Group _A____

FOIA/PA NO: 2013-0240

RECORDS BEING RELEASED IN PART

The following types of information are being withheld:

Ex. 1: ☐ Records properly classified pursuant to Executive Order 13526
Ex. 2: ☐ Records regarding personnel rules and/or human capital administration
Ex. 3: ☐ Information about the design, manufacture, or utilization of nuclear weapons
☐ Information about the protection or security of reactors and nuclear materials
☐ Contractor proposals not incorporated into a final contract with the NRC
☐ Other ____________________
Ex. 4: ☐ Proprietary information provided by a submitter to the NRC
☐ Other ____________________
Ex. 5: ☐ Draft documents or other pre-decisional deliberative documents (D.P. Privilege)
☐ Records prepared by counsel in anticipation of litigation (A.W.P. Privilege)
☐ Privileged communications between counsel and a client (A.C. Privilege)
☒ Other ____________________
Ex. 6: ☒ Agency employee PII, including SSN, contact information, birthdates, etc.
☐ Third party PII, including names, phone numbers, or other personal information
Ex. 7(A): ☐ Copies of ongoing investigation case files, exhibits, notes, ROI's, etc.
☐ Records that reference or are related to a separate ongoing investigation(s)
Ex. 7(C): ☐ Special Agent or other law enforcement PII
☐ PII of third parties referenced in records compiled for law enforcement purposes
Ex. 7(D): ☐ Witnesses' and Allegers' PII in law enforcement records
☐ Confidential Informant or law enforcement information provided by other entity
Ex. 7(E): ☐ Law Enforcement Technique/Procedure used for criminal investigations
☐ Technique or procedure used for security or prevention of criminal activity
Ex. 7(F): ☒ Information that could aid a terrorist or compromise security

Other/Comments: ____________________

Schaperow, Jason

NRO RES

From:	Schaperow, Jason
Sent:	Thursday, June 28, 2012 1:59 PM
To:	Gibson, Kathy; Esmaili, Hossein; Helton, Donald; Fuller, Edward; Santiago, Patricia; Lee, Richard
Subject:	FW: the additional questions
Attachments:	RESPONSE: Commissioner Magwood Question on SFPs

This morning, we sent Commissioner Magwood's office the attached email. Subsequently, Rebecca Tadesse from Commissioner Magwood's office called me and sent me the email below. At 1:15 p.m., I called her back

I told her the following with respect to her first question below:

The post-9/11 security studies included holes in the bottom of the pool and in the side of the pool. For a hole in the bottom of the pool, the pool drained down completely so the fuel was not covered. For a hole in the side of the pool, the pool drained down to the elevation of the hole so the fuel was partially covered.

The Spent Fuel Pool Scoping Study only has cases with a hole in the bottom of the pool. In the Spent Fuel Pool Scoping Study, the pool drains down completely and the fuel heats up and catches fire. The pool drain-down and heat-up plots for the two cases (1.75" hole (aka small leak) and 4.5" hole (aka moderate leak)) are given on slides 15 and 16 of our briefing for Chairman Jaczko on June 12, 2012.

I told her the following with respect to her second question below:

Plugging or clogging a hole in the pool wall might be accomplished by sliding a steel plate down into the pool. Such mitigation strategies (to slow or stop a leak) were considered as part of the site-specific assessments that each licensee conducted and the inspections that the NRC conducted at each site to address vulnerability and mitigation measures. I do not know whether any of the plants have such mitigation strategies in place. As far as I know, there are no NRC requirements for this. However, Eric Bowman in NRR would know exactly whether any of the plants have such mitigation strategies (to stop or slow a leak) in place.

From: Tadesse, Rebecca
Sent: Thursday, June 28, 2012 10:30 AM
To: Schaperow, Jason
Subject: the additional questions

Commission

Hi Jason,

Per our conversation these are the two questions that the Commissioner wants to get information on:

Do these analyses assume that the bottom of the fuel is still covered?

