account for the possibility that some components, such as cabling, can be exposed to more severe conditions than the sensor and may limit instrumentation system performance. Information from prior studies was used to specify EQ ranges for this scoping evaluation. For this evaluation, it was concluded that relative humidity would not affect availability because prior studies indicate that instrumentation systems were qualified for operation in an environment with 100% humidity.

**Table 3-2.** Critical instrumentation location, operating range, and EQ ranges.

| Instrumentation[a] | Location | Operating Range/Comments | EQ ranges |
|---|---|---|---|
| SG | | | |
| Wide Range SG Level | Containment | 0 to 100% (entire height of SG secondary side) | Maximum Temperature - 149 °C (300 °F)<br>Maximum Pressure - 0.41 MPa (60 psia)<br>Dose = 1 x $10^8$ rad; Humidity - 100% |
| SG secondary pressure | Containment | 0 to 9.8 MPa (0 to 1400 psia) | |
| *Narrow Range SG Level* | *Containment* | *0 to 100% (above U-bend region)* | |
| *TDAFW pump discharge header pressure* | *Turbine Building* | *0 to 9.8 MPa (0 to 1400 psia)* | Maximum Temperature - 38 °C (100 °F)<br>Maximum Pressure - 0.1 MPa (14.7 psia)<br>Dose = << $10^8$ rad; Humidity - 100% |
| *SG main steamline pressure* | *Steamline between SG and Main Steam Isolation Valve (MSIV)* | *0 to 9.8 MPa (0 to 1400 psia)* | Maximum Temperature - 149 °C (300 °F)<br>Maximum Pressure - 0.41 MPa (60 psia)<br>Dose = 1 x $10^8$ rad; Humidity - 100% |
| RCS | | | |
| Wide range RCS pressure | RCS and containment | 0 to 20.7 MPa (0 to 3000 psig) | *Containment Locations*<br>Maximum Temperature - 149 °C (300 °F)<br>Maximum Pressure - 0.41 MPa (60 psia)<br>Dose = 1 x $10^8$ rad; Humidity - 100%<br>*RCS Locations*<br>Maximum Temperature - 1260°C (2300 °F)<br>Maximum Pressure - 17.2 MPa (2500 psia)<br>Dose = 1 x $10^8$ rad<br>Humidity - 100% |
| *Accumulator pressure* | | *0 to 5.5 MPa (0 to 800 psig)* | |
| *Pressurizer pressure* | | *11.7 to 17.2 MPa (1700 to 2500 psig)* | |
| *CETCs* | *RCS and containment* | *93 to 1260 °C (200 to 2300 °F)* | |
| *Cold leg RTD* | *RCS and containment* | *18 to 371 °C (0 to 700 °F)* | |
| *Hot Leg RTD* | *RCS and containment* | *18 to 371 °C (0 to 700 °F)* | |
| RVLIS | RCS and containment | 0 to 100% vessel height | |
| *Power Range Monitors* | *RCS and containment* | *0 to 120% power* | |
| *Source Range Monitors* | | *0 to $10^6$ CPS* | |
| RCS Injection Flow | | | |
| *LHSI pump flow rate* | *Auxiliary building* | *760 to 12,870 liter/m (200 to 3400 gpm)* | Maximum Temperature - 38 °C (100 °F)<br>Maximum Pressure - 0.1 MPa (14.7 psia)<br>Dose = << $10^8$ rad; Humidity - 100% |
| *LHSI pump discharge pressure* | *Auxiliary building* | *0 to 3.5MPa (0 to 500 psig)* | |
| *Charging pump injection flowrate* | *Auxiliary building* | *0 to 12,870 liter/m (0 to 3400 gpm)* | |
| *Charging pump discharge pressure* | *Auxiliary building* | *0 to 17.2 MPa (0 to 2500 psig)* | |
| Containment | | | |
| Containment Sump Level (wide range monitor */narrow range monitor)* | Containment | 0 to 108 inches/ 0 to 30 inches | Maximum Temperature - 149 °C (300 °F)<br>Maximum Pressure - 0.41 MPa (60 psia)<br>Dose = 5 x $10^7$ rad; Humidity - 100% |
| *Containment Temperature* | *Containment* | *4 to 182 °C (40 to 360 °F)* | Maximum Temperature - 149 °C (300 °F)<br>Maximum Pressure - 0.41 MPa (60 psia)<br>Dose = 1 x $10^8$ rad; Humidity - 100% |
| Containment Hydrogen Monitor | Containment | 0 to 10%; degraded performance due to hydrogen burns, containment heating. | |
| *Sampling for Hydrogen* | *Auxiliary Building* | *high temperatures and radiation fields could limit access.* | Maximum Temperature - 38 °C (100 °F)<br>Maximum Pressure - 0.1 MPa (14.7 psia)<br>Dose = << $10^8$ rad; Humidity - 100% |
| Wide range containment Pressure | Containment-Auxiliary building | -0.03 (vacuum) to 1.2 MPa [- 4 psig (vacuum) to 180 psig]; degraded performance due to hydrogen burns, containment heating, and in auxiliary building due to a bypass conditions. | *Containment Locations*<br>Maximum Temperature - 149 °C (300 °F)<br>Maximum Pressure - 0.41 MPa (60 psia)<br>Dose = 1 x $10^8$ rad; Humidity - 100%<br>*Auxiliary Building Locations*<br>Maximum Temperature - 38 °C (100 °F)<br>Maximum Pressure - 0.1 MPa (14.7 psia)<br>Dose = << $10^8$ rad; Humidity - 100% |
| *RWST Level* | *Outdoors* | *0 to 90%* | Maximum Temperature - 38 °C (100 °F)<br>Maximum Pressure - 0.1 MPa (14.7 psia)<br>Dose = << $10^8$ rad; Humidity - 100% |

a. Non italic text designates instrumentation PWROG proposed in References 4 and 53; Italics indicate typical plant instrumentation suggested by R. Lutz, Lutz-NSC, in email dated July 16, 2015,[74] that should be generally applicable to Surry.

References 33 through 35 assumed that instrument performance was degraded if pressure and temperature environments exceed instrumentation EQ ranges. These earlier studies recognized that the assumption of degraded instrument performance for all conditions exceeding the EQ ranges may be conservative, particularly if environmental conditions exceed EQ values by small amounts or for short periods of time. Furthermore, for parameters such as containment temperature, sensor operating temperature ranges exceed EQ values. However, these studies emphasized that instrument system performance is not well known when qualification conditions are exceeded. It was not possible for prior studies or the current study to determine if components of the instrument systems are sufficiently protected to withstand the temperature pulse expected during risk important events. Hence, there is a need to consider specific conditions expected during accident scenarios, the failures of instrumentation system components, such as cabling and splicing, and the plant-specific location of instrumentation components. An assessment of the relationship between the instrument uncertainties and the timing and degree to which the qualification conditions are exceeded would require a detailed study of basic instrument capabilities and failure modes. Although this is beyond the scope of the current study, such an evaluation is warranted if predicted conditions are well beyond qualification conditions.

## 3.4. Summary

Critical information needs and instrumentation capable of providing these needs are identified in this section using information proposed by the PWROG, insights from prior NRC and industry studies, Surry plant-specific information, and PWR expert opinion. An effort to quantify the locations, operating ranges, and EQ ranges for these critical sensors was completed using available plant-specific information and other relevant sources. This information is provided for comparison with information in Section 4.

# 4. MELCOR MODEL AND RESULTS

This section presents environmental conditions predicted by Sandia National Laboratory (SNL)[76-77] using the current version of the MELCOR code (Version 2.1) for the unmitigated STSBO SOARCA sequence. The MELCOR code[18,20] is a fully integrated, engineering-level computer code, capable of modeling accident progression in LWRs. The MELCOR model for the Surry STSBO is also described in this section. MELCOR results for this scoping evaluation provide insights about the conditions that instrumentation systems must survive. As discussed in References 78 and 79, improvements were made in the MELCOR code and the Surry plant model between the time that the original SOARCA calculations were performed and the time that supplemental calculations were completed for this scoping evaluation. These differences, which are identified in this section, appear to primarily affect the timing of the temperature and pressure response within the RCS and containment.

## 4.1. MELCOR SURRY MODEL AND EQUATIONS

Figures 4-1 through 4-3 show the configuration of the SOARCA hydrodynamic model used to simulate the Surry plant with the MELCOR Version 1.8.6 code.[11] The model includes explicit representation of the entire RCS including each of the three reactor coolant loops¸ SGs, the steam lines out to the isolation valves, and associated safety and power-operated relief valves. The pressurizer, associated safety and power-operated relief valves, and the pressurizer relief tank are modeled in Loop C. Boundary conditions are used to represent the turbine pressure and feedwater flow to allow direct calculation of the nominal, full-power steady state operating conditions. The SG nodalization (Figure 4-2) explicitly models the primary-side tubes, the SG inlet and outlet plena, the secondary side, the steam lines, and the relief valves. The red flow paths are only active in natural circulation conditions, and the hot leg and SG tubes are split to permit modeling of counter-current natural circulation flows.

Figure 4-4 shows the containment hydrodynamic nodalization. The containment is divided into nine control volumes (CVs) and seventeen flow paths. The control volumes represent the basement, the cavity under the reactor, the three separate SG cubicles, the pressurizer cubicle, the pressurizer relief tank (PRT) cubicle, the lower dome, and the upper dome. The basement region includes lower containment locations as well as the surrounding cavity that lies between the outer wall and internal crane wall.

The MELCOR model for the SOARCA program included the Auxiliary Building for bypass sequences (e.g., ISLOCA and SGTR events). The Surry MELCOR model also considers the Safeguards Area, Containment Spray Pump Area, and Main Steam Valve House. However, as discussed in Section 4.2, Reference 11 indicates that containment failure was predicted at a location that allowed all fission products to be released directly to the environment in this unmitigated STSBO sequence. In addition, heating in the Auxiliary Building due to electrical loads required to run instrumentation would not be significant in this STSBO because all AC and DC power is lost. Hence, no Auxiliary Building locations were evaluated in MELCOR calculations for this unmitigated STSBO sequence. In bypass sequences with release into the Auxiliary Building, instrumentation may be adversely affected. Likewise, instrumentation components could be challenged by heating due to electrical loads in the Auxiliary Building. As noted in Section 2.4.3, current industry efforts include strategies to circulate air in key rooms in the auxiliary building to prevent any adverse impact on power supplies and/or critical instrumentation. Efforts to expand the current scoping evaluation could be used to quantify the timing and conditions that challenge instrumentation in the Auxiliary Building.

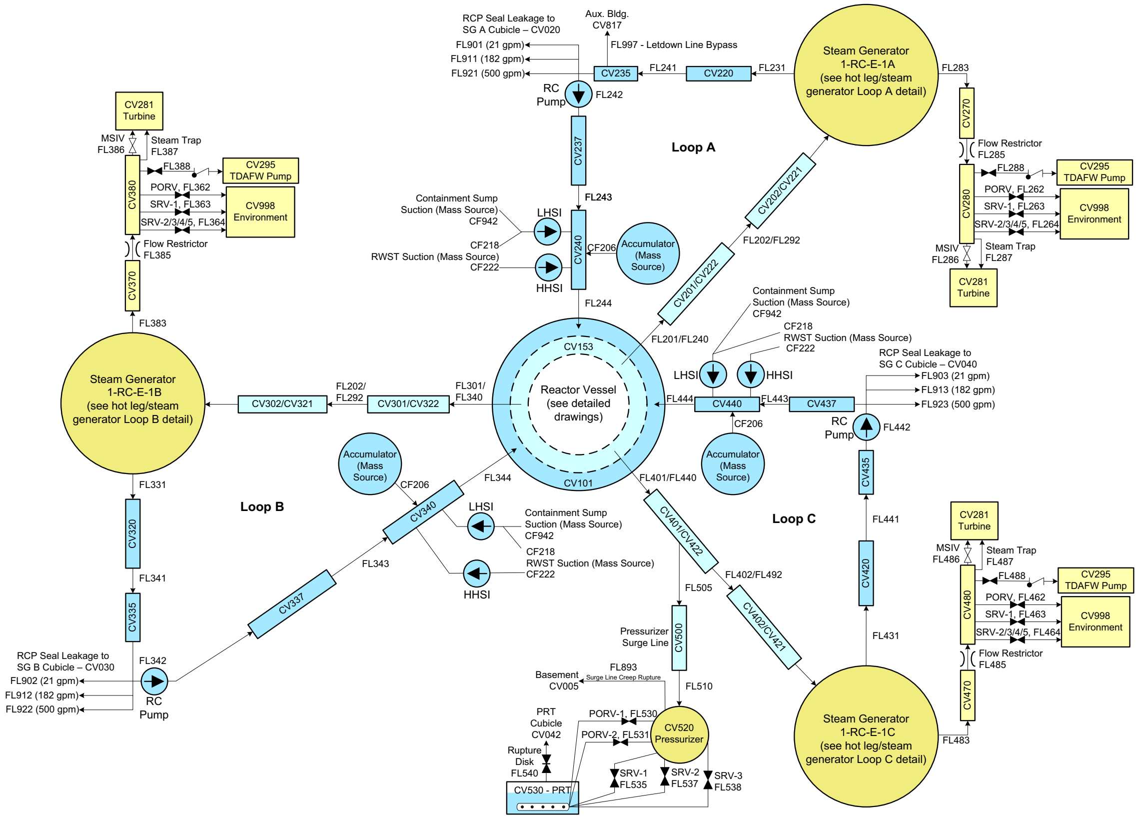
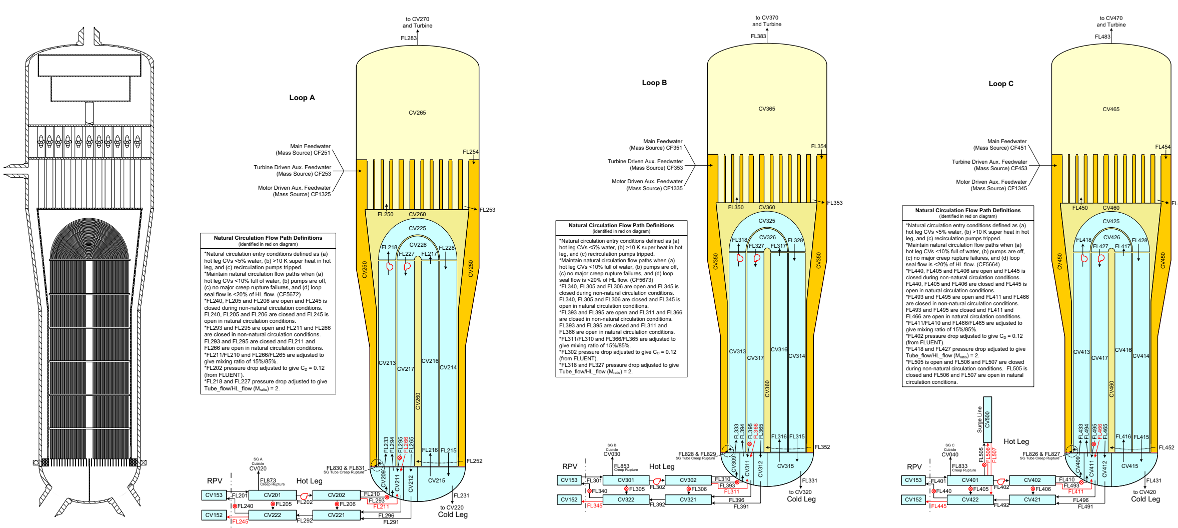



**Figure 4-1.** MELCOR model for Surry RCS.[11]




a) Loop A Nodalization

b) Loops B and C Nodalization

**Figure 4-2.** MELCOR model for Surry SG.[11]

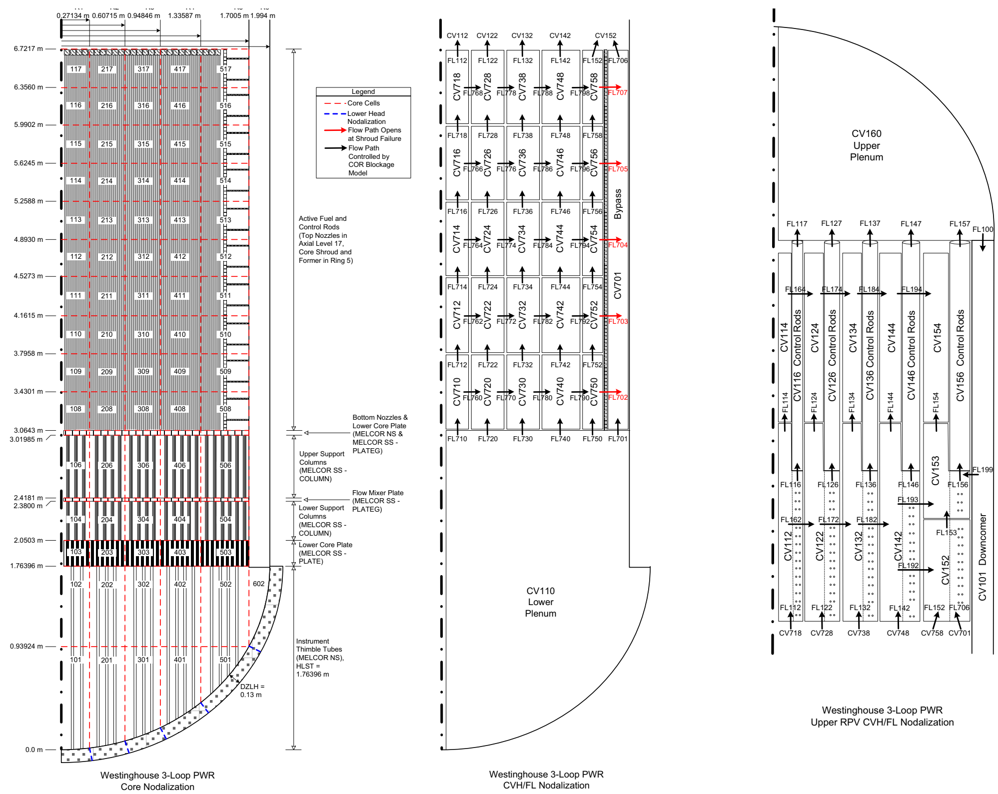



**Figure 4-3.** MELCOR model for Surry reactor vessel nodalization.[11]

## 4.2. SOARCA STSBO ACCIDENT SCENARIO

As discussed in Section 2.2, risk-important events were identified in the SOARCA effort using results from existing PRAs performed by the US NRC and by the licensee for the Surry plant.[9] In general, core damage sequence groups with a CDF equal to or greater than $10^{-6}$ per reactor-year were considered. However, the SOARCA program also evaluated sequence groups leading to an early failure or bypass of the containment with a CDF equal to or greater than $10^{-7}$ per reactor-year because these sequence groups have a potential for higher consequences and risk. This approach allowed a more detailed analysis of accident consequences for the more likely, although still remote, accident scenarios.

As shown in Figure 4-5, three sequence groups were selected in the SOARCA evaluations: STSBO, LTSBO, and ISLOCA. A broad range of challenges provides more insights related to the conditions that instrumentation may experience during severe accidents. Results in NUREG/CR-5691[34] indicate that different sequences provide different challenges to instrumentation, based on the location of the sensor, its EQ ranges, and the duration of the challenge. In particular, it is observed that LTSBO events and containment bypass events, such as ISLOCA or SGTR events, will present different challenges since fission products are released into the Auxiliary Building. However, the current scoping study only considered SOARCA results for an unmitigated STSBO sequence [e.g., as discussed in Section 2.2, an analysis without assuming the benefit of actions specified in EOPs and SAMGs or the use of 10 CFR 50.54(hh) equipment].