Doesn't this suggest that having some mechanism to slow the loss of water (plugging or clogging the hole) would be important? Have they considered that?

Rebecca Tadesse
U.S. Nuclear Regulatory Commission
Policy Advisor for Materials
Office of Commissioner William D. Magwood
301-415-8425
Rebecca.Tadesse@nrc.gov

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Schaperow, Jason

From:	Rini, Brett
Sent:	Thursday, June 28, 2012 7:54 AM
To:	Chen, Yen-Ju
Cc:	Gibson, Kathy
Subject:	RESPONSE: Commissioner Magwood Question on SFPs

RES

Yen,

Here's our response to Commissioner Magwood's question from this week's briefing. Please forward to his office.

A staff briefing for Commissioners Magwood and Ostendorff on spent fuel safety on June 25, 2012, included predictions of minimum decay times for which fuel in a spent fuel pool is air coolable for different fuel loading patterns such as one recently offloaded assembly surrounded by four low power assemblies (1 by 4). During the briefing, Commissioner Magwood asked how long it takes for the pool to drain and a zirc fire to begin for scenarios in which the fuel is not air coolable.

For a 1.75"-diameter hole in the pool with the most recently offloaded fuel with 37 days of decay arranged in a 1 by 4 pattern, the water level reaches the top of the fuel at 20 hours and rapid cladding oxidation (zirc fire) begins at 40 hours. For a 4.5"-diameter hole, the water level reaches the top of the fuel at 3 hours and rapid cladding oxidation begins at 16 hours. These predictions are part of the staff's analysis for the ongoing Spent Fuel Pool Scoping Study that estimated that the frequency of SFP liner damage due to the seismic event studied is roughly 1.6e-6 per year. The frequency of a radiological release is lower than this value by 1 to 3 orders of magnitude for the scenario considered, due to the additional considerations of AC power fragility, air coolability during large portions of the operating cycle, and the beneficial effects of deploying mitigation (when it is credited).

If there are any questions or for further information, please contact Katie Wagner at (301) 251-7917 or Katie.Wagner@nrc.gov.

Thanks,
Brett

/01

Ahn, Tae

Subject: 4/10, Update - Seminar on Spent Nuclear Fuel
Location: E2C19

Start: Fri 04/26/2013 8:30 AM
End: Fri 04/26/2013 12:30 PM

Recurrence: (none)

Meeting Status: Meeting organizer

Organizer: Ahn, Tae
Required Attendees: 'Vincenzo.RONDINELLA@ec.europa.eu'; Guttmann, Jack; Rubenstone, James; Lee, Richard; Einziger, Robert; Voglewede, John; Raynaud, Patrick; Scott, Harold; Flanagan, Michelle; David Pickett; Gwo, Jin-Ping; Doolittle, Elizabeth; Smith, Shawn
Optional Attendees: Cao, Tianqing; Kim, Yong; Gray, Anita; McCartin, Timothy; Hill, Brittain; Pantalo, Charity; Dunn, Darrell; Jagannath, Banad; 'Pickett, David A.'; Gonzalez, Hipolito

Bridge Number: 800-779-6447, Passcode: (b)(6)
Video: CNWRA

Guest Speaker:

slides-copy

Dr. Vincenzo Rondinella

European Commission, Joint Research Centre, Institute for Transuranium Elements (ITU), P.O.Box 2340, 76125 Karlsruhe, Germany

---

8:30 am to 9:00 am, meet with SFAS Director, Josephine Piccone

- annealing
- bubble - in doped
- 20, 40, 60 (doped / SNF) bench

Main Presentation, 9:00 to 10:00 am

"Safety of Spent Nuclear Fuel (SNF) after Discharge,"

including accelerated damage studies and SNF measurements.

- approximation ← studies
- displacement (point defects) early / He (bubbles) latter

Presentation/Exchange, 10:00 – 11:00 am

- Properties and Behavior of the High Burnup Structure and Data on Corrosion/Leaching of SNF and Analogues.
- SIMFUEL Dissolution Test Results (NRC/T Ahn, 20 minutes)

- stress measurements
- reactor - annealing fast

Presentation/Exchange, 11:00 am – 12:30 pm

- other properties ~ similar to SNF
- data (available / not available) answers

- reactor: annealing
- spins data
- examine (Japan / ...)

(next page)

deformable

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E314
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- An Overview of Some Ongoing EU Projects on Radionuclides Relevant to Performance Assessment and SNF Characterization; Severe Accident Studies
- Scoping of Options and Analyzing Risk (SOAR) Models for Waste Form Dissolution (NRC/T Ahn, 20 minutes)

Lunch, 12:30 pm

Notes:

(1) Time change
(2) New topic: severe accident
(3) Vincenzo will hear about two more NRC activities on cladding stress (he is a collaborator) and spent fuel drying on 4/29 during International High-Level Radioactive Waste Management Conference in the following week, Albuquerque.

E3 15

Two B5b Pumps

- Goodwin fire pump
  - 650 gpm

7F (b)(7)(F)

- Goodwin Model 130
  - 1300 gpm

7F (b)(7)(F)

Notes:

- Max small leak: ~ 250 gpm
- Max moderate leak: ~1900 gpm
- An Inject nozzle capacity: 500 gpm
- A spray nozzle capacity: 250 gpm

A5-3

Palmrose, Donald

From:	Spencer, Michael
Sent:	Friday, October 28, 2011 4:10 PM
To:	Palmrose, Donald
Cc:	Hart, Michelle; Brown, David; Muir, Jessie; Whited, Ryan; Clayton, Brent
Subject:	RE Info related to SFPs

OGC

Thanks, this is helpful. Long story short: Our FEIS references the section of the GEIS that discusses SFP accidents and concludes that impacts are SMALL.

Michael

From: Palmrose, Donald
Sent: Friday, October 28, 2011 3:56 PM
To: Spencer, Michael
Cc: Hart, Michelle; Brown, David; Muir, Jessie; Whited, Ryan; Clayton, Brent
Subject: Info related to SFPs

NRO

Michael,

As you mentioned in our prior phone call, Section 6.4.6 of the GEIS has info related to SFP accidents. From reading the text, it appears that the following documents formed the basis for the conclusion in the GEIS: 55 FR 38474 and NUREG-1092. I cannot download 55 FR 38474 from GAO's Federal Register website since they only go back to Volume 59 (1994). NUREG-1092 is related to ISFSIs so I would not try to pull up that document. If you have a way of getting 55 FR 38474, this is probably the best one to try reading. Also, if you can get a copy of this FR, please send Michelle and me a copy.

Michelle sent me an email that points to 73 FR 46204 on the denial of two petitions for rulemaking related to SFPs. This document is attached.

Finally, there is also an NRC Fact Sheet on "Reducing Hazards from Stored Spent Nuclear Fuel" at http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/reducing-hazards-spent-fuel.html This may only help to put into context past NRC actions.

Hope this helps,
Don

Don Palmrose
Sr. Project Manager
NRO/DSER/RAP3
301-415-3803
T7-F38

A5-4

original

Spitzberg, Blair

From:	Spitzberg, Blair	
Sent:	Monday, June 10, 2013 9:37 AM	
To:	Kellar, Ray	
Subject:	FW: Diablo Canyon Seismic Faults	
Attachments:	ser chpt15.pdf	
Tracking:	**Recipient**	**Read**
	Kellar, Ray	Read: 6/10/2013 9:37 AM

From: Ray Kellar
Sent: Thursday, March 19, 2009 9:58 AM
To: Shana Helton
Cc: Eric Benner; Blair Spitzberg
Subject: Diablo Canyon Seismic Faults

Hi Shana,

(b)(5)

Thanks,

Ray

Ray L. Kellar, P.E.
US NRC RIV
Inspector
Phone: 817-860-8164
Fax: 817-860-8188
Email: ray.kellar@nrc.gov

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15 ACCIDENT ANALYSIS

15.1 Conduct of Review

The staff evaluated the applicant's accident analysis by reviewing Chapter 8, "Accident Analysis," of the Diablo Canyon ISFSI SAR (Pacific Gas and Electric Company, 2003), documents cited in the SAR, and other relevant publicly available information, including web sites on the Internet.