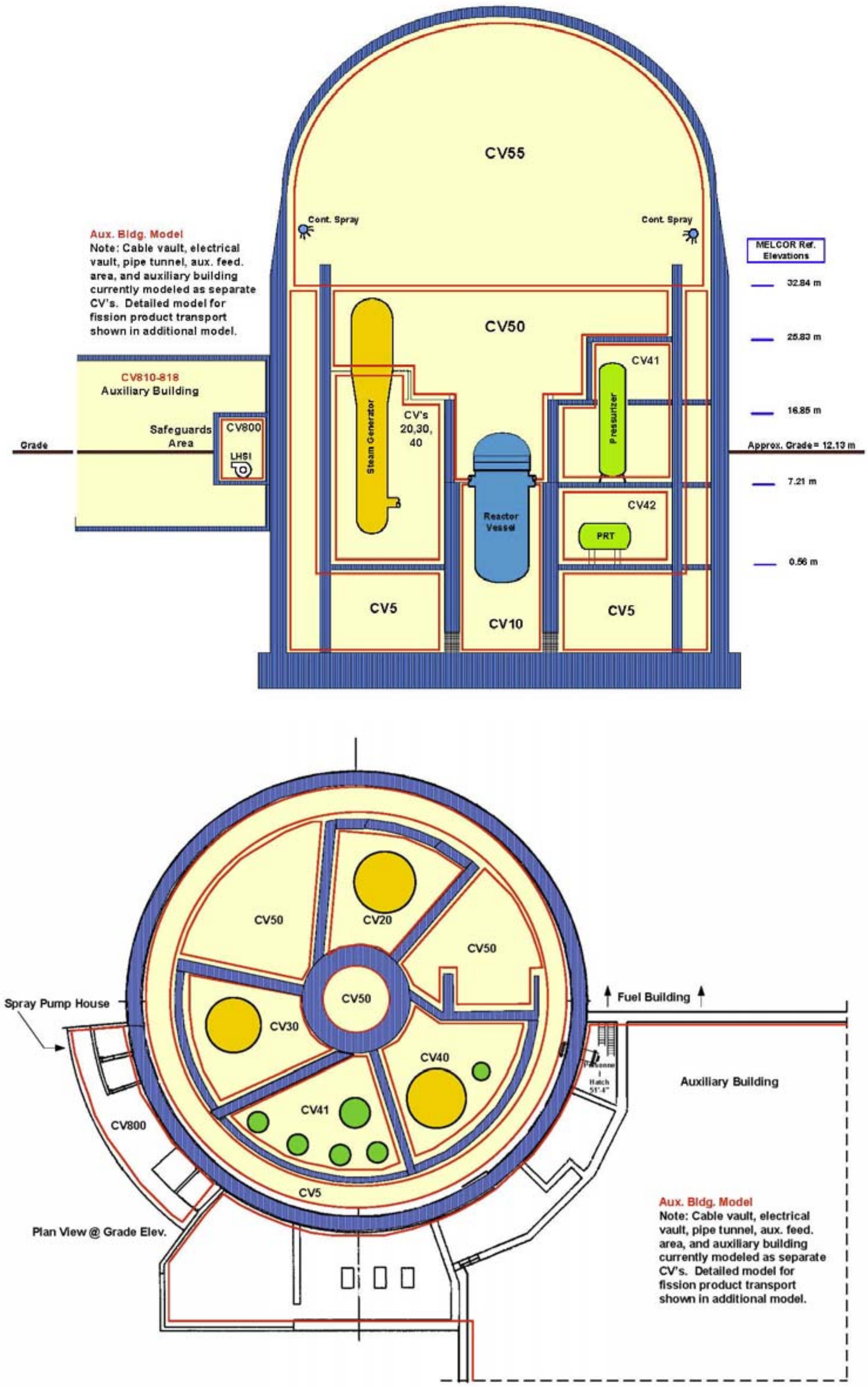



**Figure 4-4.** MELCOR model for Surry containment.[11]

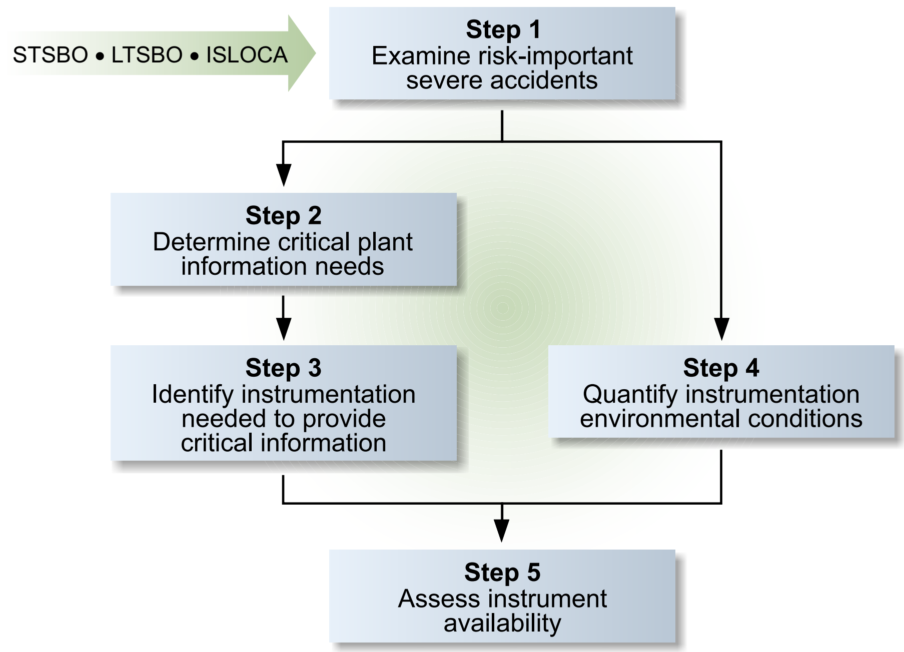



**Figure 4-5.** Accident sequences identified for Surry instrumentation survivability study (only the unmitigated STSBO was evaluated in this scoping evaluation).

The unmitigated STSBO is assumed to be initiated by an earthquake with a 0.5 to 1.0 g peak ground acceleration and to have a CDF ranging from 1 to 2 x $10^{-6}$ per reactor year. Timing for key events for this accident scenario are summarized in Table 4-1. Predicted responses for primary and secondary pressure, reactor vessel water level, peak fuel cladding/debris temperature, and containment pressure are shown in Figures 4-6 through 4-10.[11] Note that MELCOR results presented in these figures are from Reference 11 and were obtained using MELCOR Version 1.8.6.[18] Results for the unmitigated STSBO presented in Reference 11 are included because they provide perspectives that are needed for understanding the progression of this accident sequence. Supplemental results presented in Section 4.3 were obtained using MELCOR Version 2.1[20] and an upgraded Surry plant model. The supplemental calculation results differ because of improvements made in the updated code and the Surry plant models. However, as elaborated in Section 4.3, these differences appear to primarily affect the timing of the temperature and pressure response within the RCS and containment.

At the start of the sequence, the site loses all power, e.g., offsite power, onsite emergency power, and batteries. This results in a station blackout where neither onsite nor offsite AC power is recoverable, and the sequence is referred to as a STSBO because all of the safety systems become quickly inoperable in the "short term." The earthquake does not initially damage the RCS and containment, but no instrumentation is assumed to be available. Significant structural damage, including structural failure of the turbine building occurs. Auxiliary Building accessibility is assumed to be difficult, due to fallen piping and cabling, steam and water leaks, and damaged stairways.

**Table 4-1.** Timing of key events in SOARCA STSBO.[11]

| Event Description | Time (hh:mm) |
|---|---|
| Initiating event<br>Station blackout – loss of all onsite and offsite AC and DC power | 00:00 |
| MSIVs close<br>Reactor trip<br>RCP seals initially leak at 79 liters/min/pump (21 gpm/pump)<br>TDAFW starts but fails to inject due to ECST rupture | 00:00 |
| First SG SRV (Safety Relief Valve) opening | 00:03 |
| SG dryout | 01:16 |
| Pressurizer SRV opens | 01:27 |
| PRT rupture disk opens | 01:46 |
| Start of fuel heatup | 02:19 |
| RCP seal failures | 02:45 |
| First fission product gap releases | 02:57 |
| Creep rupture failure of the C loop hot leg nozzle | 03:45 |
| Accumulators start discharging | 03:45 |
| Accumulators are empty | 03:45 |
| Vessel lower head failure by creep rupture | 07:16 |
| Debris discharge to reactor cavity | 07:16 |
| Cavity dryout | 07:27 |
| Containment at design pressure (0.31 MPa/45 psig) | 11:00 |
| Start of increased leakage of containment ($P/P_{design}$ = 2.18) | 25:32 |
| Containment pressure increase slows | 32:00 |
| Containment pressure stops decreasing | 44:14 |
| End of calculation | 48:00 |

In response to the loss of power, the reactor successfully trips, and the main steam line isolation and containment isolation valves close. The reactor coolant and main feedwater pumps also trip due to the loss of power. This causes primary and secondary system pressures to rise (see Figure 4-6). The secondary system relief valve opens, but then closes when the pressure falls below the closing setpoint. This relief flow through the SG SRVs is the principle energy removal mechanism from the primary system during the first hour. There is also a small amount of energy removal through the leakage from the RCP seals.

The seismic event also causes a loss of DC power and the TDAFW system. The complete loss of all feedwater leads to a rapid decrease of the water inventory in the SGs. As the SG water inventory boils away (at 1 hour and 16 minutes), the primary system pressure increases until the pressurizer SRV opens to remove excess energy. The pressurizer relief valve flow causes a steady decrease in primary coolant system inventory, leading to fuel uncovery at 2 hours and 19 minutes (see Figure 4-7). Following fuel uncovery, an in-vessel natural circulation flow develops between the hot fuel and cooler structures in the upper plenum and between hot gases in the vessel and the SG. The hot gases from inside the vessel flow along the top of the hot leg into the SG and are predicted to cause hot leg nozzle failure at 3 hours and 45 minutes. This failure rapidly depressurizes the RCS, permitting accumulator injection. Although the water level temporarily increases, decay heat from the fuel subsequently boils off the injected water [all active Emergency Core Cooling Systems (ECCS) are unavailable due to loss of AC power].

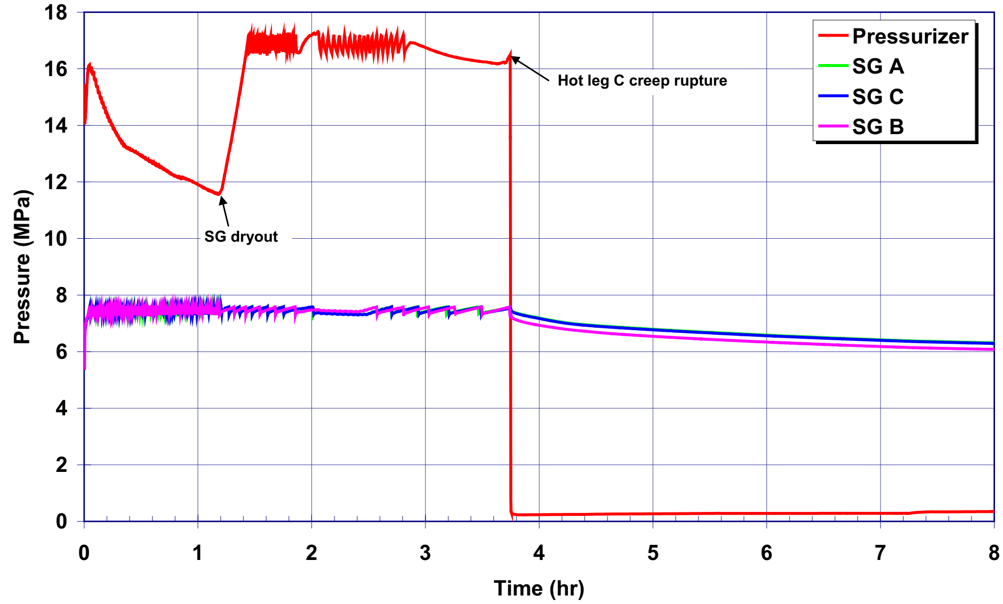



**Figure 4-6.** STSBO primary and secondary response.[11] [Supplemental SNL data discussed in Section 4.3 indicate that the magnitude of pressure response is similar, but the transient progresses slower.]

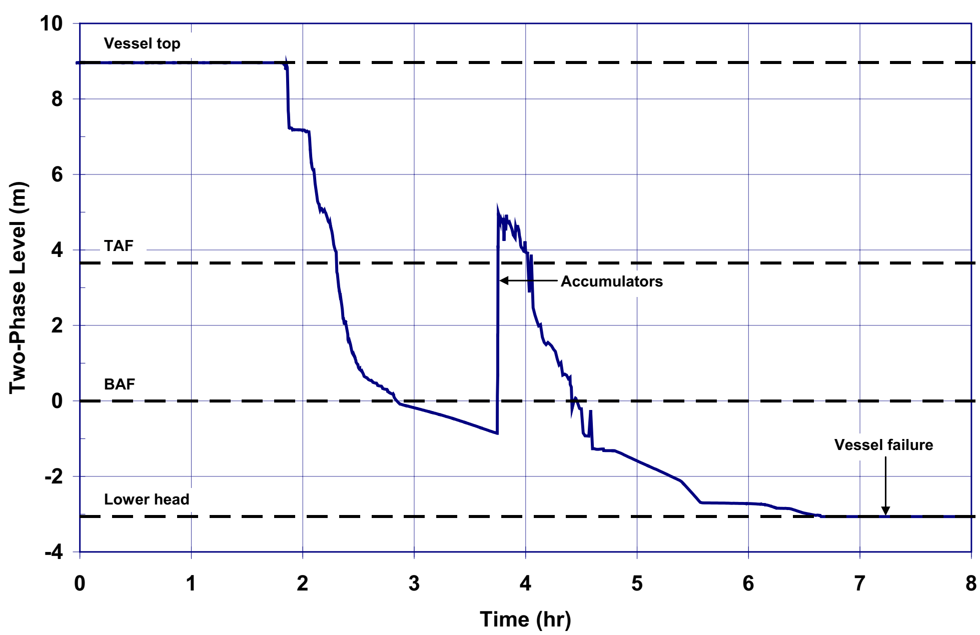



**Figure 4-7.** STSBO vessel water level.[11]

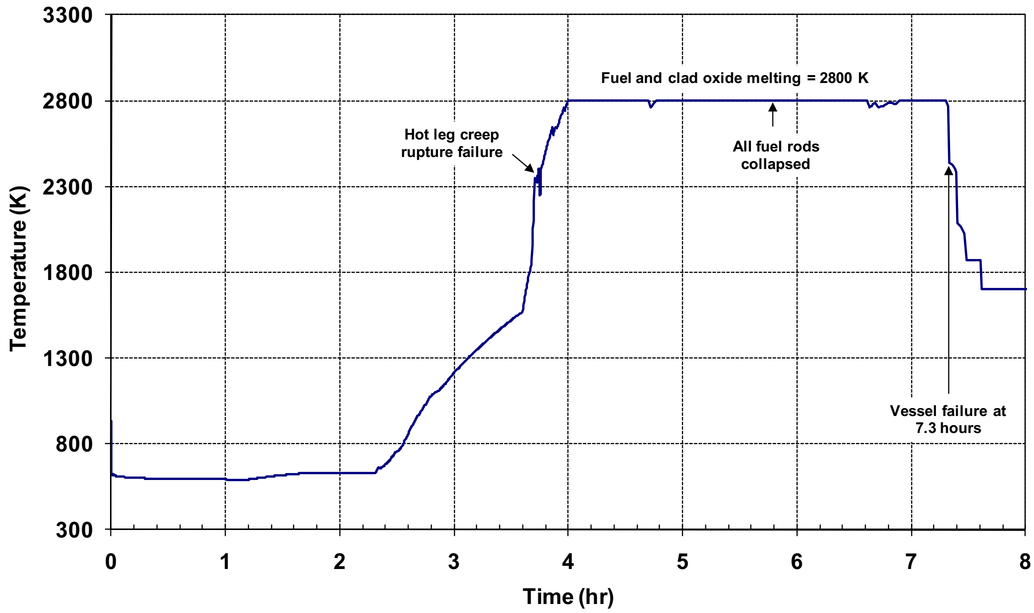



**Figure 4-8.** STSBO peak cladding/debris temperature.[11]

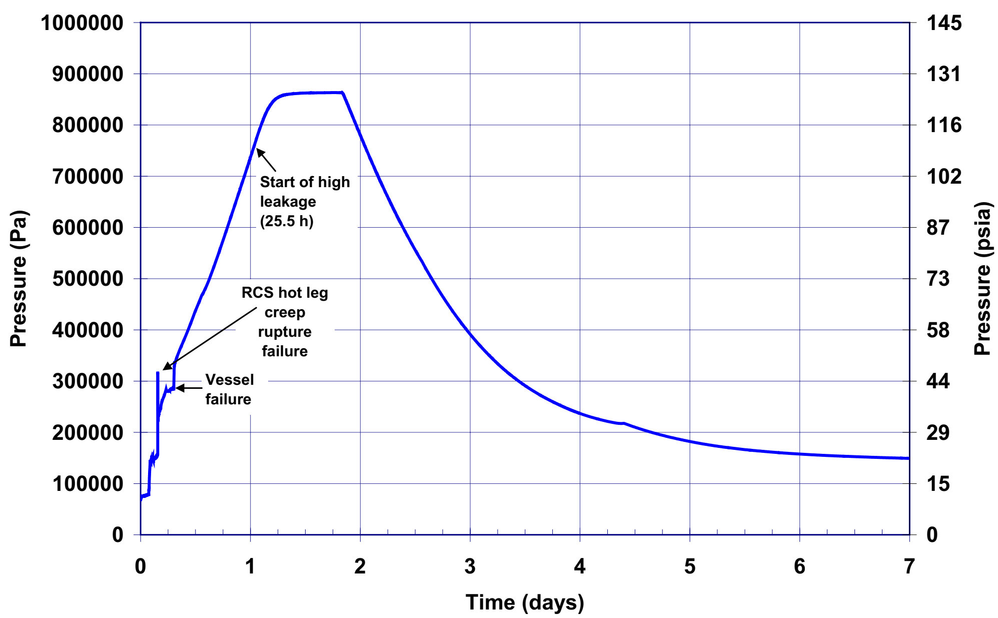



**Figure 4-9.** STSBO containment pressure[11] [Reference 11 doesn't designate which containment volume was selected. However, supplemental SNL data discussed in Section 4.3 indicate similar pressures for all containment volumes and a slower pressure response.].

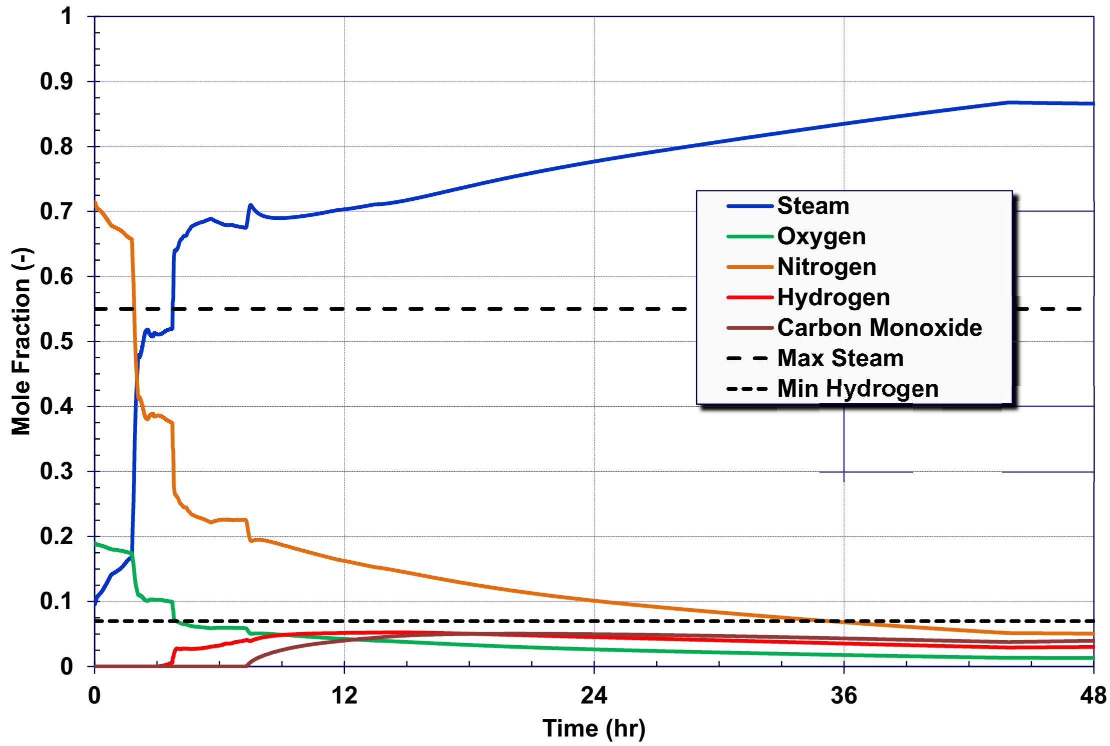



**Figure 4-10.** STSBO containment gas concentrations.[11]

A large debris bed is predicted to form within the core (see Figure 4-8). This debris bed relocates to the core plate at 6.6 hours, resulting in core plate failure and debris relocation to the lower head of the reactor vessel. At 7 hours and 16 minutes, the lower head is predicted to fail. Containment failure (at a location around the equipment hatch) and associated radiological release are predicted to occur at 25.5 hours (see Figure 4-9). This predicted failure is located on the side of the containment without a surrounding building (e.g., not adjacent to the auxiliary or safeguards buildings). Hence, Reference 11 predicts that all fission products are released directly into the environment; and instrumentation components located within the Auxiliary Building are not expected to see any significant changes in environmental conditions.

Figure 4-10 compares predicted containment gas concentrations during this sequence. The steam concentration in the containment rapidly increases to ~51% following PRT failure at 1 hour and 46 minutes. Then, the steam concentration remains relatively constant until hot leg failure. At hot leg failure, the steam concentration increases to ~68% and remains above this level until the end of the calculation. Experimental research[80] indicates that it is essentially impossible to ignite hydrogen or sustain a burn at steam concentrations of 55% or higher. Similarly, the same experimental research showed that a minimum concentration of hydrogen and oxygen is needed before hydrogen will ignite and burn. For this case, Reference 11 indicates that the hydrogen concentration must be above the minimum of 7% for combustion, and the steam concentration must be above 55% to preclude combustion. These values are also shown in Figure 4-10. Although the in-vessel hydrogen production is significant, Reference 11 concluded that combustible conditions did not exist in the containment throughout the evaluated 48 hours. The steam concentration is above the minimum threshold (55%) for combustion whenever any significant amount of hydrogen is present, and the hydrogen concentration is always below the minimum ignition threshold (7%).