In the ISFSI SAR and in its response to the staff's Request for Additional Information (RAI) (Pacific Gas and Electric Company, 2002), PG&E described the basis for selecting off-normal and accident events to ensure that all relevant potential scenarios have been considered. The selection of these off-normal and accident event scenarios is based on guidance in NUREG–1567 (U.S. Nuclear Regulatory Commission, 2000). In addition, PG&E also reviewed other site-specific applications and associated NRC evaluations in developing the spectrum of postulated events to be analyzed.

The dry cask storage system to be used at the proposed facility is the HI-STORM 100 System, which has been reviewed by the NRC and approved for general use under Certificate of Compliance (CoC) No. 1014-1 (U.S. Nuclear Regulatory Commission, 2002a). As discussed in Chapters 4 and 5 of this SER, the design-basis loads considered in the HI-STORM 100 System Final Safety Analysis Report (FSAR) bound the loading conditions at the proposed Diablo Canyon ISFSI. Thus, where applicable, the staff relied on the review carried out during the certification process for the HI-STORM 100 cask system, as documented in the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b).

The staff reviewed the accident analysis to determine if the following regulatory requirements have been met:

- 10 CFR §72.90 requires that: (a) site characteristics that may directly affect the safety or environmental impact of the ISFSI must be investigated and assessed; (b) proposed sites for the ISFSI must be examined with respect to the frequency and the severity of external natural and man-induced events that could affect the safe operation of the ISFSI; (c) design basis external events must be determined for each combination of proposed site and proposed ISFSI design; (d) proposed sites with design basis external events for which adequate protection cannot be provided through ISFSI design shall be deemed unsuitable for the location of the ISFSI; (e) pursuant to subpart A of Part 51 of Title 10 for each proposed site for an ISFSI, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and esthetic values; and (f) the facility must be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains.

- 10 CFR §72.92 requires that: (a) natural phenomena that may exist or that can occur in the region of a proposed site must be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important

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natural phenomena that affect the ISFSI design must be identified; (b) records of the occurrence and severity of those important natural phenomena must be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued; and (c) appropriate methods must be adopted for evaluating the design basis external natural events based on the characteristics of the region and the current state of knowledge about such events.

- 10 CFR §72.94 requires that: (a) the region must be examined for both past and present man-made facilities and activities that might endanger the proposed ISFSI. The important potential man-induced events that affect the ISFSI design must be identified; (b) information concerning the potential occurrence and severity of such events must be collected and evaluated for reliability, accuracy, and completeness; and (c) appropriate methods must be adopted for evaluating the design basis external man-induced events, based on the current state of knowledge about such events.

- 10 CFR §72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI be identified.

- 10 CFR §72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR §72.98(a) and §72.98(b) be investigated as appropriate with respect to: (1) the present and future character and the distribution of population, (2) consideration of present and projected future uses of land and water within the region, and (3) any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR §72.102(f)(1) requires that the design earthquake for use in the design of structures be determined as follows: (1) for sites that have been evaluated under the criteria of Appendix A of 10 CFR Part 100, the design earthquake must be equivalent to the safe shutdown earthquake for a nuclear power plant; and (2) Regardless of the results of the investigations anywhere in the continental U.S., the design earthquake must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.

- 10 CFR §72.106(b) requires that any individual located on or beyond the nearest boundary of the controlled area not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens of the eye dose equivalent shall not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or to any extremity shall not exceed 0.5 Sv (50 rem). The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.

- 10 CFR §72.122(b) requires that (1) structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents; and (2) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lighting, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect: (i) structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect: (A) appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (B) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. (ii) The ISFSI also should be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel, high-level radioactive waste or on to structures, systems, and components important to safety.