## 4.3. STSBO ENVIRONMENTAL CONDITIONS FOR INSTRUMENTATION

The SOARCA results presented in Reference 11 for an unmitigated STBSO sequence did not include all of the locations of interest for instrumentation survivability evaluations. Hence, as part of this scoping evaluation, SNL was funded to provide supplemental thermal-hydraulic and dose conditions at locations of interest.[76,77] Selected results from these supplemental evaluations are presented in this section. Additional results may be found in Appendix C.

### 4.3.1. Limitations and Considerations associated with the Scoping Evaluation

As discussed throughout this document, there are several limitations and considerations associated with this scoping evaluation. Limitations and considerations with respect to respect to results from these MELCOR calculations are highlighted in this section.

First, as discussed in Section 4.2, it is unclear that instrumentation would survive the earthquake that initiated this event. Furthermore, in this unmitigated STSBO, all AC and DC power is lost, so there is no power for instrumentation. In fact, Reference 11 assumes that no instrumentation is available.

Second, the predicted containment failure location precluded any releases from entering the auxiliary building for this unmitigated STSBO sequence and the loss of all AC and DC power precluded heatup from electrical loads associated with powering instrumentation. Hence, instrumentation located within the Auxiliary Building are not expected to be exposed to any significant changes in environmental conditions in this STSBO.

Third, as discussed in Section 4.2, Reference 11 predicted that concentrations of combustible gases and steam within the containment would preclude any concerns about hydrogen burns during this event. Hence, there was no need to further evaluate information related to gas concentrations for this sequence.

Fourth, there are discrepancies between the MELCOR results documented in the Reference 11 SOARCA report and the supplemental MELCOR results provided by SNL for this scoping evaluation. It was concluded that these differences were due to improvements in the MELCOR code and the Surry model used for the scoping evaluation and that results primarily affected timing rather than peak conditions within the RCS and containment. Hence, it is judged that these differences do not impact conclusions from this scoping evaluation.

Fifth, there are limitations associated with the methods used by SNL to estimate dose in these supplemental calculations. Although the MELCOR code tracks time-dependent fission product inventory in various control volumes, the code does not calculate radiation dose. However, SNL staff developed an algorithm for estimating time-dependent radiation dose in a node based on the fission product inventory in the node. Although this algorithm provides an initial estimate for radiation dose, there are several limitations:

- It neglects dose contributions from neighboring nodes or the potential effects of structures within the nodes that could shield certain components.
- There is a lack of node-to-node communications (e.g., radiation from core debris on the containment floor would not be incorporated).
- It neglects steam shielding effects (e.g., steam provides a degree of shielding by absorbing the beta and, to a lesser extent, gamma radiation). Assuming only air is present may overestimate the dose.

- It neglects the effects of deposited activity on surfaces near (or on) components of interest. This can be important for components on or near surfaces that are not washed by condensation flow.
- It neglects the effects of piping that carries superheated steam, such as the effects on electrical components located near an uninsulated line downstream of a PWR pressurizer relief valve.

Despite the above limitations, useful insights can be obtained with respect to critical instrumentation survivability by evaluating supplemental SNL results for this unmitigated STSBO sequence.

### 4.3.2. Results for Representative Locations

The SNL supplemental results were reviewed, and the representative CVs listed in Table 4-2 were selected to represent RCS and containment building locations of interest for survivability assessments (e.g., environmental conditions were bounded by conditions predicted for the selected CVs). Predicted peak temperatures and pressures and cumulative doses are listed for each of these representative control volumes in Table 4-2. Selected results for these representative CVs are discussed in this section. Additional results from these supplemental SNL calculations are included in Appendix C.

**Table 4-2.** MELCOR unmitigated STSBO results for representative CVs.

| Location | ID | Description | Peak Pressure, MPa | Peak Temperature, °C (K) | Beta Dose[a], Rad | Gamma Dose[a], Rad |
|---|---|---|---|---|---|---|
| Reactor Coolant System | CV 101 | Vessel downcomer | 17.3 | 907 (1180) | 1.8E+08 | 1.4E+08 |
| | CV 110 | Vessel lower plenum | 17.4 | 1643(1916) | 1.7E+10 | 3.1E+08 |
| | CV 160 | Vessel upper plenum | 17.3 | 894 (1167) | 5.1E+08 | 7.7E+08 |
| | CV201 | Hot leg loop A | 17.3 | 885 (1158) | 3.0E+08 | 4.2E+08 |
| | CV202 | SGA hot leg top | 17.3 | 874 (1147) | 2.8E+08 | 1.6E+08 |
| | CV 209 | SGA lower plenum | 17.3 | 868 (1141) | 2.8E+08 | 3.3E+07 |
| | CV220 | SGA relief line to auxiliary building | 17.3 | 370 (643) | 1.3E+08 | 1.8E+07 |
| | CV221 | SGA hot leg bottom | 17.3 | 584 (857) | 2.5E+08 | 4.9E+08 |
| | CV 237 | CL RCS piping from RCP | 17.3 | 776 (1049) | 9.8E+07 | 1.2E+07 |
| | CV240 | RCS piping (accumulator injection) | 17.3 | 759 (1032) | 1.1E+08 | 3.4E+07 |
| | CV301 | Hot leg loop B | 17.3 | 885 (1158) | 3.0E+08 | 4.1E+08 |
| | CV302 | SGB hot leg top | 17.3 | 874 (1147) | 2.9E+08 | 1.6E+08 |
| | CV309 | SGB lower plenum | 17.3 | 868 (1141) | 2.8E+08 | 3.3E+07 |
| | CV321 | SGB hot leg bottom | 17.3 | 586 (859) | 2.5E+08 | 4.9E+08 |
| | CV 401 | Hot leg loop C | 17.3 | 897 (1170) | 2.3E+08 | 8.0E+08 |
| | CV 402 | SGC hot leg top | 17.3 | 886 (1159) | 2.3E+08 | 2.0E+08 |
| | CV 421 | SGC hot leg bottom | 17.3 | 614 (887) | 2.1E+08 | 6.1E+08 |
| | CV 701 | Vessel bypass | 17.3 | 1580 (1853) | 1.3E+08 | 7.9E+08 |
| Pressurizer and Piping | CV 500 | Pressurizer surge line | 17.3 | 705 (978) | 3.0E+08 | 2.2E+08 |
| | CV 520 | Pressurizer | 17.2 | 455 (728) | 1.8E+08 | 4.2E+07 |
| Containment | CV 5 | Basement | 0.80 | 351 (624) | 1.6E+09 | 3.4E+09 |
| | CV 10 | Reactor vessel cavity | 0.80 | 1640 (1913) | 1.1E+08 | 2.3E+06 |
| | CV 20 | SG A cubical | 0.80 | 238 (511) | 7.5E+07 | 1.9E+08 |
| | CV 30 | SG B cubical | 0.80 | 242 (515) | 7.4E+07 | 8.8E+07 |
| | CV 40 | SG C cubical | 0.80 | 381 (654) | 7.5E+07 | 1.1E+08 |
| | CV 41 | Pressurizer cubical | 0.80 | 267 (540) | 7.4E+07 | 2.2E+07 |
| | CV42 | Pressurizer relief tank cubical | 0.80 | 276 (549) | 8.6E+07 | 4.5E+07 |
| | CV 50 | Lower dome | 0.80 | 283 (556) | 1.0E+08 | 5.5E+08 |
| | CV55 | Upper dome | 0.80 | 227 (500) | 7.3E+07 | 4.3E+06 |

a. Cumulative dose after one year. As indicated by figures presented in this section, a significant fraction of the dose is accumulated within the first few days of the accident.

#### 4.3.2.1. Reactor Coolant System

As indicated in Section 3.3, instrumentation system components located within the RCS are tested at EQ pressures of 17.2 MPa, temperatures of 1260 °C (1533 K), and gamma doses of $1x10^8$ Rads. Results in Table 4-2 indicate that predicted peak RCS conditions exceed these values. In particular, sensors located in the lower plenum (CV 110) and the bypass region (CV 701) of the reactor vessel are exposed to conditions beyond their EQ values. In fact, in some cases, peak temperatures exceed melting temperatures of structures within the RCS. However, as observed in Section 3.3, it is unclear if sensors exceeding the EQ temperatures will fail, particularly if the environmental conditions exceed the EQ values by only small amounts or for short periods of time. As shown in Figure 4-11, pressures in the RCS only slightly exceeded EQ values for 2 hours (until the time when the RCS depressurized due to hot leg failure).* Data in Figure 4-12 indicate that temperatures in the lower plenum and bypass volumes exceed EQ conditions for less than 2 hours and that structures within the lower plenum exceed qualification environments for approximately 4 hours, respectively. It should be noted that temperatures are predicted to start peaking at around 6 hours, which is before the time when vessel lower lead failure is predicted. Hence, sensors are exposed to conditions beyond their EQ ranges during the time when it is critical for operators to have access to accurate sensor data.

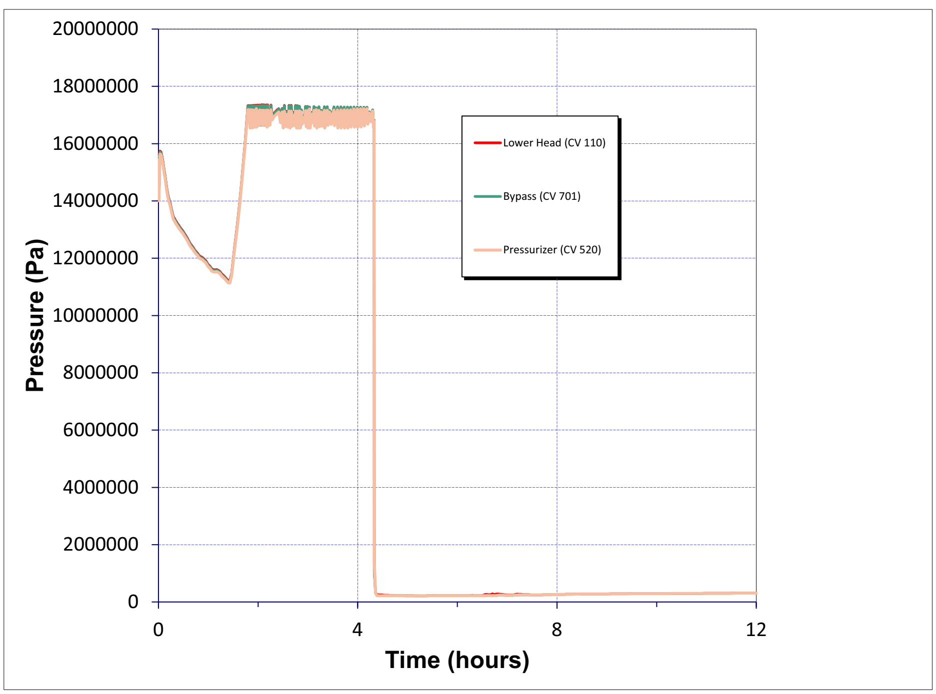



**Figure 4-11.** Primary system component control volume pressure.

* The predicted time of depressurization from hot leg failure differs from values shown in Table 4-1 and Figure 4-6 due to difference in code version and Surry plant models.

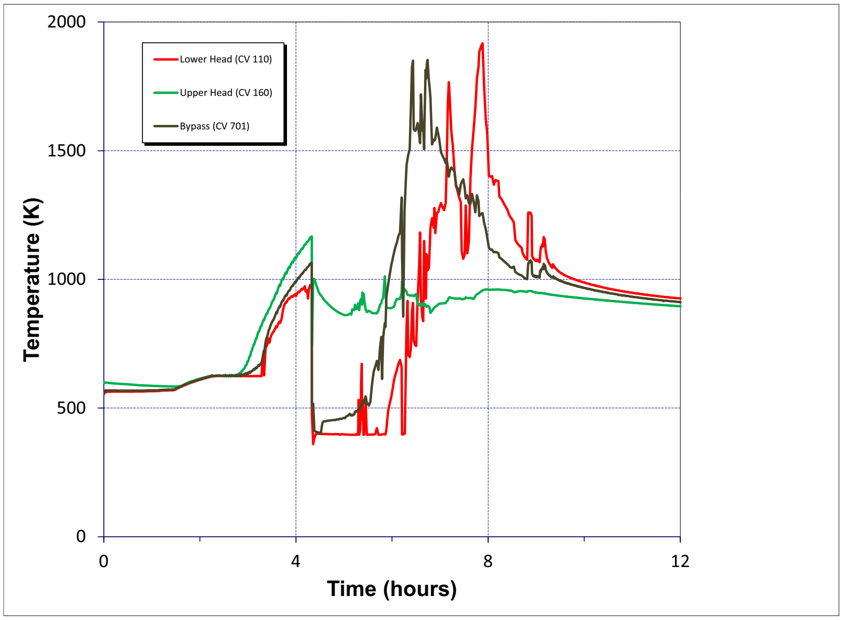



**Figure 4-12.** Primary system component control volume temperature.

Appendix C contains figures showing pressure and temperature response within the RCS for additional control volumes. Also, as noted previously, there are differences in MELCOR RCS results presented in Reference 11 and the supplemental results provided by SNL. These differences are attributed to improvements made in the MELCOR code and the Surry plant model. These changes did not significantly affect peak temperature or pressure predictions. Rather, they affected the timing at which the transient progressed. For example, a comparison of the pressure response of the pressurizer in Figures 4-6 and 4-11 shows that the earlier SOARCA results indicates that steam generator dryout (and subsequent pressure increases) occurs at 1.2 hours (about 0.5 hours earlier than indicated in the supplemental results), but peak pressures of approximately 17 MPa are predicted in both figures.

#### 4.3.2.2. Containment

As indicated in Section 3.3, instrumentation system components located within the containment are tested at EQ pressures of 0.41 MPa, temperatures of 149 °C (422 K), and gamma doses of 0.5 to 1 x$10^8$ Rads. Predicted pressures for this unmitigated STSBO exceeded EQ value of 0.41 MPa for instrumentation system components throughout the containment. As shown in Figures 4-13, containment pressure is predicted to reach EQ values at 20 hours into the sequence and remain above this value until calculations cease at 48 hours. However, SOARCA results presented in Reference 11 indicate that pressures in excess of EQ values occur at 12 hours and start to decrease after 72 hours. As noted previously, discussions with cognizant SNL researchers attributed these differences to improvements in MELCOR and Surry plant models.[78] However, SNL researchers performed an additional calculation for an extended duration to determine peak containment pressure behavior with these modeling enhancements. Results indicate that peak values and durations at peak pressures were similar to those shown in Figure 4-9.[79]

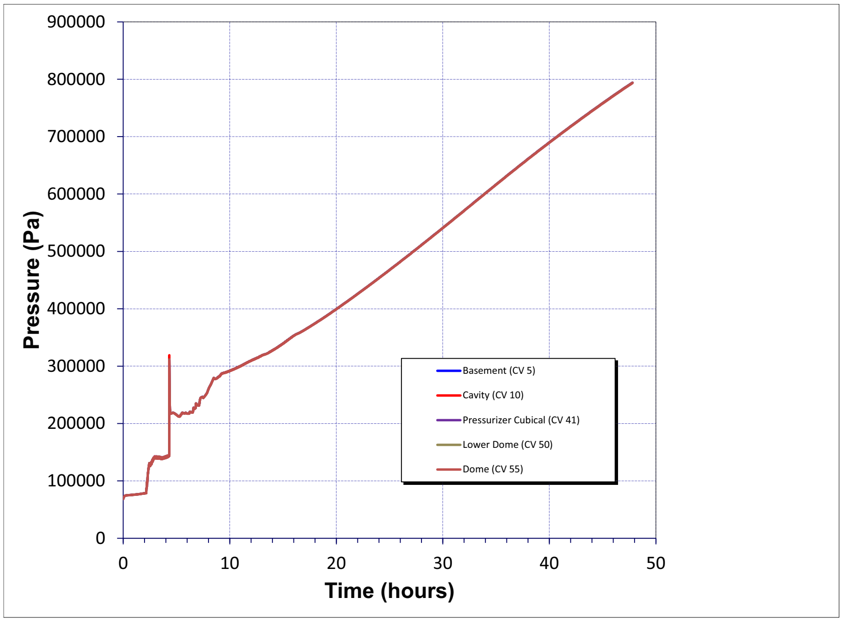



**Figure 4-13.** Containment control volume pressure.

Figure 4-14 indicates that SNL supplemental calculations predict that the reactor vessel cavity control volume (CV 10) temperature exceeds the 422 K EQ temperature approximately 5 hours into the sequence (e.g., after the time of hot leg failure) and reaches values in excess of 1900 K within 8 hours (after the time of vessel lower head failure). Results in Figures 4-14 and 4-15 show that temperatures at other locations within the containment also exceed 422 K prior to 10 hours into the sequence and remain elevated throughout the 48 hours that calculations were performed.

Doses at selected containment volumes are also predicted to exceed EQ values. For example, gamma doses within the containment basement (and at other locations) are predicted to exceed EQ values of 0.5 to 1.0 x$10^8$ Rad within the first few hours of this sequence and continue to increase throughout the 48 hours that the calculation was performed (see Figures 4-16 and 4-17).

Again, as observed in Section 3.3, it is unclear if sensors exceeding the EQ temperatures will fail, particularly if the environmental conditions exceed the EQ values by only small amounts or for short periods of time. However, as shown in Figures 4-13 through 4-16 indicate that conditions within the containment significantly exceed EQ conditions for extended time periods and that the conditions exist at times when it is critical for operators to have access to accurate sensor data.

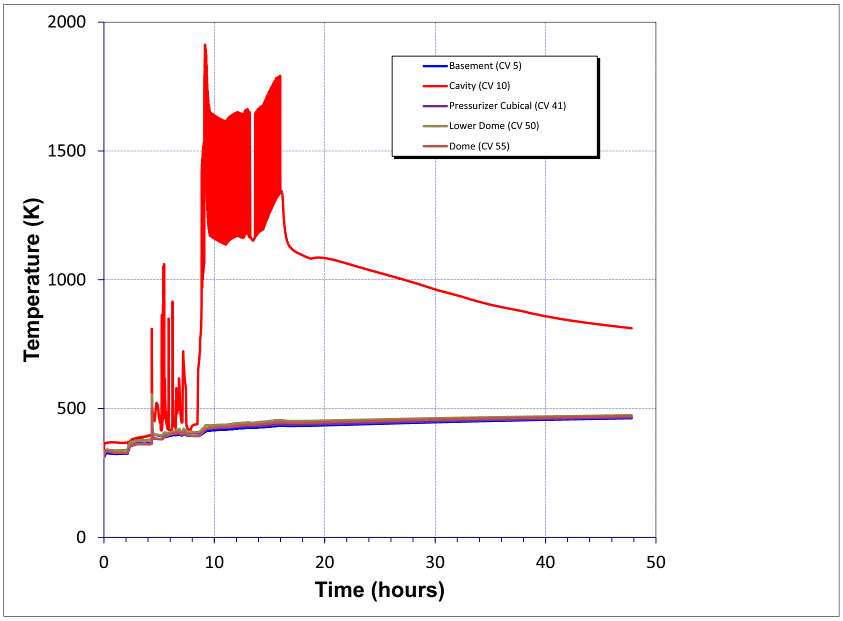



**Figure 4-14.** Containment control volume temperatures.

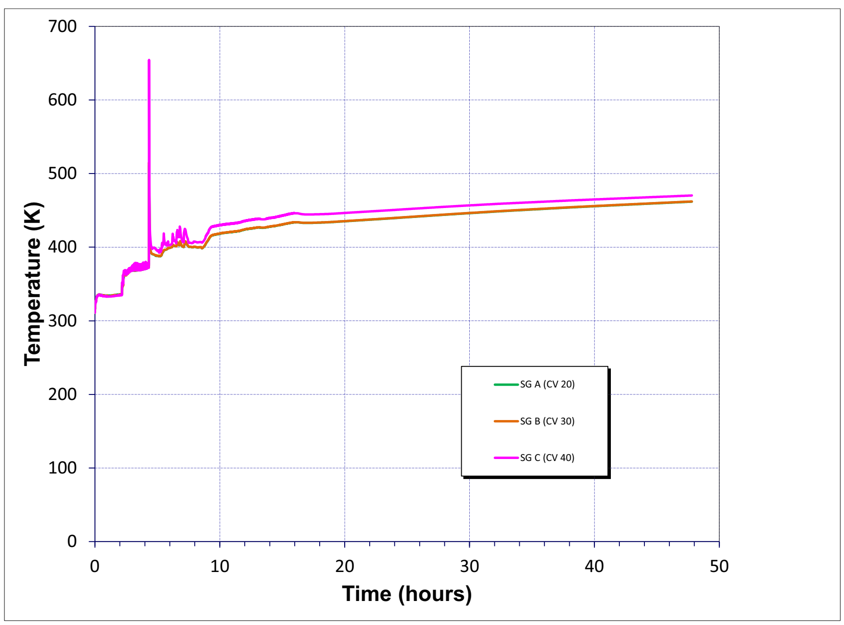



**Figure 4-15.** Containment SG cubicle control volume temperatures.

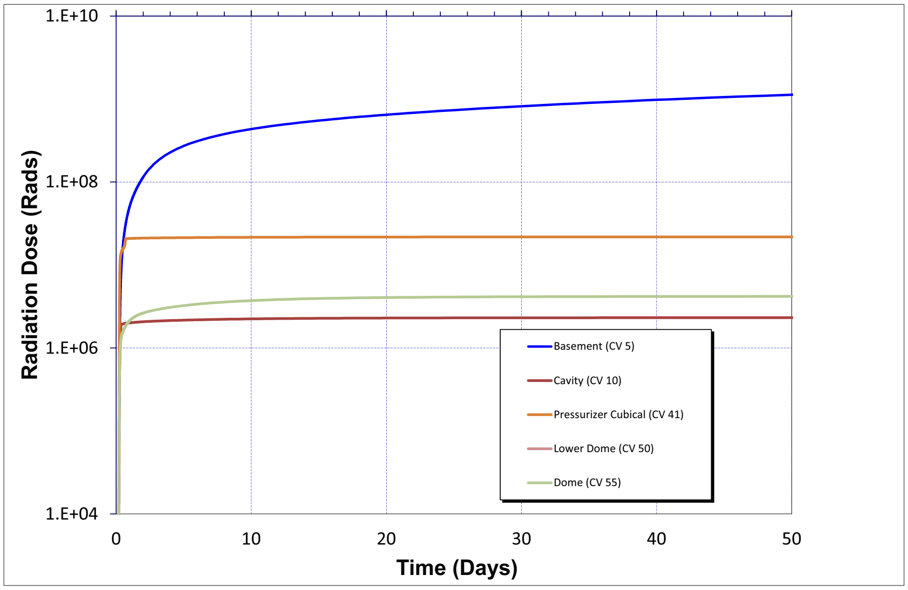



**Figure 4-16.** Containment control volume gamma dose.

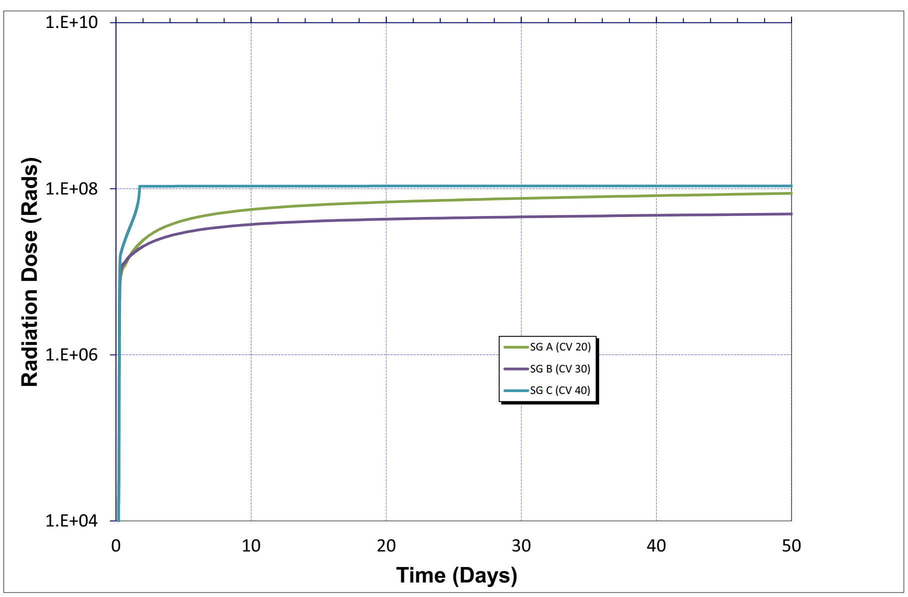



**Figure 4-17.** Containment SG control volume gamma dose.

## 4.4. Summary

This section describes the MELCOR code model for the Surry nuclear power station that was used in SOARCA program and provides results for an unmitigated STSBO sequence at this plant.

As discussed in this section, results for the unmitigated STSBO sequence indicate that some of the instrumentation system components proposed to provide critical information will be exposed to conditions beyond their EQ ranges. Pressures exceed EQ values at many locations within the RCS and containment. Temperatures were predicted to exceed EQ values at locations within the RPV lower plenum and at locations within the containment, especially at locations within the reactor cavity beneath the RPV. Radiation levels exceeded EQ values in both the RPV and the containment.

At some locations, MELCOR calculations predict that conditions significantly exceeded EQ ranges for extended time periods. For example, temperatures within the reactor cavity (see Figure 4-14) exceed the EQ value of 422 K at approximately 5 hours into the sequence and remain above this value throughout the remainder of the transient. Results show that temperatures at other locations within the containment also exceed EQ values of 422 K prior to the time of vessel failure and remain elevated throughout the remainder of the 48 hours that the accident was simulated. In some locations, peak values were predicted to be well-above EQ values. Likewise, doses within the containment are predicted to exceed EQ ranges within the first few hours of the sequence. In summary, results indicate that conditions within the RCS and the containment exceed EQ ranges at time periods when it is critical for operators to have access to accurate instrumentation data (e.g., EQ ranges are exceeded at certain locations within the RCS and the containment prior to the time when vessel failure is predicted).

However, as noted in Section 3.3, instrumentation exposed to conditions in excess of their EQ doses do not necessarily fail. Additional considerations related to instrumentation survivability and the ability of plant instrumentation to provide critical information to operators during this unmitigated STSBO are discussed in Section 5.

# 5. INSTRUMENTATION AVAILABILITY ASSESSMENT

Section 3 of this report identifies instrumentation that could provide critical information during a severe accident and EQ ranges for this instrumentation. Section 4 reports environmental conditions predicted during an STSBO for the Surry plant. This section compares EQ ranges for this critical instrumentation with predicted environmental conditions to provide insights about instrumentation availability, the final step in the selected approach outlined in Section 3.

Table 5-1 summarizes the environmental conditions that were predicted to occur in an unmitigated STSBO at Surry plant locations of interest for key instrumentation identified in this study. Peak or maximum EQ values, which were estimated in Section 3 using information from prior studies and authoritative PWR instrumentation references, are also listed for this instrumentation in Table 5-1.

Clearly, results indicate that instrumentation system components at locations within the RCS and containment are exposed to conditions beyond their EQ values for this accident sequence. Peak pressures exceed EQ values at many RCS and containment locations. Temperatures were predicted to exceed EQ values at locations within the RPV lower plenum and at locations within the containment, especially at locations within the reactor cavity beneath the RPV. Radiation levels exceeded EQ values in both the RPV and the containment.

As discussed within this report, information needs and instrumentation availability differ during the phases of a severe accident. However, MELCOR results shown in Section indicate that EQ values were significantly exceeded at some locations at times when it is critical for operators to have access to accurate sensor data. For example, temperatures at locations within the RCS start peaking at around 6 hours, which is before the time when vessel lower head failure is predicted. Likewise, temperatures within the reactor cavity exceed EQ values at approximately 5 hours into the sequence (e.g., after hot leg failure) and remain high throughout the 48 hours that the transient was simulated. Results indicate that peak values were well-above EQ values at some locations. As indicated in Table 5-1, peak temperatures were over 1000 °C higher than EQ values and pressures were nearly 0.4 MPa higher than EQ values.

Some of the critical information needs identified by the PWROG in Section 3 could be met by instrumentation systems located in the Auxiliary Building. However, as discussed in Section 4.1, MELCOR calculations for the STSBO sequence did not evaluate conditions in the Auxiliary Building. Containment failures were predicted at a location that allowed all fission products to be released directly to the environment. In addition, heating in the Auxiliary Building due to electrical loads required to run instrumentation would not be significant in this STSBO because all AC and DC power is lost. Hence, in this STSBO sequence, instrumentation system components located in this building should remain operational for obtaining data to meet critical information needs for parameters, such as RCS injection flow, RWST level, and containment hydrogen monitoring.

**Table 5-1.** Comparison of predicted STSBO environmental conditions with EQ Values

| Instrumentation[a] | Location[b] | Predicted Peak or Maximum Value[c] | | | EQ Values[d] |
|---|---|---|---|---|---|
| | | T, °C | P, MPa | Dose, Rad | |
| Steam Generator | | | | | |
| Wide Range SG Level | CV 20, CV 30, or CV 40 | 381 | 0.80 | 1.9E+08 | Maximum Temperature - 149 °C |
| SG secondary pressure | CV 20, CV 30, or CV 40 | 381 | 0.80 | 1.9E+08 | Maximum Pressure - 0.41 MPa |
| *Narrow Range SG Level* | *CV 20, CV 30, or CV 40* | 381 | *0.80* | *1.9E+08* | Dose = 1 x $10^8$ rad |
| *SG main steamline pressure* | *CV 50* | *283* | *0.80* | *5.5E+08* | |
| RCS | | | | | |
| RCS Locations | | | | | |
| Wide range RCS pressure | CV 240 | 749 | 17.3 | 3.4E+07 | Maximum Temperature - 1260°C |
| *Accumulator pressure* | *CV 240* | *< 749[e]* | *17.3* | *3.4E+07* | |
| *Pressurizer pressure* | *CV 500* | *705* | *17.3* | *2.2E+08* | Maximum Pressure - 17.2 MPa |
| *Cold leg RTD* | *CV 237* | *776* | *17.3* | *1.2E+07* | Dose = 1 x $10^8$ rad |
| *Hot Leg RTD* | *CV 401* | *897* | *17.3* | *8.0E+08* | |
| *RVLIS[f]* | *CV 110* | *1643* | *17.4* | *3.1E+08* | |
| *Power Range Monitors[g]* | *CV 110* | *1643* | *17.4* | *3.1E+08* | |
| *Source Range Monitors[h]* | *CV 110* | *1643* | *17.4* | *3.1E+08* | |
| *CETCs[h]* | *CV 110* | *1643* | *17.4* | *3.1E+08* | |
| Containment Locations | | | | | |
| Wide range RCS pressure | *CV 50* | *283* | *0.80* | *5.5E+08* | Maximum Temperature - 149 °C |
| *Accumulator pressure* | *CV 50* | *283* | *0.80* | *5.5E+08* | Maximum Pressure - 0.41 MPa |
| *Pressurizer pressure* | *CV 41* | *267* | *0.80* | *2.2E+07* | Dose = 1 x $10^8$ rad |
| *Cold leg RTD* | *CV 50* | *283* | *0.80* | *5.5E+08* | |
| *Hot Leg RTD* | *CV 50* | *283* | *0.80* | *5.5E+08* | |
| *RVLIS[f]* | *CV 10 or CV 701* | *1640* | *0.80* | *2.3E+06* | |
| *Power Range Monitors[h]* | *CV10* | *1640* | *0.80* | *2.3E+06* | |
| *Source Range Monitors[g]* | *CV10* | *1640* | *0.80* | *2.3E+06* | |
| RCS Injection Flow | | | | | |
| *LPSI pump flow rate* | *Auxiliary Building* | *NA* | *NA* | *NA* | *Maximum Temperature - 38 °C* |
| *LPSI pump discharge pressure* | | *NA* | *NA* | *NA* | *Maximum Pressure - 0.1 MPa* |
| *Charging pump injection rate* | | *NA* | *NA* | *NA* | *Dose = << $10^8$ rad* |
| *Charging pump discharge pressure* | | *NA* | *NA* | *NA* | |
| Containment | | | | | |
| Containment Sump Level (wide range monitor /*narrow range monitor)* | CV 5, CV 20, CV 30, CV 40, or CV 50 | 381 | 0.80 | 3.4E+09 | Maximum Temperature - 149 °C<br>Maximum Pressure - 0.41 MPa<br>Dose = 0.5 to 1 x $10^8$ rad |
| *Containment Temperature* | *CV 50 or CV 55* | *283* | *0.80* | *5.5E+08* | |
| Containment Hydrogen Monitor | CV 50 or CV55 | 283 | 0.80 | 5.5E+08 | |
| Containment Pressure | CV55[i] | 227 | 0.80 | 4.3E+06 | |
| Containment Pressure (bypass sequences) | *Auxiliary Building* | *NA* | *NA* | *NA* | Maximum Temperature - 38 °C<br>Maximum Pressure - 0.1 MPa |
| *Sampling for Hydrogen* | | *NA* | *NA* | *NA* | Dose = << $10^8$ rad |
| *RWST Level* | *Outdoors* | *NA* | *NA* | *NA* | |

a. Non italic text designates instrumentation proposed by PWROG in References 4 and 53; Italics indicate typical plant specific and alternate indications suggested by R. Lutz, Lutz-NSC, in email dated July 16, 2015,[74] that should be generally applicable to Surry.

b. Based on plant-specific information where possible. When detailed plant-specific location information not available, based on expert opinion related to typical PWR configurations.[74]

c. Peak or maximum predicted values for candidate locations.

d. Based on Reference 34 and other relevant sources (see Section 3.3).

e. Accumulators have check valves that prevent inflow of RCS fluids. Hence, sensors would be exposed to much lower temperatures than 749 K.[74]

f. Available information suggests multiple penetrations, including one through an in-core instrumentation tube. Conditions conservatively assumed at this location.

g. Detailed location information not available for Surry. Some PWRs rely solely on ex-core detectors, but other plants use detectors within in-core instrumentation tubes. Assumed location provides conservative peak temperature estimates.

h. Location assumes instrumentation attached to traveling in-core probe.

i. Sensor assumed to be located in upper dome.

In summary, results indicate that some instrumentation identified to provide critical information would be exposed to conditions that significantly exceed EQ values for extended time periods for the low frequency STSBO sequence evaluated in this study. It is recognized that the CDF of the selected sequence may be considerably lower if evaluations considered the new FLEX equipment that is being implemented by industry. Furthermore, it is not clear that degradation of instrumentation systems exposed to conditions that exceed their EQ values would preclude the success of new SAMGs being proposed by industry. Nevertheless, because of uncertainties in instrumentation response when exposed to conditions beyond EQ values and alternate challenges associated with different sequences that may impact sensor performance, it is recommended that additional evaluations of instrumentation performance be completed to provide confidence that operators have access to accurate, relevant, and timely information on the status of reactor systems for a broad range of challenges associated with risk important severe accident sequences. Specific activities for future evaluations are identified in Section 6.

# 6. SUMMARY AND RECOMMENDATIONS

As noted in Section 1, DOE-NE initiated a scoping evaluation to assess critical information needs identified for severe accident management and mitigation in commercial LWRs, to quantify the environment that instruments monitoring this data would have to survive, and to identify gaps where predicted environments exceed instrumentation qualification levels. Results from the PWR scoping evaluation for an unmitigated STSBO are summarized in this section. In addition, this section summarizes limitations of the current evaluation and presents recommendations for future activities.

## 6.1. PWR Scoping Evaluation Approach and Results

As discussed within Section 2, instrumentation survivability evaluations were completed in the early 1990s. In the last two decades, advances have been made to improve our understanding and modeling of severe accidents and to improve the manner in which plant staff responds to such events. Hence, updated evaluations were needed that use state-of-the-art systems analysis codes and that consider mitigating strategies currently proposed by industry. Nevertheless, the scoping evaluation documented in this report was informed by prior instrumentation survivability studies, as well as on-going US and international efforts to assess the adequacy of nuclear power plant instrumentation during severe accidents. For example, the selected approach (see Figure 6-1) drew most heavily from the methods developed and deployed by NRC and US industry. In addition, the current study relied heavily on industry efforts to identify critical information needs during a severe accident and instrumentation that can provide data to meet these needs.

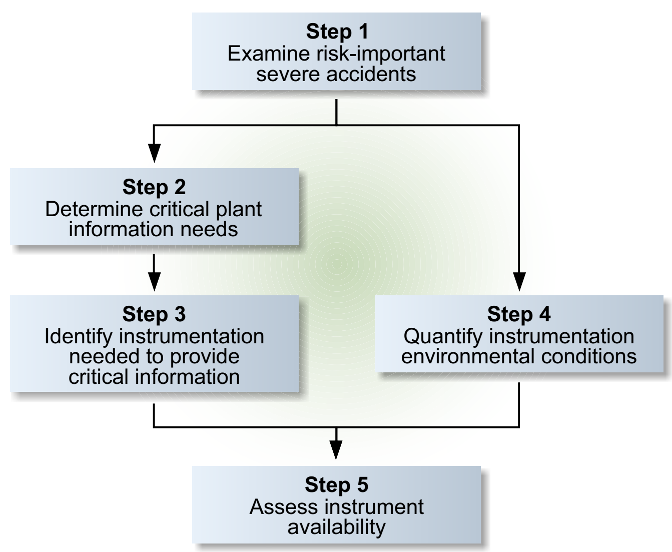



**Figure 6-1.** Approach adopted for current PWR instrumentation survivability study.

Section 3 of this report identifies instrumentation that could provide critical information during a severe accident and EQ ranges for this critical instrumentation. Environmental conditions during this unmitigated STSBO were quantified by SNL using the current version of the MELCOR code and an updated Surry plant model developed for the NRC SOARCA evaluation; results are reported in Section 4. Section 5 of this report compares EQ ranges for this critical instrumentation with predicted environmental conditions to provide insights about instrumentation availability (the final step of the approach shown in Figure 6-1).

Clearly, results in this scoping evaluation indicate that instrumentation system components at locations within the RCS and containment are exposed to conditions beyond their EQ values for this accident sequence. Peak pressures exceed EQ values at many RCS and containment locations. Temperatures are predicted to exceed EQ values at locations within the RPV lower plenum and at locations within the containment, especially at locations within the reactor cavity beneath the RPV. Radiation levels exceed EQ values in both the RPV and the containment.

Information needs and instrumentation availability differ during the phases of a severe accident. Results shown in Section 4 indicate that EQ values were significantly exceeded at some locations at times when it is critical for operators to have access to accurate sensor data. For example, temperatures at locations within the RCS start peaking at around 6 hours, which is before the time when vessel lower head failure is predicted. Likewise, temperatures within the reactor cavity exceed EQ values at approximately 5 hours into the sequence (e.g., after hot leg failure) and remain high throughout the 48 hours that the transient was simulated. Results indicate that peak values were well-above EQ values at some locations. As indicated in Section 4, peak temperatures were over 1000 °C higher than EQ values and pressures were nearly 0.4 MPa higher than EQ values.

## 6.2. Scoping Evaluation Limitations

There are several limitations and considerations associated with this scoping evaluation. These points, which are elaborated upon within this document, are summarized here:

- This evaluation only considered an unmitigated STSBO sequence. This sequence is estimated to have a very low frequency of occurrence, and its frequency may be further reduced at some plants if industry efforts to implement FLEX are considered.* Other sequences evaluated in SOARCA, such as SGTR or ISLOCA bypass events or more slowly progressing events with hydrogen burns, such as the LTSBO, would present different challenges to plant instrumentation.
- Instrumentation location, which is very plant-specific, was not provided by plant personnel. More detailed plant-specific information about instrumentation component location may impact conclusions.
- This study primarily focused on instrumentation sensors, rather than the entire instrument systems (e.g., the transducers, cabling, electronics, and other components). Although other instrumentation components could fail, more detailed plant-specific location and system design information is needed to consider predict such failures.
- Plant-specific EQ values for instrumentation were estimated based on values assumed in earlier Surry evaluations. In some cases, predicted conditions significantly exceeded estimated EQ values (and it is not expected that more precise information would change conclusions of this study). However, in other cases, sensor operating ranges exceed EQ values by smaller amounts, so it is anticipated that these sensors could survive above the estimated EQ values (but more precise EQ information would be useful).

* It would be difficult to install FLEX equipment in time to prevent SG dryout within one hour. Hence, the ability to use FLEX equipment to preclude this event will depend on the ability of RCS pump seals to resist degradation and limit leakage.

- The effects of humidity on instrumentation performance were not considered in the present study. Prior studies indicate that instrumentation was qualified for 100% humidity conditions. Although it is recognized that instrumentation systems may not remain leak-tight during service, it is beyond the scope of this evaluation to predict such failures.
- It is unclear that instrumentation would survive the earthquake that initiated this event. In fact, Reference 3 assumes that no instrumentation is available.
- MELCOR results were only provided for evaluating instrumentation environmental conditions within the RCS and containment. The predicted containment failure location precluded any releases from entering the Auxiliary Building for this unmitigated STSBO sequence, and instrumentation located within this building are not expected to be exposed to any significant changes in environmental conditions in this STSBO.
- The potential for hydrogen combustion to affect instrumentation performance was not considered because Reference 3 predicted that concentrations of combustible gases and steam within the containment preclude any concerns about hydrogen burns during this sequence, However, challenges in other sequences may affect the ability of instrumentation to accurately measure combustible gas concentrations; and if combustions occur, such challenges may significantly impact instrument performance.
- There are discrepancies between the MELCOR results documented in the Reference 11 SOARCA report and the supplemental MELCOR results provided by SNL for this scoping evaluation. It was concluded that these differences were due to improvements in the MELCOR code and the Surry model used for the scoping evaluation and that results primarily affected timing rather than peak conditions within the RCS and containment. Hence, it is expected that these differences do not impact conclusions from this scoping evaluation.
- There are limitations associated with the methods used by SNL to estimate dose. Although the MELCOR code tracks time-dependent fission product inventory in various control volumes, the code does not calculate radiation dose. However, SNL staff developed an algorithm for estimating the time-dependent radiation dose in a node based on the fission product inventory in the node.

Nevertheless, this scoping study represents a first effort for assessing the viability of the approach depicted in Figure 6-1 and for determining if more detailed evaluations are needed to assess the survivability of such sensors when they are exposed to conditions outside their EQ values. Results indicate that some instrumentation identified to provide critical information would be exposed to conditions that significantly exceed EQ values for extended time periods for the low frequency STSBO sequence evaluated in this study. It is recognized that the CDF of the selected sequence may be considerably lower if evaluations considered the new FLEX equipment that is being implemented by industry. Furthermore, it is not clear that degradation of instrumentation systems exposed to conditions that exceed their EQ values would preclude the success of new SAMGs being proposed by industry. Nevertheless, because of uncertainties in instrumentation response when exposed to conditions beyond EQ values and alternate challenges associated with different sequences that may impact sensor performance, it is recommended that additional evaluations of instrumentation performance be completed. These activities are identified in Section 6.3.