- 10 CFR §72.122(c) requires that structures, systems, and components important to safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

- 10 CFR §72.122(h)(1) requires that the spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

- 10 CFR §72.122(h)(4) requires that storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry spent fuel storage, periodic monitoring is sufficient provided that periodic monitoring is consistent with the dry spent fuel storage

cask design requirements. The monitoring period must be based upon the spent fuel storage cask design requirements.

- 10 CFR §72.122(h)(5) requires that the waste must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of 10 CFR Part 20 limits. The package must be designed to confine the high-level radioactive waste for the duration of the license.

- 10 CFR §72.122(i) requires that instrumentation and control systems must be provided to monitor systems that are important to safety over anticipated ranges for normal operation and off-normal operation.

- 10 CFR §72.122(l) requires that Storage systems must be designed to allow ready retrieval of spent fuel, high-level radioactive waste for further processing or disposal.

- 10 CFR §72.124(a) requires spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions.

- 10 CFR §72.128(a)(2) requires that spent fuel storage be designed with suitable shielding for radioactive protection under normal and accident conditions.

The proposed ISFSI facility must be sited, designed, constructed, and operated so the above-mentioned regulatory requirements are met to adequately protect public health and safety during all credible off-normal and accident events.

15.1.1 Off-Normal Events

The off-normal events are described in Section 8.1, "Off-Normal Operations," of the SAR. This section of the SER discusses results from the review of potential off-normal conditions, which include cask drop from less than design allowable height, partial vent blockage, and operational events. Where applicable, the staff relied on the analyses in the HI-STORM 100 System FSAR and the related staff evaluation as documented in the HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b).

15.1.1.1 Cask Drop Less Than Design Allowable Height

Due to the design features and administrative controls applied to the ISFSI-related activities conducted within the DCPP FHB/AB, a potential drop of the HI-TRAC 125 Transfer Cask is only

considered during the period that the loaded Transfer Cask is moved between the FHB/AB and the Cask Transfer Facility (CTF). Similarly, the drop of a loaded storage cask is only considered during movement between the CTF and the ISFSI storage pads. In its response to the staff's RAI (Pacific Gas and Electric Company, 2002), PG&E committed to design the cask transporter so it will have redundant drop protection features and will conform to the criteria of NUREG-0612 (U.S. Nuclear Regulatory Commission, 1980), American National Standards Institute (ANSI) N14.6 (American National Standards Institute, 1993), and ASME B30.9-1996 (ASME International, 1996). The staff previously determined that a specific limit on cask lift height during transfers between the FHB/AB, CTF, and the storage pads is not necessary if these cask transporter design requirements are met (U.S. Nuclear Regulatory Commission, 2002a). Therefore, based on the applicant's commitment to these design standards, transfer and storage cask drop events are not considered credible and an evaluation of a cask drop less than the design allowable height is not required.

15.1.1.2 Partial Vent Blockage

The staff previously determined that the HI-STORM 100 storage cask provides adequate heat removal capacity under partial vent blockage conditions, so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to and the environmental characteristics of the site are bounded by the corresponding design criteria (see Section 6.1.3 of this SER). The proposed Diablo Canyon ISFSI Technical Specifications include surveillance requirements for ensuring that the cask heat removal system is operational during storage (i.e., the air ducts are inspected every 24 hours to ensure that the ducts are free of blockages).

15.1.1.3 Operational Events

Failure of Instrumentation

No off-normal events that involve failure of instruments and control systems are postulated because the passive dry cask storage system does not rely on permanent instruments to monitor the heat and radiation at the ISFSI storage pad site. The HI-STORM 100 storage casks will be visually inspected as required by the Technical Specifications to ensure that the overpack inlet and outlet air ducts remain free from blockages. If a blockage is detected, it will be removed within one operating shift. Radiation and airborne radioactivity will be monitored using portable hand-held radiation protection instruments and dosimeters during transfer operations at the CTF and routine maintenance at the ISFSI storage area.