## 6.3. Recommendations for Future Evaluations

As discussed above, it is recommended that additional evaluations of instrumentation performance be completed to provide confidence that operators have access to accurate, relevant, and timely information on the status of reactor systems for a broad range of challenges associated with risk important severe accident sequences. Specific activities suggested for future evaluations include:

- *Additional scenario evaluations to quantify conditions associated with a broader range of challenges.* The initial scoping study documented in this report was limited to one scenario, an unmitigated STSBO sequence. Because MELCOR calculations did not predict conditions with hydrogen burns, the effects of ignition were not considered in this STSBO sequence. Likewise, because containment failure was predicted to result in releases directly into the environment and because a loss of all AC and DC power was assumed, MELCOR calculations did not simulate the effects of releases in the Auxiliary Building and the heatup associated with electrical loads in rooms with instrumentation in this building. Current post-Fukushima efforts by industry include developing strategies to circulate air in key rooms in the auxiliary building to prevent any adverse impact on power supplies and/or critical instrumentation. If the current scoping evaluation were expanded, results from could be used to quantify the timing and conditions that challenge instrumentation in the Auxiliary Building. As discussed within this document, the SOARCA program did consider a broad range of events for the Surry plant.
- *Plant-specific information related to the types and locations of components in instrumentation systems.* The EQ conditions and locations of instrumentation assumed in this document were primarily based on information in earlier plant-specific references and typical values for instrumentation systems used in other PWR plants. It was not possible to determine plant-specific instrumentation EQ ranges, to identify the actual locations of instrumentation system components, or assess if components of the instrument systems are sufficiently protected to withstand conditions expected during risk important events.
- *Insights from post-accident examinations at the affected reactors at Daiichi.* Similar to insights gained from post-accident examinations at TMI-2, it is expected that evaluations of instrumentation at the affected reactors at Daiichi will provide critical insights related to instrumentation survivability and the proposed use of trending information from sensors exposed to conditions outside their EQ ranges.
- *Test data to quantify instrumentation system performance when exposed to conditions beyond their EQ ranges*. MELCOR results for the STSBO sequence indicate that there are substantial time periods when instrumentation system components are exposed to conditions beyond their EQ ranges. Testing could provide confidence in the ability of sensors to survive such conditions for the required time durations. In cases where instrumentation is expected to decalibrate, testing could provide confidence that data from instrumentation will yield the expected trends that are relied upon in updated SAMGs.

As discussed within this report, on-going industry efforts may further limit the number of sensors that are deemed critical for operator actions during severe accidents. With effective coordination of the above tasks with industry efforts, it should be possible to obtained a well-defined set of test conditions (e.g., pressures, temperatures, flux levels, etc. for appropriate durations) for evaluating the performance of a limited number of sensor components. Results from such evaluations could provide additional confidence in updated SAMGs being implemented by industry.

# 7. REFERENCES

1. US Nuclear Regulatory Commission, "Recommendations for Enhancing Reactor Safety in the 21st Century," The Near-Term Task Force Review of Insight from the Fukushima Daiichi Accident," July 12, 2011. (ML111861807).

2. J. Rempe, M. Farmer, M. Corradini, L. Ott, R. Gauntt, and D. Powers, "Revisiting Insights from Three Mile Island Unit 2 Post-Accident Examinations and Evaluations in View of the Fukushima Daiichi Accident," *Nuclear Science and Engineering*, **172**, November 2012, pp 223-248.

3. J.L. Rempe and D. L. Knudson, "Instrumentation Performance during the TMI-2 Accident," *IEEE Transactions on Nuclear Science,* Vol. 61, Issue 4, pp 1963-1970, Jan 2014.

4. R. Lutz, R. Linthicum, N. Labarge, S. Pierson and J. Ford., "PWROG Severe Accident Management Technical Support Guidance for Instrumentation," *9th International Conference on Nuclear Plant Instrumentation, Control & Human-Machine Interface Technologies (NPIC & HMIT 2015),* Charlotte, North Carolina, Feb 21-26, 2015.

5. B. Williamson and P. Ellison, "Instrument Challenges Observed at Fukushima Daiichi from an Operations Point of View," *9th International Conference on Nuclear Plant Instrumentation, Control & Human-Machine Interface Technologies (NPIC & HMIT 2015),* Charlotte, North Carolina, Feb 21-26, 2015.

6. N. P. Neureiter (Chair), et al., Committee on Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Plants, National Research Council of the National Academies, Lessons Learned from the Fukushima Nuclear Accident for Improving Safety of U.S. Nuclear Plants, The National Academies Press, Washington, D.C., USA (2014).

7. J. Niel (Chair), et al., NEA Committee on Nuclear Regulatory Activities (CNRA), B. Sheron (Chair), et al., NEA Committee on the Safety of Nuclear Installations (CNSI), and A. McGarry (Chair) et al., NEA Committee on Radiation Protection and Public Health (CRPPH), The Fukushima Daiichi Nuclear Power Plant Accident: OECD/NEA Nuclear Safety Response and Lessons Learnt, NEA No. 7161, Organisation for Economic Co-operation and Development, Paris, France (2013).

8. International Atomic Energy Agency, A*ccident Monitoring Systems for Nuclear Power Plants,* IAEA Nuclear Energy Series Report No. NP-T-3.16, 2015.

9. US Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," NUREG-1935, Washington, DC, November 2012.

10. US Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis," NUREG/CR-7110, Volume 2, Rev. 1, Washington, DC, January 2012.

11. US Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis," NUREG/CR-7110, Volume 2, Rev. 1,Washington, DC, August 2013.

12. D. Clayton and M. Poore, "Post-severe Accident Environmental Conditions for Essential Instrumentation For Boiling Water Reactors," ORNL/TM-2015/278, June 2015.

13. Dominion, "Surry Power Station," https://www.dom.com/corporate/what-we-do/electricity/generation/nuclear/surry-power-station, last accessed July 23, 2015.

14. D. C. Aldrich, J. L. Sprung, D. J. Alpert, K. Diegert, R. M. Ostmeyer, L. T. Ritchie, D. R. Strip, J. D. Johnson, K. Hansen, and J. Robinson, "Technical Guidance for Siting Criteria Development," NUREG/CR-2239, December 1982.

15. US Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five Nuclear Power Plants, NUREG-1150, Vol. 1 and 2, December 1990.

16. US Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, October 1975.

17. Office of the Federal Register, 10CFR50, "Domestic Licensing of Production and Utilization Facilities," http://www.ecfr.gov, updated July 28, 2015.

18. R. O. Gauntt, J. E. Cash, R. K. Cole, C. M. Erickson, L. L. Humphries, S. B. Rodriguez, M. F. Young, "MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide, Version 1.8.6," NUREG/CR 6119, Vol. 1, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, DC, 2005.

19. D. Chanin and M. L. Young, "Code Manual for MACCS2 - User's Guide" NUREG/CR - 6613, May1998.

20. Sandia National Laboratories, "MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide, Version 2.1", NUREG/CR-6119, Rev. 3179 (Draft), September 2011.

21. D. L. Rahn, "Criteria for Enhancing Accident Monitoring Instrumentation to Address Lessons Learned from Fukushima," *9th International Conference on Nuclear Plant Instrumentation, Control & Human-Machine Interface Technologies (NPIC & HMIT 2015)*, Charlotte, North Carolina, Feb 21-26, 2015.

22. Office of the Federal Register, 10CFR50 Appendix A, "General Design Criteria for Nuclear Power Plants", http://www.ecfr.gov, updated July 28, 2015.

23. US Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, Volumes 1 and 2, May 1980.

24. US Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980 and Supplement 1, January 1983.

25. Office of the Federal Register, 10CFR34(f), Additional TMI-related Requirements, http://www.ecfr.gov, updated July 28, 2015.

26. Office of the Federal Register, 10CFR52, "Licenses, Certification, and Approvals for Nuclear Power Plants", http://www.ecfr.gov, updated July 28, 2015.

27. US Nuclear Regulatory Commission, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and Following an Accident," Regulatory Guide 1.97, Revision 3, 1983.

28. US Nuclear Regulatory Commission, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Regulatory Guide 1.97, Revision 4, 2006.

29. J. A. Gieseke et al., *Radionuclide Release Under Specific LWR Accident Conditions - PWR Large Dry Containment Design (Surry Plant Recalculations),* BMI-2104, Vol. V, July 1984.

30. J. A. Gieseke et al., *Radionuclide Release Under Specific LWR Accident Conditions - PWR Large Dry Containment Design (Zion Plant),* BMI-2104, Vol. VI, July 1984.

31. R. S. Denning et al., *Radionuclide Release Calculations for Selected Severe Accident Scenarios - PWR, Subatmospheric Containment Design*, NUREG/CR-4624, Vol. 3, July 1986.

32. R. S. Denning et al., Radionuclide Release Calculations for Selected Severe Accident Scenarios - PWR, Large Dry Containment Design, NUREG/CR-4624 Vol. 5, July 1986.

33. D. J. Hanson, L. W. Ward, W. R. Nelson, and O.R. Meyer, A*ccident Management Information Needs Methodology Development and Application to a Pressurized Water Reactor (PWR) with a Large, Dry Containment*, NUREG/CR-5513, Vols. 1 and 2, April 1990.

34. W. C. Arcieri and D. J. Hanson, *Instrumentation Availability for a Pressurized Water Reactor With a Large Dry Containment During Severe Accidents*, NUREG/CR-5691, EGG-2638, March 1991.

35. D. J. Hanson, W. C. Arcieri, and L. W. Ward, "Accident Information Needs," *Transactions of the Workshop on Computerized Accident Management Support,* Halden, Norway, November 19-20, 1992.

36. D. B. King, V. F. Nicolette, and V.J. Dandini, *Safety Related Equipment Survival in Hydrogen Burns in Large Dry PWR Containment Buildings*, NUREG/CR-4763, March 1988.

37. J. A. Achenbach, R. B. Miller, V. Srinivas, *Large-Scale Hydrogen Burn Equipment Experiments,* Electric Power Research Institute, NP-4354, December 1985.

38. US Nuclear Regulatory Commission, Tier 3 Program Plans and 6-month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami, SECY-12-0095, July 13, 2012.

39. US Nuclear Regulatory Commission, "Issuance of Order To Modify Licenses With Regard To Requirements For Mitigation Strategies For Beyond-design-basis External Events," EA-12-049, March 12, 2012.

40. US Nuclear Regulatory Commission, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," EA-12-051, March 12, 2012.

41. US Nuclear Regulatory Commission, "Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," EA-13-109, June 6, 2013.

42. US Nuclear Regulatory Commission, "Fifth 6-month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," SECY-14-0046, April 17, 2014.

43. Electric Power Research Institute, "Severe Accident Management Guidance Technical Basis Report Volume 1: Candidate High-Level Actions and Their Effects and Volume 2: The Physics of Accident Progression," EPRI TR-101869, Palo Alto, CA, December 1992.

44. Electric Power Research Institute, "Severe Accident Management Guidance Technical Basis Report, Volume 1: Candidate High- Level Actions and Their Effects" EPRI, Palo Alto, CA: 2012. 1025295.

45. Electric Power Research Institute, "Instrument Performance under Severe Accident Conditions - Ways to Acquire Information from Instrumentation affected by an Accident," EPRI TR-102371, November 1993.

46. Electric Power Research Institute, "Assessment of Existing Plant Instrumentation for Severe Accident Management," EPRI TR-103412, December 1993.

47. Fauske and Associates, LLC, "Transmittal Document for MAAP5 Code Revision MAAP 5.02," FAI/13-0801 (2013).

48. Nuclear Energy Institute, "Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," NEI 12-02, Rev. 1, May 2012.

49. Nuclear Energy Institute, "Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents," NEI-14-01, Rev. 0, April 2014.

50. Nuclear Energy Institute, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" NEI-12-06, Rev. 1, April 2015.

51. PWR Owner's Group, "Loss of Vital Instrumentation or Control Power," Rev. 1, FSG-7, December 31, 2014.

52. N. J. Stringfellow, Chairman, PWR Owner's Group, "Generic Severe Accident Guidance," *Testimony at Commission Briefing on Mitigation of Beyond Design Basis Events Rulemaking,* July 9, 2015, http://video.nrc.gov/vPlayer.php?eventID=1052&type=JW, last accessed August 22, 2015.

53. N. R. Labarge, R. J. Lutz, et. al., "Insights from Development of the Combined PWR SAMG,"*ANS PSA 2013 International Topical Meeting on Probabilistic Safety Assessment and Analysis*, Columbia, SC, September 22-26, 2013, American Nuclear Society, LaGrange Park, IL 2013.

54. Electric Power Research Institute, "Technical Advisory Group I&C for Beyond Design Basis Events & Severe Accidents (BDBE &SA) Charter, January 17, 2014.

55. Electric Power Research Institute, "Instrumentation & Control for Beyond Design Basis Events and Severe Accidents Technical Advisory Group Conference Call," January 22, 2014.

56. M. L. Corradini (editor), D. Peko, M. Farmer, J. Rempe, J. Coleman, D. Knudson, J. O'Brien, D. Clayton, M. Francis, K.Robb, R. Gauntt, and D. Osborn, *Reactor Safety Technologies Pathway Technical Program Plan,* INL/EXT-15-35976, June 2015.

57. M. T. Farmer (editor), R. Bunt, M. Corradini, P. Ellison, M. Farmer, M. Francis, J. Gabor, R. Gauntt, C. Henry, R. Linthicum, W. Luangdilok, R. Lutz, C. Paik, M. Plys, C. Rabiti, J. Rempe, K. Robb, R. Wachowiak, *Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis,* ANL/NE-15/4, March 2015.

58. J. L. Rempe (editor), P. Amway, R. Bunt, M. Corradini, P. Ellison, M. Farmer, M. Francis, J. Gabor, R. Gauntt, C. Henry, D. Kalinich, S. Kraft, R. Linthicum, W. Luangdilok, R. Lutz, D. Luxat, C. Paik, M. Plys, C. Rabiti, J. Rempe, K. Robb, R. Wachowiak, and B. Williamson, *US Efforts in Support of Examinations at Fukushima Daiichi,* ANL/NE-15/2, August 2015.

59. Civil Nuclear Energy and Research Working Group, "The present state and the vision for the R&D of SA-Keisou (Severe Accident - Instrumentations & Monitoring Systems) in Japan," April 2014.

60. SA Keisou R&D Working Team (Hitachi-GE, Toshiba, and Mitsubishi), "Status of R&D program for Advanced Instrumentation System for Severe Accidents in Nuclear Power Plants," in Japan" presentation at the CNWG, January 27-29, 2015, Argonne, IL.

61. American Nuclear Society, ANS Std 4.5-1980, Criteria for Accident Monitoring Functions in Light-Water- Cooled Reactors.

62. Institute of Electrical and Electronics Engineers, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," Standard 497-2002, Reaffirmed June 12, 2008.

63. J. Naser, EPRI, email to Technical Advisory Group (TAG), "Summery of Quarterly Conference Call," July 30, 2015.

64. International Electrotechnical Commission 61226 (2009) Nuclear power plants – Instrumentation and control important to safety – Classification of instrumentation and control functions.

65. International Electrotechnical Commission, 60064, (2011), Nuclear power plants - Control Rooms - Design, IEC Standard 60964.

66. International Electrotechnical Commission 60780 Nuclear power plants – Electrical equipment of the safety system – Qualification

67. International Electrotechnical Commission 62003 Nuclear Power Plants – Instrumentation and control systems important to safety – Requirements for electromagnetic compatibility testing

68. International Electrotechnical Commission 60980 Recommended practices for seismic qualification of electrical equipment of the safety system for nuclear generating stations.

69. Kerntechnischer Ausschuss Safety Standard 3502 (11/2012), Accident Measuring Systems.

70. STUK, Regulatory Guides on Nuclear Safety (YVL), YVL 1.0, Safety criteria for design of nuclear power plants," January 1996.

71. J.R. Field, Jr., Dominion, letter to J.L. Rempe, INL, “Transmittal of Surry EOPs and SAMGs,” dated April 4, 2014.

72. J. D. Neighbors, Project Manger, US NRC, letter to W. L. Steward, Virgina Electric and Power Company, “Amendment No 99, Facility Operating License No. DPR-37 and Amendment 100 Facility Operating License No. DPR-32,” Changes to Technical Specification to add requirements related to NUREG-0737,” October 15, 1984.

73. L. N. Hartz, Virginia Electric and Power Company, letter to US NRC, “Surry Power Station Units 1 and 2 Revision of Regulatory Guide 1.97 Type A Variables”, DPR-32 and DPR-37, December 1999.

74. R. Lutz, Lutz-NSC, emails and comments provided to J. Rempe, Rempe and Associates, LLC, dated July 16 through August 21, 2015.

75. H. Hashimian, *Maintenance of Process Instrumentation in Nuclear Power Plants*, Springer Science + Business Media; Springer -Verlag Berlin Heidelberg, 2006.

76. K. Ross, SNL, email to D. Knudson, INL, “Surry STSBO TH Results.xlsx,” dated April 1, 2015.

77. K. Ross, SNL, email to D. Knudson, INL, “Surry STSBO Dose Results,” dated March 31, 2015.

78. K. Ross, SNL, email to J. Rempe, Rempe and Associates, LLC, “Response to Questions about Surry STSBO TH Results.xlsx,” dated July 24, 2015.

79. K. Ross, SNL, teleconference with J. Rempe, Rempe and Associates, LLC, July 28, 2015.

80. A. L. Camp, et al., *Light Water Reactor Hydrogen Manual,* NUREG/CR-2726, SAND92-1137, August 1983.

81. J. J. DiNunno, F. D. Anderson, R. E. Baker, and R. L. Waterfield, “Calculation of Distance Factors for Power and Test Reactor Sites,” Technical Information Document TID-14844, March 1962.

# APPENDIX A. REGULATORY CONSIDERATIONS

As noted in Section 2, this appendix provides additional information related to regulatory requirements and guidance information pertaining to plant instrumentation and current regulatory activities related to severe accident instrumentation.

## A.1. Regulatory Requirements and Guidance

Regulatory requirements and guidance differ for licensees of the operating fleet of commercial nuclear power plants versus applicants pursuing licenses for new LWR designs. This section provides additional details related to regulatory requirements and guidance information discussed in Section 2.3.

Appendix A of 10 CFR 50, "General Design Criteria for Nuclear Power Plants," includes several requirements with respect to variables and systems that must be monitored by instrumentation during an accident and what parameters must be monitored to achieve safe shutdown of the plant and maintain containment integrity. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," contains requirements for new reactor design certification and combined license applications to complete severe accident performance analyses that provide assessments of severe accident equipment needs, predicted environments, and equipment survivability. For the instrumentation system to provide information necessary to support operators in responding to severe accident events, the instrumentation components must survive severe conditions and be provided with a functional supply of power.

Appendix A, "General Design Criteria for Nuclear Power Plants" to 10 CFR Part 50,"Domestic Licensing of Production and Utilization Facilities," includes several instrumentation requirements. Criterion 13, "Instrumentation and Control," requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety. Criterion 19, "Control Room," includes a requirement that a control room be provided from which actions can be taken to maintain the nuclear power plant in a safe condition under accident conditions, including loss-of-coolant accidents, and that equipment, including the necessary instrumentation, at appropriate locations outside the control room be provided with a design capability for prompt hot shutdown of the reactor. Criterion 64, "Monitoring Radioactivity Releases," includes a requirement that means be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents.

Requirements for instrumentation can also be found in 10 CFR 50.34(f)(2). 10 CFR 50.34(f)(2)(ix)(c). These sections of 10 CFA 50 require that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after exposure to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100-percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system. 10 CFR 50.34(f)(2)(xvii) requires that licensees provide instrumentation to measure, record, and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. 10 CFR 50.34(f)(2)(xix), which is **ONLY** applicable to applicants for a construction permit after January 10, 1997, requires operating reactor licensees to provide adequate instrumentation for use in monitoring plant conditions following an accident that includes core damage. 10 CFR 50.44 (b)(4)(ii) stipulates that reliable, functional equipment must be provided for monitoring hydrogen in the containment following a significant beyond design basis accident for accident management purposes.

As noted in Section 2.3, Regulatory Guide 1.97[27,28] provides guidance for instrumentation required to comply with requirements during and following an accident. Several updates were performed to Regulatory Guide 1.97. Currently, Revision 3 of Regulatory Guide 1.97,[27] which contains a prescriptive list of the minimum number of variables to monitor in BWR and PWR plants with design and qualification criteria, remains in effect for licensees of operating reactors that continue to adhere to its guidance. However, requirements in Regulatory Guide 1.97 are for design-basis events, not severe accidents. Revision 4 of Regulatory Guide 1.97[28] was issued for licensees of new reactor plants. Rather than providing a list of instrument variables to monitor, Regulatory Guide 1.97 Revision 4 provides performance-based criteria for how the variables should be selected. Revision 4 of RG 1.97, issued June 2006, states that licensees should provide instrumentation with expanded ranges capable of surviving the accident environment (with a source term that considers a damaged core) in which it is located for the length of time its function is required. Revision 4 also endorses (with some clarifying regulatory positions) a standard issued by the Institute of Electrical and Electronics Engineers (IEEE) Standard 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations."[62] However, current regulatory guidance does not include a comprehensive evaluation of the instrumentation required for severe accident conditions. In addition, current reactor and containment instrumentation are not specifically designed to remain functional under severe accident conditions.