Based on the staff's review of the information provided regarding failure of instrumentation, there is reasonable assurance that important to safety functions will not be affected for the proposed cask system or the proposed ISFSI.

Vehicular Impact

The staff reviewed the information presented in the ISFSI SAR Chapters 3 and 4, "Principal Design Criteria," and, "ISFSI Design;" and Section 8.2.4, "Drops and Tip-Over." Vehicular impact is postulated by the staff to occur during movement of a loaded transfer cask from the FHB/AB to the CTF, or movement of a loaded storage overpack from the CTF to the storage

pads, or in the storage pad area. Vehicular impacts are postulated to result from an interaction between the cask transporter, an onsite service vehicle, or an off-site vehicle used by site personnel and a loaded transfer or storage cask. Equipment failure, operator error, or a natural event (e.g., tornado) may lead to this off-normal event. Occurrence of this event would be easily identifiable from visual evidence, such as dents or scratches on casks, onsite vehicles, and other ISFSI facility structures, systems, and components (SSC).

As discussed in the HI-STORM 100 System FSAR, the HI-STORM 100SA storage cask and HI-TRAC 125 Transfer Cask are designed to withstand a tornado missile equivalent to the impact of an automobile weighing 1,800 kg [3,968 lb] traveling at a speed of 202 km/h [56 m/s] {126 mph [185 ft/s]} (SAR Table 3.2-2). This tornado-missile analysis for the storage cask and the staff evaluation are provided in the HI-STORM 100 System FSAR and the related NRC SER. That analysis indicated that such impacts would not result in damage to the cask contents. Since onsite vehicles at the DCPP are assumed to be traveling at a much lower speed than that assumed in the tornado missile analysis, postulated vehicular impacts for the HI-STORM 100 System transfer and storage casks are bounded by the tornado missile analysis, and no damage to the spent fuel contents will result from these events.

The cask transporter and CTF are designed to withstand a tornado missile equivalent to the impact of an automobile weighing 1,800 kg [3,968 lb] traveling at a speed of 15 m/s {48.8 ft/s (33.3 mph)} (SAR Table 3.2-2) (Pacific Gas and Electric Company, 2003). The tornado missile analysis and the staff's evaluation are provided in Section 8.2.2 the ISFSI SAR and Section 15.1.2.10 of this SER, respectively. Onsite vehicles will generally be traveling at a much lower speed. Therefore, vehicular impacts for the cask transporter and CTF are also bounded by the tornado missile analysis.

The staff finds that potential vehicular impact will not impair the ability of the SSCs to maintain subcriticality, confinement, and sufficient shielding of the stored fuel.

Loss of Electrical Power

The staff reviewed the information presented in Section 8.1.6, "Loss of Electrical Power," of the SAR as an off-normal event. Total loss of external alternating current power is postulated to occur during the facility operations. The loss of electrical power at the Diablo Canyon ISFSI facility may occur because of natural phenomena, such as lightning or high winds, or as a result of failure of the electrical distribution system or equipment. A loss of electrical power will be detected through loss of functions of the electric-powered equipment.

No safety features required for lifting, upending, and lowering of the HI-TRAC 125 Transfer Cask, multi-purpose canister (MPC) and HI-STORM 100SA storage cask at the CTF will be affected by a loss of power, because these operations will be conducted by the cask transporter, which is driven by an on-board diesel engine. Similarly, the emplacement operations of a HI-STORM 100SA storage cask on the ISFSI storage pad location are also conducted using the cask transporter and do not rely on electric power from other onsite or offsite sources.

Electrical power is supplied through onsite sources to each of the three lifting screw jack motors and control systems that operate the CTF lifting platform. The CTF lifting platform will raise and lower the MPC during the transfer operation of the MPC from the HI-TRAC 125 Transfer Cask to a HI-STORM 100SA storage cask. In the event of a power loss during the operations of the lifting platform, all three screw jack motors will stop simultaneously to prevent a potential uncontrolled descent of the storage cask inside the CTF. The lift jacks will remain stopped and will require manual action to restart upon restoration of power. In the unlikely event of an extended period of power loss, the storage cask (including the MPC) will be raised to grade level from the CTF lifting platform within 22 hours using the cask transporter to ensure that short-term cladding temperature limits will not be exceeded.