Regulatory Guide 1.97 Rev. 3 groups instrumentation by the function of its data (e.g. evaluating plant response, informing operator of plant or system status, etc.) and the design and qualification classification or 'Category' of the sensor. A graded approach is used, with more rigor applied to Category 1 sensors providing data important to safety and less rigor applied to Category 3 sensors. For example, Category 1 sensors typically require full qualification, redundancy, and continuous real-time display and require on-site (standby) power. These variables are used by the control room operating personnel to perform their role in the emergency plan in the evaluation, assessment, monitoring, and execution of control room functions when the other emergency response facilities are not effectively manned. A single sensor may provide data for accomplishing multiple functions, and its required operating envelope may vary depending on the function being performed. Because the current effort is focusing on the Surry plant, Table A-1 identifies the required PWR instrumentation identified in Regulatory Guide 1.97 with their maximum operating envelope. As noted in Reference 27, accuracy requirements, operating ranges, and other minors changes deemed to not affect plant safety were changed after Regulator Guide 1.97, Rev 2 was initially issued.

**Table A-1.** PWR instrumentation required in Regulatory Guide 1.97, Rev. 3.

| Variable | Maximum Required Operating Range | Function(s) |
|---|---|---|
| Neutron Flux | $10^{-6}$% to 100% full power | Reactivity Control (Category 1) |
| Control Rod Position | Full in or not full in | Reactivity Control |
| RCS Soluble Boron Concentration | 0 to 6000 ppm | Reactivity Control |
| RCS Cold Leg Water Temperature | 10 to 372 ºC (50 to 700 ºF) | Reactivity Control, Core Cooling (Category 1) |
| RCS Hot Leg Water Temperature | 10 to 372 ºC (50 to 700 ºF) | Core Cooling (Category 1) |
| RCS Pressure | 0 to 20.7 MPa (0 to 3000 psig) [25.6 MPa (4000 psig) for Combustion Engineering plants] | Core Cooling (Category 1), Maintain RCS Integrity (Category 1), RCS Intact (Category 1), Containment Intact (Category 1) |
| Core Exit Temperature | 93 to 1260 ºC (200 to 2300 ºF) | Core Cooling, Fuel Cladding Intact (Category 1) |
| Coolant Inventory | Bottom of hot leg to top of vessel | Core Cooling (Category 1) |
| Degree of Subcooling | 111 ºC (200 ºF) subcooling to 19 ºC (35 ºF) superheat | Core Cooling |
| Containment Sump Water Level | Narrow range (sump) | Maintain RCS Integrity, RCS Intact |
| | Wide Range (plant specific) | Maintain RCS Integrity (Category 1), RCS Intact (Category 1) |
| Containment Pressure | -5 psig to 3 times design pressure for concrete; 4 times design pressure for steel (-10 psig for subatmospheric containments) | Maintain RCS Integrity (Category 1), Maintain Containment Integrity (Category 1), RCS Intact (Category 1), Containment Intact (Category 1) |
| Containment Isolation Valve Position | closed-not closed | Maintain Containment Integrity (Category 1) |
| Radioactivity Concentration in Primary Coolant | 1/2 to 100 times technical specification limits | Fuel Cladding Intact (Category 1) |
| Gamma Spectrum in Primary Coolant | 10 μCi/ml to 10 Ci/ml or TID-14844[81] source term | Fuel Cladding Intact |
| Containment Area Radiation - Low Range | 1 to $10^4$ R/hr | RCS Intact |
| Containment Area Radiation - High Range | 1 to $10^7$ R/hr | Containment Radiation - Detection and Assessment of Release and Long Term Surveillance Release (Category 1) |
| Effluent Radioactivity - Noble gas effluent from condenser air removal system exhaust | $10^{-6}$ μCi/cc to $10^{-2}$ μCi/cc | RCS Intact |
| Containment Hydrogen Concentration | 0 to 10 vol% (capable of operating from -5 psig to design pressure) 0 to 30 vol% for ice-condenser-type containment | Containment Intact (Category 1) |
| Containment Effluent Radioactivity - Noble gases (from identified release points) | $10^{-6}$ μCi/cc to $10^{-2}$ μCi/cc | Containment Intact |
| Effluent Radioactivity - Noble gases (from buildings or areas where penetrations and hatches are located, e.g., secondary containment and auxiliary buildings and fuel handling buildings that are in direct contact with primary containment | $10^{-6}$ μCi/cc to $10^{3}$ μCi/cc | Containment Intact |
| RHR System Flow | 0 to 110% design flow | RHR or Decay Heat Removal |
| RHR Heat Exchanger Outlet Temperature | 4 to 177 ºC (40 to 350 ºF) | RHR or Decay Heat Removal |
| Accumulator Tank Level | 10 to 90 vol% | Safety Injection Operation |
| Accumulator Tank Pressure | 0 to 5.2 MPa (0 to 750 psig) | Safety Injection Operation |
| Accumulator Isolation Valve Position | Closed or Open | Safety Injection Operation |
| Boric Acid Charging Flow | 0 to 110% design flow | Safety Injection Operation |
| Flow in HPI System | 0 to 110% design flow | Safety Injection Operation |
| Flow in HPI System | 0 to 110% design flow | Safety Injection Operation |
| Refueling Water Storage Tank Top to bottom Level | Top to Bottom | Safety Injection Operation |
| RCS Pump Statue | Motor Current | Primary Cooling System Operation |

**Table A-1.** PWR instrumentation required in Regulatory Guide 1.97, Rev. 3.

| Variable | Maximum Required Operating Range | Function(s) |
|---|---|---|
| Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines | Closed-not closed | Primary Cooling System Operation |
| Pressurizer Level | Top to bottom | Primary Cooling System Operation (Category 1) |
| Pressurizer Heater Status | Electric current | Primary Cooling System Operation |
| Quench Tank Level | Top to Bottom | Primary Cooling System Operation |
| Quench Tank Temperature | 10 to 399 ºC (50 to 750 ºF) | Primary Cooling System Operation |
| Quench Tank Pressure | 0 to design pressure | Primary Cooling System Operation |
| SG Level | From tube sheet to separators | Secondary System Operation (Category 1) |
| SG Pressure | From atmospheric pressure to 20% above the lowest safety valve setting | Secondary System Operation |
| Safety/Relief Valve Positions or Main Steam Flow | Closed - not closed | Secondary System Operation |
| Main Feedwater Flow | 0 to 110% design flow | Secondary System Operation |
| Auxiliary or Emergency Feedwater Flow | 0 to 110% design flow | Auxiliary Feedwater or Emergency Feedwater System Operation (Category 1 for B&W plants; |
| Condensate Storage Tank Water Level | Plant specific | Auxiliary Feedwater Water (Operation (Primary water supply should be listed and identified as Category 1) |
| Containment Spray Flow | 0 to 110% design flow | Containment Cooling Systems Operation |
| Heat Removal by the Containment Fan Heat Removal System | Plant specific | Containment Cooling Systems Operation |
| Containment Atmosphere Temperature | 4 to 93 ºC (40 to 400 ºF) | Containment Cooling Systems Operation |
| Containment Sump Water Temperature | 10 to 121 ºC (50 to 250 ºF) | Containment Cooling Systems Operation |
| Makeup Flow -In | 0 to 110% design flow | Chemical and Volume Control System Operation |
| Letdown Flow- Out | 0 to 110% design flow | Chemical and Volume Control System Operation |
| Volume Control Tank | Top to bottom | Chemical and Volume Control System Operation |
| Component Cooling Water Temperature to ESF System | 4 to 204 ºC (40 to 200 ºF) | Cooling Water System Operation |
| Component Cooling Water Flow to ESF System | 0 to 110% design flow | Cooling Water System Operation |
| High-Level Radioactive Liquid Tank Level | Top to bottom | Radwaste System Operation |
| Radioactive Gas Holdup Tank Pressure | 0 to 115% design pressure | Radwaste System Capacity |
| Emergency Ventilation Damper Position | Open-closed status | Ventilation System Operation |
| Status of Standby Power and Other Energy Sources Important to Safety (electric, hydraulic, pneumatic) (voltages, currents, pressures) | Plant specific | Power Supply Operation |
| Radiation Exposure Rate (inside buildings or areas where access is required to service equipment important to safety) | $10^{-1}$ to $10^{4}$ R/hr | Containment Radiation - Detection and Assessment of Release and Long Term Surveillance |
| Containment or Purge Effluent1 | $10^{-6}$ το $10^{5}$μCi/cc<br>0 to 110% vent design flow (not needed if effluent discharges through common plant vent) | Airborne Radioactive Materials Release and Long Term Surveillance of Noble Gases and Vent Flow Rate |
| Reactor Shield Building Annulus (if in design) | $10^{-6}$ το $10^{4}$μCi/cc<br>0 to 110% vent design flow (not needed if effluent discharges through common plant vent) | Airborne Radioactive Materials Release and Long Term Surveillance of Noble Gases and Vent Flow Rate |
| Auxiliary Building (including any building containing primary system gases, e.g., waste gas decay tank) | $10^{-6}$ το $10^{3}$μCi/cc<br>0 to 110% vent design flow (not needed if effluent discharges through common plant vent) | Airborne Radioactive Materials Release and Long Term Surveillance of Noble Gases and Vent Flow Rate |
| Condenser Air Removal System Exhaust | $10^{-6}$ το $10^{5}$μCi/cc<br>0 to 110% vent design flow (not needed if effluent discharges through common plant vent) | Airborne Radioactive Materials Release and Long Term Surveillance of Noble Gases and Vent Flow Rate |

**Table A-1.** PWR instrumentation required in Regulatory Guide 1.97, Rev. 3.

| Variable | Maximum Required Operating Range | Function(s) |
|---|---|---|
| Common Plant Vent or Multipurpose Vent Discharging Any of Above Releases (if containment purge is included) | $10^{-6}$ τo $10^{3}$μCi/cc<br>0 to 110% vent design flow | Airborne Radioactive Materials Release and Long Term Surveillance of Noble Gases and Vent Flow Rate |
| Vent From Steam Generator Safety Relief Valves or Atmospheric Dump Valves | $10^{-1}$ τo $10^{3}$μCi/cc<br>(Duration of releases in seconds and mass of steam per unit time) | Airborne Radioactive Materials Release and Long Term Surveillance of Noble Gases and Vent Flow Rate |
| All Other Identified Release Points | $10^{-6}$ τo $10^{2}$ μCi/cc<br>0 to 110% vent design flow (not needed if effluent discharges through common plant vent) | Airborne Radioactive Materials Release and Long Term Surveillance of Noble Gases and Vent Flow Rate |
| All Identified Plant Release Points (except steam generator safety relief valves or atmospheric steam dump valves and condenser air removal system exhaust). Sampling with Onsite Analysis Capability | $10^{-3}$ τo $10^{2}$ μCi/cc<br>0 to 110% vent design flow | Airborne Radioactive Materials Release and Long Term Surveillance of Particulates and Halogens and Vent Flow Rate |
| Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability) | $10^{-9}$ τo $10^{-3}$ μCi/cc | Environs radiation and radioactivity release assessment and analysis |
| Plant and Environs Radiation (portable instrumentation) | $10^{-3}$ to $10^{4}$ R/hr, photons<br>$10^{-3}$ to $10^{4}$ rads/hr, beta radiations low-energy photons | Environs radiation and radioactivity release assessment and analysis |
| Plant and Environs Radioactivity (portable instrumentation) | Isotopic Analysis | Environs radiation and radioactivity release assessment and analysis |
| Wind Direction | 0 to 360º (±5º accuracy with a deflection of 10º). Starting speed less than 0.4 mps (1.0 mph). Damping ratio greater than or equal to 0.4, delay distance less than or equal to 2 meters. | Meteorology for release assessment |
| Wind Speed | 0 to 22 mps (50 mph). ±0.2 mps (0.5 mph) accuracy for speeds less than 2 mps (5 mps), 10% for speeds in excess of 2 mps (5 mph), with a starting threshold of less than 0.4 mps (1.0 mph) and a distance constant not to exceed 2 meters. | Meteorology for release assessment |
| Estimation of Atmospheric Stability | Based on vertical temperature difference from primary meteorological system, -5ºC to 10ºC (-9 °F to 18 ºF) and ± 0.15°C accuracy per 50-meter intervals (_± 0.3 ºF accuracy per 164-foot intervals) or analogous range for alternative stability estimates | Meteorology for release assessment |
| Grab Sample Gross Activity | 1 τo 10 μCi/ml | Accident Sampling Capability (Analysis Capability On Site) of Primary Coolant and Sump for release assessment |
| Grab Sample Gamma Spectrum | Isotopic Analysis | Accident Sampling Capability (Analysis Capability On Site) of Primary Coolant and Sump for release assessment |
| Grab Sample Boron Content | 0 to 6000 ppm | Accident Sampling Capability (Analysis Capability On Site) of Primary Coolant and Sump for release assessment |
| Grab Sample Chloride Content | 0 to 20 ppm | Accident Sampling Capability (Analysis Capability On Site) of Primary Coolant and Sump for release assessment |
| Grab Sample Dissolved Hydrogen or Total Gas | 0 to 2000 cc (STP)/kg | Accident Sampling Capability (Analysis Capability On Site) of Primary Coolant and Sump for release assessment |

**Table A-1.** PWR instrumentation required in Regulatory Guide 1.97, Rev. 3.

| Variable | Maximum Required Operating Range | Function(s) |
|---|---|---|
| Grab Sample Dissolved Oxygen | 0 to 20 ppm | Accident Sampling Capability (Analysis Capability On Site) of Primary Coolant and Sump for release assessment |
| Grab Sample pH | 1 to 13 | Accident Sampling Capability (Analysis Capability On Site) of Primary Coolant and Sump for release assessment |
| Grab Sample Hydrogen Content | 0 to 10 vol%<br>0 to 30 vol% (for ice condensers_ | Accident Sampling Capability (Analysis Capability On Site) of Containment Air for release assessment |
| Grab Sample Dissolved Oxygen | 0 to 10 vol% | Accident Sampling Capability (Analysis Capability On Site) of Containment Air for release assessment |
| Grab Sample Gamma Spectrum | Isotopic Analysis | Accident Sampling Capability (Analysis Capability On Site) of Containment Air for release assessment |

## A.2. Current Regulatory Efforts

This section provides additional details about current regulatory efforts related to severe accident instrumentation. Clearly, the ability of operators to understand the condition of the Fukushima Daiichi reactors, containments, and SFPs was hampered because existing instrumentation was either lacking electrical power or providing erroneous readings. A post-Fukushima action item (Identifier SECY-12-0025, Enclosure 2)[38] was established to address this concern and to evaluate the regulatory basis for requiring reactor and containment instrumentation to be enhanced to withstand severe accident conditions. This activity was prioritized as Tier 3 because it requires further staff study and depends on the outcome of other lessons-learned activities. A program plan to address this action item is detailed in SECY-12-0095.[1] This program plan outlines several steps needed to achieve a basis for a regulatory decision. The first step was to ensure that licensees are appropriately considering instrumentation needs during implementation of other post-Fukushima actions:

- NTTF Recommendations 2.3 - This recommendation involves severe storm, seismic and flood protection walkdowns to identify and address plant-specific vulnerabilities and verify the adequacy of monitoring and maintenance for protection features. Resolution of this recommendation may reveal severe hazard conditions that may inform assessments of equipment survivability for severe accident instrumentation.
- NTTF Recommendation 4.1 - This recommendation involves strengthening SBO mitigation measures. Resolution of this recommendation will improve capabilities for powering equipment supporting core cooling and SFP cooling, as well as reactor coolant system and containment integrity in extended loss of AC conditions. These capabilities will inform assessments of equipment survivability for severe accident instrumentation.
- NTTF Recommendation 8 - This recommendation involves strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs, and will reveal site response needs for condition monitoring and instrumentation, which will support identification of severe accident instrumentation.
- Order EA-12-049[39] - This order involves developing strategies to mitigate beyond-design basis external events. These strategies will address both multi-unit events and reasonable protection of equipment identified under such strategies. These capabilities will inform assessments of equipment survivability for severe accident instrumentation.

- Order EA-12-051[40] - This order involves installing enhanced SFP instrumentation to withstand beyond-design-basis external events to provide emergency responders with reliable information on the condition of the SFP. This will expand the list of instrumentation needed to fully monitor severe accident conditions and it will also inform assessments of equipment survivability.
- Order EA-13-109[41] - This order requires that all BWRs with Mark I and II designs install reliable hardened containment vents that remain functional under severe accident conditions.

The NRC staff is meeting with appropriate Tier 1 teams to review instrumentation-needs formulations and review pertinent licensee submittals for instrumentation-needs identification. In addition, the NRC staff is reviewing information from previous and ongoing research efforts for severe accident management analysis, and is monitoring results of DOE, industry, and international research activities and reviewing new guidance being developed by domestic and international organizations (see Section 2.5).[21] Tasks to develop new information and insights include: (1) reviewing DOE modeling of the Fukushima event, (2) meeting with DOE and EPRI regarding research activities, (3) participating in development of an IAEA document on this topic,[8] (4) meeting with the ANS Standards Board, and (5) interfacing with the IEEE Standards Committee for IEEE-497, "Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." Once the NRC staff has accumulated sufficient knowledge and data, they have indicated that, if a safety-significant instrumentation performance gap is identified, regulatory action will be taken through the appropriate mechanism (rulemaking, generic communication, etc.).

Some of the questions that will be addressed by the NRC staff include:

- Is the current instrumentation identified in RG 1.97 adequate to cover the full range of severe accident conditions suggested by the Fukushima event?
- Will the instrumentation qualified to address the guidance of RG 1.97 survive with adequate capability to ensure monitoring of severe accident conditions?

As indicated in Section 2.3, Reference 21 indicates that the NRC is considering several options, such as dedicated independent power sources for critical plant instrumentation for time periods before diverse and flexible coping capability or "FLEX"* equipment could be installed, analyses and environmental testing that demonstrate that critical instrumentation will survive 'well into the accident progression', and operating procedures that incorporate insights from such analyses and testing. Reference 42 indicates the NRC will make a regulatory determination on this topic by December 2015.

* FLEX is a strategy developed by the U.S. nuclear industry in response to the accidents at Fukushima Daiichi wherein portable equipment such as pumps and generators kept on site or delivered from one of two regional FLEX facilities and used in a "flexible" way to respond to various potential challenges to core cooling and power restoration.

# APPENDIX B. PRIOR INDUSTRY EFFORTS

As noted in Section 2, this appendix provides additional details related to selected prior industry efforts to evaluate instrumentation survivability.

As discussed in Section 2.4.2, Reference 43 describes results from a systematic process followed by EPRI to evaluate what types of information might be expected from various types of installed instrumentation during severe accident conditions. The information and fourteen generic instrumentation groups listed in Table B-1 were identified in this process.

**Table B-1.** Instrumentation loop group numbers and associated information

| Generic Instrumentation Group Number | Information |
|---|---|
| 1 | Core temperature (as indicated by core exit thermocouples) |
| 2 | Core outlet temperature (as indicated by a variety of measuring devices) |
| 3 | RPV upper internals / structure temperature (inferred from water temperature or control rod drive temperature) |
| 4 | Core water level |
| 5 | Hot leg temperature |
| 6 | Core external power monitors |
| 7 | Pressurizer water level |
| 8 | Reactor system pressure |
| 9 | Containment pressure |
| 10 | Containment temperature |
| 11 | Containment radiation levels |
| 12 | Containment hydrogen levels |
| 13 | Suppression / refueling pool temperature |
| 14 | Valve position indication |

In addition, tables were developed that list ranges of interest for various types of parameters during different phases or conditions that may occur during a severe accident [e.g., OX (intact fuel), BD (fuel significantly oxidized), and EX (core relocated ex-vessel), etc.]). Conditions of primary interest (and associated measurable parameters for the PWR pilot plant evaluation) are summarized below in Table B-2. As shown in this table, the accident phases are similar to those identified in the NRC studies. Additional information about each of these phases, which were developed based on information in References 34, 43, and 44, is found in Table B-3 through B-5. In some cases, highly accurate data are not required by plant operators. Rather, it is sufficient to simply know data trends.

**Table B-2.** Major damage condition descriptors and possible symptoms identified in Reference 44.

| Damage Condition | | Possible Symptoms |
|---|---|---|
| ID | Description | |
| OX | Intact fuel (clad ballooning, oxidation, or collapse might have occurred; no core structural materials—fuel, clad, or steel—molten) or RCS damage | • Core outlet temperature (where appropriate) > 650 °C (1200°F).<br>• Considerable superheat [> 93 °C (200°F]measured in hot legs.<br>• Core water level: collapsed water height at or below core mid-plane.<br>• Loss of pressurizer level (for PWRs without loop seal).<br>• External core power monitors increasing.<br>• Some or increasing hydrogen measured in containment.<br>• Hot leg and/or surge line temperature at or near maximum measured value along with indications of damage condition OX.<br>• RCS pressure at or near nominal operating value along with indications of damage condition OX.<br>• Hydrogen measured in containment<br>• Limited radiation in containment, perhaps due to primary coolant activity and the release of fission product gases from fuel clad gap, as well as limited diffusion from the fuel matrix. |
| BD | Core significantly oxidized and not intact (core structural components have melted and are relocating downward); RCS pressure boundary (hot leg, surge line, and/or SG tube failure) | • High radiation in containment with indications of BD.<br>• Increasing hydrogen measured in containment with RCS at or near operating pressure<br>• Core outlet temperatures (where appropriate) > 1090 ° C (2000°F)/<br>• Loss of pressurizer level (in PWRs without loop seal) with indications of damage condition BD<br>• External core power monitors increasing.<br>• Collapsed water level at or below 40% core height for 10 minutes or longer.<br>• Hot leg and/or surge line temperature at or near maximum measured value.<br>• High radiation in steam generator |
| EX/CH | Core debris relocated ex-vessel into the primary containment (RPV failed); Containment is closed but challenged; Core concrete attack | • Core outlet temperatures (where appropriate) > 1090 ° C (2000°F)/<br>• RCS depressurization combined with containment pressurization.<br>• RCS pressure essentially equal to the containment pressure.<br>• High radiation in containment.<br>• Hydrogen measured in containment (>20% of the active fuel cladding reacted).<br>• Continually increasing containment pressure<br>• Continually increasing containment temperatures (more than saturation temperature).<br>• No indication of water injection or containment heat removal<br>• CO and/or $CO_2$ measured in containment and increasing.<br>• Indication from heat balance on RCS and containment that the removal rate is less than decay heat. |
| I | Containment boundary impaired (containment isolation function not complete). | • Isolation not complete.<br>• Steam release detected outside containment.<br>• High radiation detected outside containment.<br>• Decrease in containment pressure in absence of containment heat removal. |
| B | Containment bypassed (RCS isolation function not complete). | • Indication that containment isolation is not complete.<br>• High pressure or ruptured disk in the pressurized quench tank (for systems with relief valves on the low-pressure systems piped to the quench tank).<br>• High humidity or flooding detected in the secondary containment/auxiliary building.<br>• High temperatures detected in the secondary containment/auxiliary building.<br>• High radiation detected outside containment.<br>• High RCS pressure (near nominal operating condition) and condition BD.<br>• Water accumulation detected in secondary containment/auxiliary building.<br>• Activation of fire suppression system or isolation dampers in secondary containment/auxiliary building.<br>• High radiation detected in the standby gas treatment system.<br>• High radiation detected in steam generators |

**Table B-2.** Major damage condition descriptors and possible symptoms identified in Reference 44.

| Damage Condition | | Possible Symptoms |
|---|---|---|
| ID | Description | |
| SC-CC | Secondary containment undamaged, closed, and cooled. | • Building ventilation system is available.<br>• Releases from the building are monitored and filtered or released at a high elevation. |
| SC-CH | Secondary containment closed but challenged. | • Releases from the building are un-monitored or at ground level.<br>• Conditions in the RCS pose potential for containment bypass.<br>• Containment pressure and temperature high and increasing.<br>• Building atmospheric temperature high and increasing.<br>• The concentration of hydrogen, if measured, is potentially increasing in the secondary containment/auxiliary building. The concentration of CO and $CO_2$, if measured is potentially increasing in the secondary containment/auxiliary building. |
| SC-F | Secondary containment has failed with large path to the environment. | • Secondary containment pressure at ambient environmental conditions.<br>• Visual inspection of the exterior of the building could indicate the failure location (for example, a failed blowout panel).<br>• Increase in measured dose rates at the site boundary |

**Table B-3.** Parameter Table - Condition OX

| Functional Need | Range of Interest | Generic Instrumentation Group Providing Data[a] |
|---|---|---|
| Core outlet temperature | > 650 °C (1200 °F) | 1,2 |
| Core temperature | not listed | 1,2 |
| Average core temperature | 635 - 1270 °C (1175-2335 °F) | 1,2 |
| Core exit gas temperature | 1260 °C (2300 °F) | 1,2 |
| Upper plenum structure temperature | 620 °C (1150 °F) | 1,2 |
| Maximum reactor system pressure | 17.8 MPa (2550 psia) | 8 |
| Minimum reactor system pressure | 0.25 - 0.28 MPa (36-40 psia) | 8 |
| RPV exit gas temperature | 680 ° C (1250 °F) | 1,2 |
| Hot leg temperature | > 110 °C (200 °F) superheat; 427 ° C(800 °F) | 5,8 |
| Containment pressure (with and without hydrogen burns) | 0.18 to 0.20 MPa (26 to 29 psia) | 9 |
| Containment temperature (with and without hydrogen burns) | 99 - 119 °C (211- 246 °F) | 10 |
| Containment radiation levels | limited | 11 |
| Core water level | at or below core mid-plane | 4 |
| Pressurizer water level | lowering | 7 |
| External core power monitors | “increasing” | 6 |
| Containment hydrogen levels | “present or increasing” | 12 |
| Location of core material | in-vessel or ex-vessel | 11 |

a. See Table B-1.

**Table B-4.** Parameter Table - Condition BD

| Functional Need | Range of Interest | Generic Instrumentation Group Providing Data[a] |
|---|---|---|
| Average core temperature | 2360 °C (4285 °F) | 2 |
| Core outlet temperature | > 1090 ° C (2000 °F) | 1,2 |
| Core exit gas temperature | 2040-2150 ° C (3700-3900 °F) | 1,2 |
| Upper plenum structure temperature | 982-1900 ° C (1800-3450 °F) | 1,2 |
| Maximum reactor system pressure | 15.5 MPa (2550 psia) | 8 |
| Minimum reactor system pressure | 0.22 to 0.23 MPa (32-34 psia) | 8 |
| RPV exit gas temperature | 816-982 ° C (1500-1800 °F) | 1,2 |
| Hot leg temperature | >110 °C (200 °F) superheat; 427 ° C(800 °F) | 1,2,5 |
| Containment pressure (with and without hydrogen burns) | 0.26 to 1.1 MPa (37 to 149 psia) | 9 |
| Containment temperature (with and without hydrogen burns) | 93-1100 ° C (200-2031 °F) | 10 |
| Containment radiation levels | limited | 11 |
| Core water level | at or below 40% core height | 4 |
| Pressurizer water level | zero | 7 |
| External core power monitors | "increasing" | 6 |
| Containment hydrogen levels | "present or increasing" | 12 |
| Location of core material | in-vessel or ex-vessel | 11 |

a. See Table B-1.

**Table B-5.** Parameter Table - Condition EX

| Functional Need | Range of Interest | Generic Instrumentation Group Providing Data[a] |
|---|---|---|
| Core water level | lost | 4,6 |
| Containment pressure (with and without hydrogen burns) | 0.79 to 1.0 MPa (114 to 150 psia) | 9, trend |
| Containment temperature (with and without hydrogen burns) | 362-2400 °F | 10, trend |
| Containment radiation levels | high | 11 |
| Containment hydrogen levels | "substantial" | 12 |

a. See Table B-1.

# APPENDIX C. MELCOR UNMITIGATED STSBO RESULTS

As noted in Section 4, this appendix provides additional plots from the supplemental MELCOR calculations provided by SNL for an unmitigated STSBO at the Surry PWR. Figures 4-1 through 4-3 show the configuration of the SOARCA hydrodynamic model used to simulate the Surry plant with the MELCOR code. Figure 4-4 shows the hydrodynamic nodalization of the containment.

SNL provided time-dependent temperature, pressure, and dose results for an unmitigated STSBO evaluated in the SOARCA event. These results were reviewed, and it was determined that the representative volumes listed in Table C-1 can be used to predict conditions for each building location of interest in these survivability assessments (e.g., predicted environmental conditions in the listed control volumes bounded conditions in other control volumes). Recognizing that instrumentation is often attached to structures, temperatures provided by SNL for structure locations of interest were also reviewed. Table C-2 summarizes peak lower head structure (LHS) temperatures predicted at locations where instrumentation may be impacted. Sections C.1 and C.2 present MELCOR results for the RCS and containment, respectively.

**Table C-1.** MELCOR STSBO results for representative Control Volumes.

| Location | ID | Description | Peak Pressure, MPa | Peak Temperature, °C | Beta Dose[a], Rad | Gamma Dose[a], Rad |
|---|---|---|---|---|---|---|
| Reactor Coolant System | CV 101 | Vessel downcomer | 17.3 | 907 (1180) | 1.8E+08 | 1.4E+08 |
| | CV 110 | Vessel lower plenum | 17.4 | 1643(1916) | 1.7E+10 | 3.1E+08 |
| | CV 160 | Vessel upper plenum | 17.3 | 894 (1167) | 5.1E+08 | 7.7E+08 |
| | CV 201 | Hot leg loop A | 17.3 | 885 (1158) | 3.0E+08 | 4.2E+08 |
| | CV 202 | SGA hot leg top | 17.3 | 874 (1147) | 2.8E+08 | 1.6E+08 |
| | CV 209 | SGA lower plenum | 17.3 | 868 (1141) | 2.8E+08 | 3.3E+07 |
| | CV220 | SGA relief line to auxiliary building | 17.3 | 370 (643) | 1.3E+08 | 1.8E+07 |
| | CV221 | SGA hot leg bottom | 17.3 | 584 (857) | 2.5E+08 | 4.9E+08 |
| | CV 237 | CL RCS piping from RCP | 17.3 | 776 (1049) | 9.8E+07 | 1.2E+07 |
| | CV 240 | RCS piping (accumulator injection) | 17.3 | 759 (1032) | 1.1E+08 | 3.4E+07 |
| | CV 301 | Hot leg loop B | 17.3 | 885 (1158) | 3.0E+08 | 4.1E+08 |
| | CV 302 | SGB hot leg top | 17.3 | 874 (1147) | 2.9E+08 | 1.6E+08 |
| | CV 309 | SGB lower plenum | 17.3 | 868 (1141) | 2.8E+08 | 3.3E+07 |
| | CV 321 | SGB hot leg bottom | 17.3 | 586 (859) | 2.5E+08 | 4.9E+08 |
| | CV 401 | Hot leg loop C | 17.3 | 897 (1170) | 2.3E+08 | 8.0E+08 |
| | CV 402 | SGC hot leg top | 17.3 | 886 (1159) | 2.3E+08 | 2.0E+08 |
| | CV 421 | SGC hot leg bottom | 17.3 | 614 (887) | 2.1E+08 | 6.1E+08 |
| | CV 701 | Vessel bypass | 17.3 | 1580 (1853) | 1.3E+08 | 7.9E+08 |
| Pressurizer and Piping | CV 500 | Pressurizer surge line | 17.3 | 705 (978) | 3.0E+08 | 2.2E+08 |
| | CV 520 | Pressurizer | 17.2 | 455 (728) | 1.8E+08 | 4.2E+07 |
| Containment | CV 5 | Basement | 0.80 | 351 (624) | 1.6E+09 | 3.4E+09 |
| | CV 10 | Reactor vessel cavity | 0.80 | 1640 (1913) | 1.1E+08 | 2.3E+06 |
| | CV 20 | SG A cubical | 0.80 | 238 (511) | 7.5E+07 | 1.9E+08 |
| | CV 30 | SG B cubical | 0.80 | 242 (515) | 7.4E+07 | 8.8E+07 |
| | CV 40 | SG C cubical | 0.80 | 381 (654) | 7.5E+07 | 1.1E+08 |
| | CV 41 | Pressurizer cubical | 0.80 | 267 (540) | 7.4E+07 | 2.2E+07 |
| | CV42 | Pressurizer relief tank cubical | 0.80 | 276 (549) | 8.6E+07 | 4.5E+07 |
| | CV 50 | Lower dome | 0.80 | 283 (556) | 1.0E+08 | 5.5E+08 |
| | CV55 | Upper dome | 0.80 | 227 (500) | 7.3E+07 | 4.3E+06 |

a. Cumulative dose after one year.

**Table C-2.** RCS heat structure temperatures

| Description | ID | Peak Temperature, °C |
|---|---|---|
| Lower head structures (center location) | TLH 101 | 1533 (1806) |
| Lower head structures | TLH 201 | 1800 (2073) |
| Lower head structures | TLH 301 | 1515 (1788) |
| Lower head structures | TLH 401 | 1447 (1720) |
| Lower head structures | TLH 501 | 1455 (1728) |
| Lower head structures (peripheral location) | TLH 601 | 815 (1088) |

# C.1. RCS Conditions

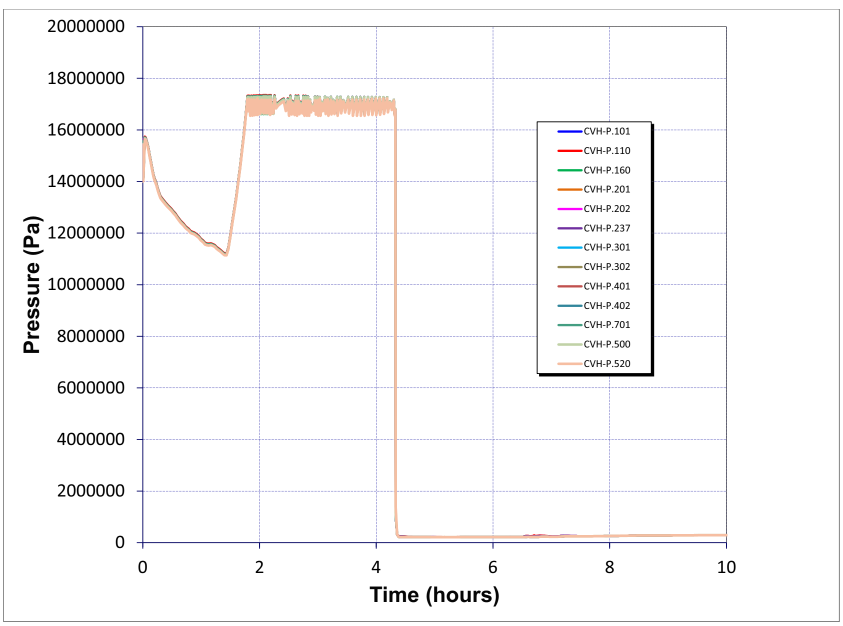



**Figure C-1.** Primary system component control volume pressure.

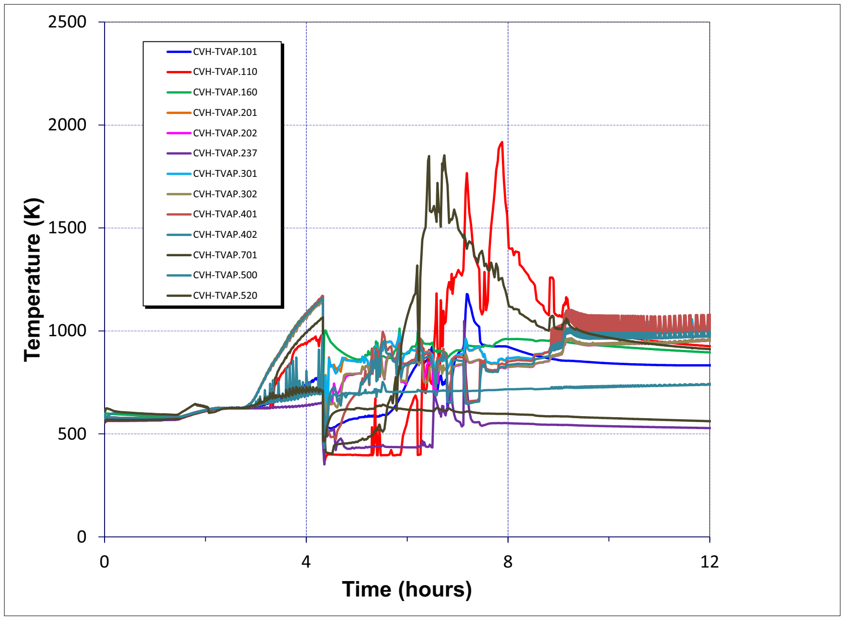



**Figure C-2.** Primary system component control volume temperature.

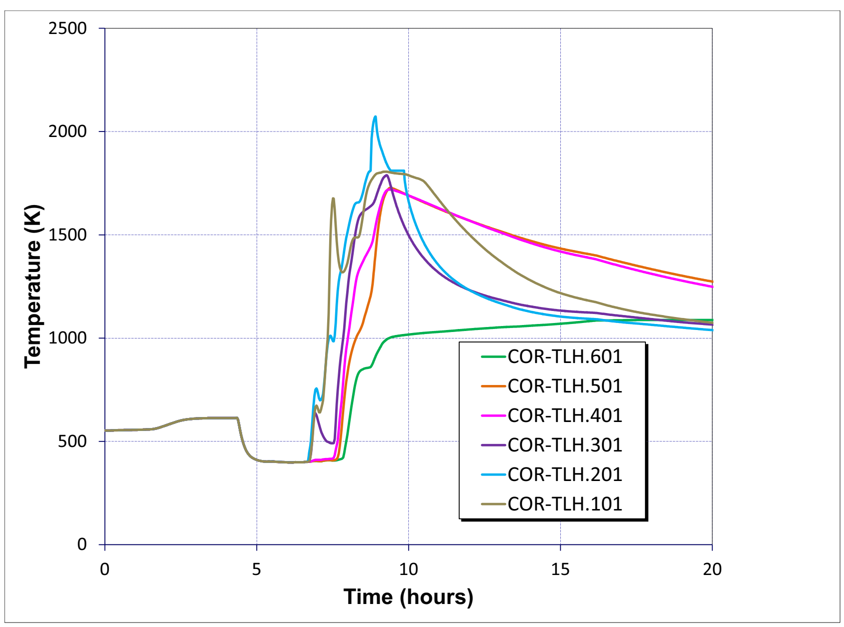



**Figure C-3.** RCS lower plenum structure temperatures.

## C.2. Containment Conditions

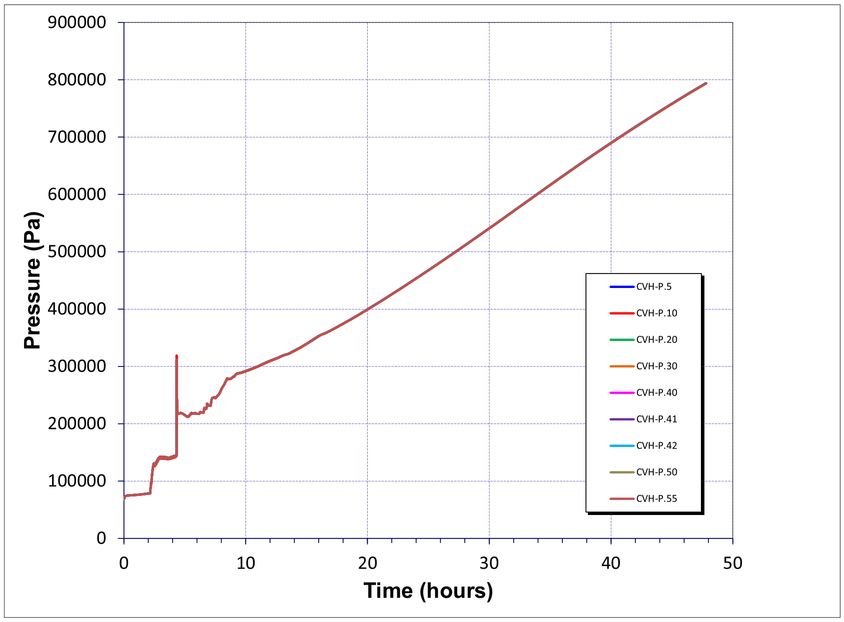



**Figure C-4.** Containment control volume pressure.

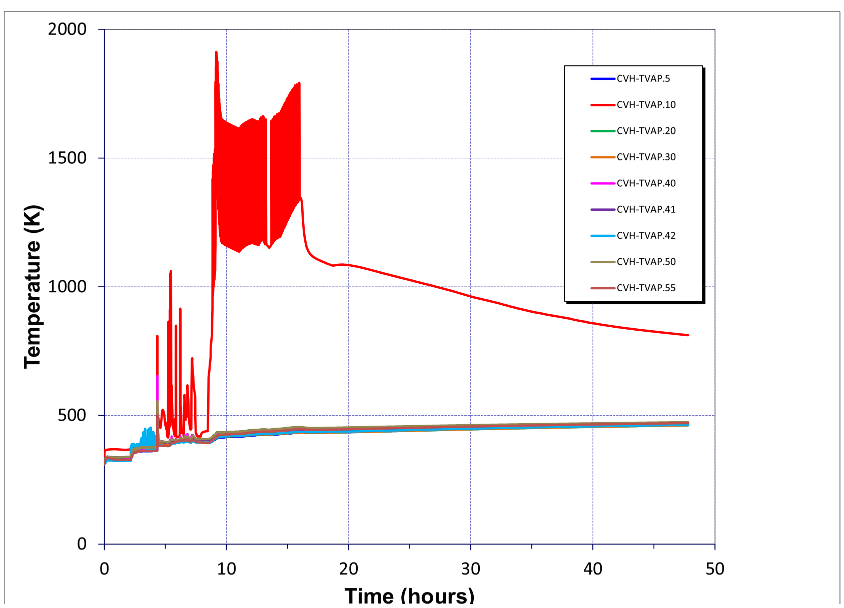



**Figure C-5.** Containment control volume temperatures.

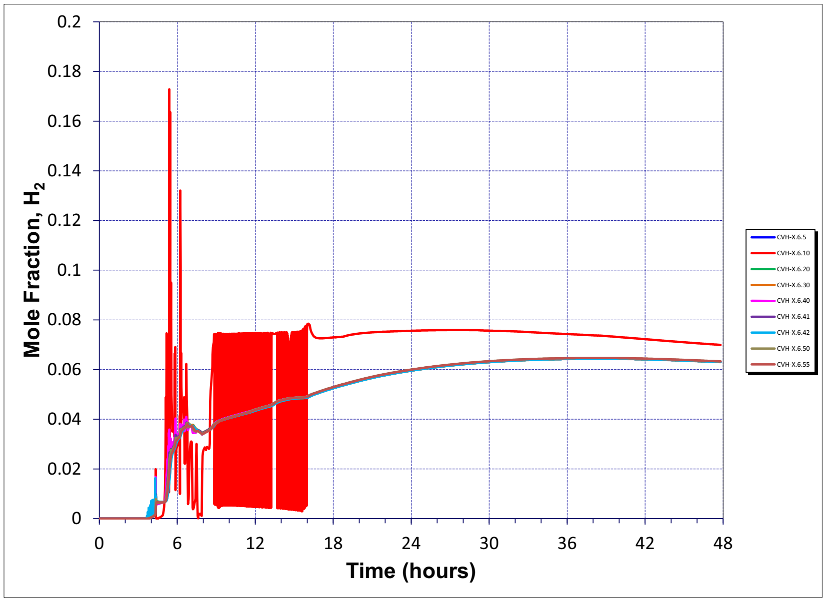



**Figure C-6.** Containment control volume hydrogen mole fraction.

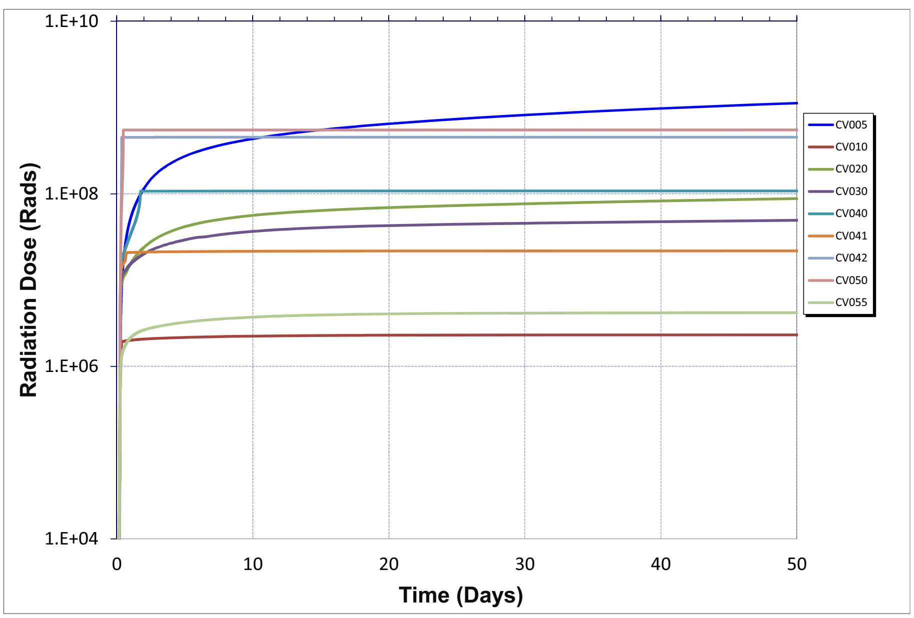



**Figure C-7.** Containment control volume gamma dose.

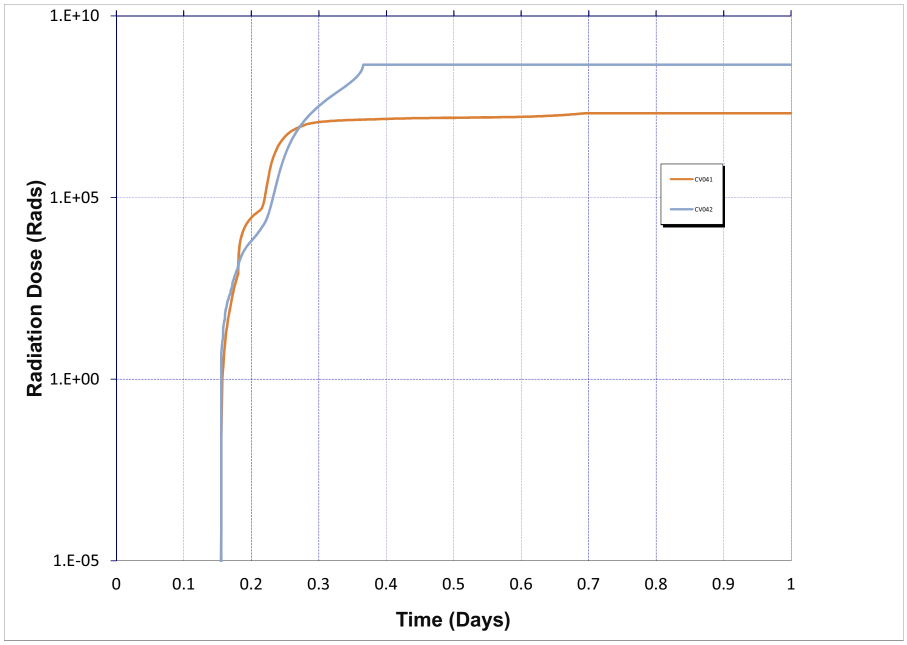



**Figure C-8.** Detail for gamma dose in containment control volumes 41 and 42.

# APPENDIX D. PEER REVIEW COMMENTS AND RESOLUTION

As part of this effort, a draft version of this document was provided to knowledgeable individuals for review. Comments were received from individuals from Dominion, EPRI, the PWROG, and the US NRC. These comments are listed in this appendix with the response taken to address this comment.

**Table D-1.** Peer Review Comments and Resolution

| Organization / Reviewer(s) | Comment | Response/Action |
|---|---|---|
| Dominion / William Webster | Before resources are spent on additional research for instrument survivability, EPRI and/or PWROG should investigate the consequences of the failure of any particular instrument to determine if further work is warranted. Since the primary focus of SAMG strategies is on containment and bypass sequences, then the list of "required instruments" may be even less if the consequence of failure of an instrument doesn't change the recommended strategy that would be implemented with or without the instrument or if alternative methods or calculations to support actions for making appropriate decisions is available (e.g. containment hydrogen explosions used in existing SAMGs). | On-going work by the PWROG is addressing the use of alternative methods or calculations as well as alternate sensors to support operator decision making. The revised report reflects this point. In addition, many of these points are mentioned in the updated Section 6 (Summary and Recommendations). |
| | Additionally, as stated in the report, one of the unmitigated sequences for the Surry SOARCA was that the seismic event failed DC, no instruments are available for this case or alternate methods for reading the instruments would have to be deployed. Operator actions, however, would still be appropriate for mitigation which would reduce the probability of offsite releases with or without given instruments (e.g., fill containment). | The authors concur with this point. However, as indicated in the revised report, additional evaluations could provide insights related to the range of conditions when one can rely on such instruments and what instrumentation must be available for such mitigating actions. |
| | Additionally, since these are very low probability events (especially after implementation of FLEX) coupled with the redundancy and expected survivability of at least some of the existing instruments, there would be limited or no justification for the sites to make plant modifications as a result of any additional research. | The authors concur that, at this time, there is no justification for plant hardware modifications. However, as indicated in the revised report, additional research could provide confidence in industry assumptions related to instrumentation survivability and trending performance when systems are exposed to conditions outside of EQ ranges. |
| EPRI / Joe Naser | Overall this was a well-organized and clear report; the companion BWR report should be redone to match this report's flow, thoroughness and clarity; I agree with the approach and they did a good job reviewing the past work on the subject | The authors appreciate this comment. The suggestion for the BWR report was forwarded to the Reactor Safety Technology lead for consideration. |
| | Drew Mantey should review for EQ items. | This comment was received after the end of the comment period, and it was too late to enlist a new reviewer. Rather, Robert Lutz was added as an author. |
| | As the report discusses Surry directly as an example plant, Dominion should at least be given a "heads up" about this report's pending publication by INL; this heads-up should come from DOE. | The report was previously transmitted to Dominion. As noted in this table, Dominion comments were provided from William Webster. |
| | The Abstract and Introduction overstate the criticality of instrumentation at TMI-2 and Fukushima; I still maintain that not having sufficient instrumentation did NOT contribute to or worsen the Fukushima accident; I wish the language in both these sections could be toned down. | The authors disagree with this comment. References were added after this statement to support this statement about the importance of having accurate, relevant, and timely information on the status of reactor systems during a severe accident. |

**Table D-1.** Peer Review Comments and Resolution

| Organization / Reviewer(s) | Comment | Response/Action |
|---|---|---|
| EPRI / Joe Naser (Continued) | I think the current Section 5 needs to be retitled as "Assess Instrument Availability", and a new "Summary & Conclusions" section added. | The authors agreed with and separated Section 5 into two sections. |
| | The rest of the report, like I said, was very well done. Some ideas for future follow on work (which is admittedly beyond the scope of this report) might be:<br>• Which of the instruments identified in Table 5-1 would survive, which are questionable, and which are likely toast? (toast is a technical term).<br>• For the ones that are questionable (for example, the Wide-Range SG level, which exceeds temperature and pressure, but only by a factor of 2) would they likely survive or what additional work is needed to determine the survivability?<br>• For the ones that are likely toast (for example, the RVLIS, which would see a temperature of 1,643 C, so it's gone) can you still do what you need to do to mitigate the accident without this system? Bear in mind that for the accident to have progressed to this point, RVLIS might be a moot point by now anyway; same goes for the power range monitors (someone with more ops experience can confirm this)<br>• If the information from toasted instruments are in fact needed, what alternative means of getting the information might be employed? | The authors appreciate this comment. However, as observed by the reviewer, it is beyond the current workscope. Nevertheless, the revised report does address many of these comments. The revised report also notes the need for more detailed, plant specific, information so that these suggested actions could be performed as follow-on work. In addition, the revised report observes that industry is performing the last task, as part of on-going industry efforts to update SAMGs and develop TSG for implementing the new SAMGs. |
| PWROG/ Comments from S. Pierson, M. Weiner, and R. Lutz | Page 32 says we lose the TDAFW pump due to seismically induced loss of DC power and inability to remotely control the pump. For many plants, a loss of DC power fails open all the flow control valves, causing overall, not loss of pump (this is plant specific - some plants may lose TDAFW on loss of DC). If TDAFW fails due to overall, it pushes out SG dryout to ~10 hrs as I recall from the B5b WCAP. I think they need to say TDAFW is lost due to seismic event directly (perhaps stop valve trips due to seismic and failure of aux bldg prevents access to reset it). | The SOARCA report indicated that the ECST ruptures and precludes TDAFW injection, and additional plant-specific information related to Surry response was not available. Hence, text was revised to acknowledge that the seismic event causes a loss of DC power and the TDAFW system. |
| | Page 45 says FLEX may reduce risk during this event. That may be true for plants with low leakage seals only. If TDAFW fails at time zero, I doubt anyone can drag in FLEX equipment and start feeding SGs prior to dryout at roughly an hour. After dryout, the OEM seals fail and high RCS leakage is beyond most plants FLEX RCS makeup capability (which would have to include sump recirc capability). However, a different story exists for those with low leakage seals, since they will significantly limit seal leakage even after O-rings fail. With low leakage seals, you could prevent core damage by reestablishing SG feed with FLEX prior to boiling away all primary inventory and then by establishing FLEX RCS makeup within leakage rate of low leakage seals. A few hours extra time provided by low leakage seal design may give you a chance for success with FLEX. I would say FLEX may reduce risk at some plants for this event. | This text was revised on this page and a footnote was added to address this comment. |
| | Abstract, Section 2.4.1 - Suggested changes to text for clarification. | Changes incorporated. |
| | Section 2.4 - In this section, the industry efforts to assure reliable instrumentation response for beyond design basis accidents should be addressed. | This section was substantially revised to describe industry efforts. The discussion includes a description of the FSGs and on-going efforts to develop plant-specific versions of this guidance. |

**Table D-1.** Peer Review Comments and Resolution

| Organization / Reviewer(s) | Comment | Response/Action |
|---|---|---|
| PWROG/ Comments from S. Pierson, M. Weiner, and R. Lutz<br><br>(Continued) | Section 2.4 - Suggested change to text for clarification. | Changes incorporated. |
| | Section 2.4.3 - The knowledge that can be gained from trending cannot be underestimated. Even if environmental conditions introduce errors in absolute values for parameters, comparison of parameter trends with expected trends, valuable information can be obtained for decision making. | The revised report discusses the proposed use of trending information, but it also observes that, in some cases, additional experimental data are needed to support this assertion. In addition, Section 2.5 was added to describe DOE efforts. |
| | Section 2.4.3 - Suggested change to text for clarification. | Changes incorporated. |
| | Section 3.2 - I think you are randomly mixing the "whole" and the "pieces". EOPs include ECAs and FRs; EDMGs stand by themselves, SAMG include SACRGs and CAs. The new FLEX Support Guidelines (FSG) are also a subset of the EOPs for use in certain BDB conditions to provide alternate strategies for core, containment and spent fuel cooling. | Text and Figure 3.3 were revised to address this comment. |
| | Section 3.2 - Corrections to Table 3.1. | Corrections were incorporated. |
| | Section 3.2 - Another reference would be the testimony of N.J. Stringfellow, PWROG Chairman, at the July 9, 2015 Commission Briefing on Mitigation of Beyond Design Basis Events Rulemaking. | This reference was added. |
| | Section 3.3 - Corrections to Table 3.3. | Corrections were incorporated. |
| | Section 4.1 - The heating effect from electrical loads in the aux building are also not considered. Because this scoping evaluation is based on the unmitigated STSBO, the heating from electrical loads would not be significant because all a.c. and d.c. power is lost.<br><br>If you are running any other scenario where FLEX would not be credited (see my separate proposed paragraph for Section 2.4 concerning FLEX), an acknowledgment would be appropriate that aux building heating from electrical loads to run instrumentation are not modeled in the scoping evaluation. | Section 2.4 was revised to include the provided input (and Mr. Lutz was added as a co-author). Text was added in Section 4.1 to note that an expanded scoping evaluation could be used to quantify the timing and conditions that challenge instrumentation in the Auxiliary Building. |
| | Section 4.2 - It is interesting that the unmitigated STSBO was chosen to derive environmental conditions for instrumentation survivability when in fact no instrumentation would be available due to the loss of all a.c. and d.c. power. I understand the reason, just the dichotomy of the situation is amusing. | It was originally planned that several sequences would be evaluated. The revised report recommends that evaluations of additional sequences be performed. |
| | Section 4.3.2.2 - The Ref 6 analysis says vessel failure at 7.25 hours but hot leg creep failure at about 4.25 hours. The supplemental SNL calcs show something at about 5 hours (high oscillating cavity temps) and then prolonged extremely high temps at about 9 hours. So is vessel failure really at about 5 hours in the supplemental calcs or is this hot leg creep failure? This is important in regards to my comment on page 44 (Table 5-1). | The text was modified to explicitly note times when instrumentation failure may occur (due to exceeding temperatures of concern) and if these times were before or after hot leg failure. |
| | Table 5-1 - It is interesting to note that SG and RCS parameters are only important prior to reactor vessel failure (or in the time frame immediately following vessel failure as a signature of reactor vessel failure). In the MELCOR analysis, this occurs at about 8 or 9 hours into the accident sequence. Up to this time, the containment environmental conditions are less than about 0.3 MPa and 400K which is well within their EQ envelop for the time period in which these key parameters are of most interest. I think that this report should discuss time phasing of importance of parameter indications. | Section 4, which discusses timing of events, was revised to discuss phasing. However, results don't indicate that RCS temperatures only exceed EQ values after vessel failure. Peak values start occurring at around 6 hours (which is prior to the time of vessel failure). Likewise, some of the containment conditions exceed EQ values prior to the time of vessel failure. |

**Table D-1.** Peer Review Comments and Resolution

| Organization / Reviewer(s) | Comment | Response/Action |
|---|---|---|
| PWROG/ Comments from S. Pierson, M. Weiner, and R. Lutz<br><br>(Continued) | Table 5-1 - Because accumulators have check valves that prevent inflow of RCS fluids, they would only be exposed to temperatures equivalent to the accumulator fluid - much less than 749 C. | This point has been added as a footnote to this table. Because available references indicate that the sensors weren't challenged by 749 °C and because more detailed information related to peak temperatures wasn't known, the revised report simply notes that temperatures are much lower. |
| | Table 5-1 - I am surprised that this is so low -- until hot leg creep failure, the pathway for discharging superheated steam and hydrogen from the RCS is the pressurizer SRVs. I would expect this to be more similar to the 1643C shown for CETCs. | Supplemental MELCOR calculations indicate that the surge line (CV500) reaches higher temperatures than the pressurizer (CV520). Predicted temperatures in the lower head (CV110) volume are higher because of relocated debris. |
| | Table 5-1 - I don't have detailed info available for Surry, but I believe that (at least some of the Westinghouse PWRs) the power range and source range detection is via the ex-core detectors and therefore they would not be exposed to RCS conditions. Please check this. | Location information not available for Surry. Hence, this comment was addressed by adding a footnote that acknowledges uncertainty in detector location and that the assumed location provides conservative peak temperature estimates. |
| | Table 5-1 - It is highly unlikely that measurement for these three parameters is in the reactor cavity. Sump level is most likely in one of the SG compartments (CV 20, 30 or 40) or possibly CV 5 or 50. Hydrogen monitor is typically from a higher elevation in containment such as CV 50 or 55 and may have multiple sample points. Containment temperature is most likely above the operating deck in CV 50 or 55. | Because plant-specific information was not available, the revised table indicates that all these suggested locations are possible. Listed peak temperatures, pressures, and doses are the highest values. predicted for these locations (and exceed their EQ values). |
| | Section 5 - I am not sure that this is as benign as it might appear -- the current analyzers, while located in the AB, continuously bring containment gases into the AB and also require a.c. power to heat the analyzer internals to quite high temperatures. With no ventilation for an SBO type sequence, the room where it is located will become quite hot (temperature and radioactivity). | Text was added in Section 5 (and in Section 4) of the revised report to address this point. |
| | Section 5 - However, for a mitigated sequence where containment cooling is recovered, hydrogen combustion would be a more important both from the affect on instrument performance as well as the ability to accurately estimate the combustible gas concentrations. | This point has been included in the revised report. |
| | Section 5 - There was considerable effort to put this report together and it really does represent the state of knowledge that can be ascertained through publicly available documents. But I think the next steps are weak and really do not provide a good picture of the importance of continued research. Perhaps some broad suggestions such as more detailed modeling of locations, investigating Fukushima instrument conditions after exposure to BDB environment, extended EQ testing of existing instrumentation. to determine P and T that result in errors / failures. | To address this comment and similar comments from NRC and EPRI, additional text was added to the report the defines additional tasks. Justification for each task is also provided. Because this additional work would most likely be coordinated with the on-going Reactor Safety Technology (RST) pathway within the Department of Energy, a subsection was also added to Section 2 to discuss this DOE effort. |
| | Appendix A - Are you referring to Reference 3 of the main report which was published in 2015? | Yes. However, because this paragraph was redundant with previous text in this report, it was deleted. |
| | Appendix A - Completed' rather than 'plans' | This correction was incorporated. |

**Table D-1.** Peer Review Comments and Resolution

| Organization / Reviewer(s) | Comment | Response/Action |
|---|---|---|
| US NRC/ provided by R. Sydnor and P. Chung | Abstract - Consider adding a sentence on the key limitations. | A sentence was added that describes some of the key limitations. |
| | Abstract - Why? - Industry is proposing alternative strategies when installed instrumentation is not available. | A sentence was added to address this question. |
| | Acronyms - add Institute | This correction was incorporated. |
| | Section 2.2 - consider adding a sentence on why only the STSBO was used. | The revised report observes some of the limitations of only evaluating the unmitigated SBO and (in later sections) recommends that evaluations of other sequences be performed. |
| | Section 4.3.1 - The term supplemental calculations is used many times throughout the report and it is not always clear what is being referred to by this term; e.g. in section 4.3.1 the limitations seem focused on the results not specific calculations. | The title of Section 4.3.1 was corrected. In addition, the report was reviewed to better reflect when text referred to initial calculations, supplemental calculations, and the scoping evaluation. |
| | Section 4.3.2 - Are there limitations from the selection of just using the STSBO that should be listed? | Yes. The revised report observes some of the limitations of only evaluating the unmitigated SBO and recommends that evaluations of other sequences be performed. |
| | Section 5 - Are there limitations from the selection of just using the STSBO that should be listed? | Yes. The revised report observes some of the limitations of only evaluating the unmitigated SBO and recommends that evaluations of other sequences be performed. |
| | Section 5 - Consider providing specific examples of recommended additional research, such as Instrument survivability under LTSBO, TISGTR, ISLOCA conditions and/or specific EQ analysis or tests. | To address this comment and similar comments from NRC and EPRI, additional text was added to the report the defines additional tasks. Justification for each task is also provided. Because this additional work would most likely be coordinated with the on-going Reactor Safety Technology (RST) pathway within the Department of Energy, a subsection was also added to Section 2 to discuss this DOE effort. |