No radiological impact is expected from a loss of electric power because there is no loss of MPC confinement during this off-normal event. In addition, the transfer cask is designed to provide adequate shielding and decay heat removal from the canisters. The operators would take measures to maintain adequate distance and additional shielding between themselves and the CTF to minimize exposure until power is restored and the transfer operation is resumed.

The staff concludes the applicant's evaluation of loss of electrical power as an off-normal event is adequate in providing reasonable assurance that Diablo Canyon ISFSI operations can be conducted without endangering the health and safety of the public.

Cask Transporter Off-Normal Operation

The staff reviewed the information provided in Section 8.1.7 of the SAR, "Cask Transporter Off-Normal Operation." The transporter with a loaded transfer cask will travel a distance of 1.9 km [1.2 mi] along the transporter route from the DCPP to the CTF and will take approximately 3.0 hours per transport. The transporter is also used in the transfer operation of an MPC from the HI-TRAC 125 Transfer Cask to a storage cask at the CTF and in the emplacement of storage casks on the ISFSI pads. The off-normal events from operation of the cask transporter could arise from driver error or incapacitation, transporter engine failure because of mechanical failure, or loss of hydraulic fluid in the hydraulic system. A support team will walk with the transporter and observe the driver and transporter movement. At the sight of driver distress or swerving of the transporter, the support personnel can stop the transporter using either of two stop switches located outside the transporter. The transporter is also equipped with automatic shutoff control to stop the vehicle in the event of incapacitation of the driver. The same control will also be used for emergency stops during the lifting operation at the CTF. Transporter engine failure would stop the vehicle or hydraulic brakes would engage to stop lifting operations. Hydraulic system failure would be detected by pressure instrumentation on the transporter, and any loss of hydraulic fluid will engage hydraulic brakes to stop lifting operations. The transporter is designed to operate in a "fail-safe" mode so any uncontrolled lowering of a transfer cask loaded with an MPC or storage cask is precluded.

Off-normal events associated with cask transporter operation are not expected to cause radiological dose as the confinement and shielding of spent nuclear fuel will not be affected.

The staff concludes that the applicant's assessment of cask transporter off-normal operation is adequate in providing reasonable assurance that Diablo Canyon ISFSI operations can be conducted without endangering the health and safety of the public.

15.1.1.4 Off-Normal Ambient Temperatures

The off-normal environmental temperature range for the Diablo Canyon ISFSI is -4.4 to 36.1 °C [24 to 97 °F]. This off-normal temperature range is bounded by the previously evaluated off-normal temperature ranges for the HI-STORM 100 storage casks and HI-TRAC 125 Transfer Cask. Specifically, the previously evaluated off-normal temperature range for the HI-STORM 100SA storage cask is -40 to 38 °C [-40 to 100 °F] and for the HI-TRAC 125 Transfer Cask, -18 to 38 °C [0 to 100 °F]. The staff previously determined that the HI-STORM 100SA storage casks and HI-TRAC 125 Transfer Cask designs provide adequate heat removal capacity during off-normal ambient temperature conditions so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The Diablo Canyon ISFSI Technical Specifications will ensure that the relevant conditions assumed in the previous analysis for the HI-STORM 100 system are also met for the Diablo Canyon spent fuel.

15.1.1.5 Off-Normal Pressures

Section 8.1.1.1 of the Diablo Canyon ISFSI SAR indicates that the off-normal pressure within the MPC, which is the sole pressure boundary for the HI-STORM 100SA storage cask, is evaluated considering a concurrent rupture of 10 percent of the stored fuel rods while exposed to off-normal ambient temperatures of 38 °C [100 °F]. Note that this off-normal temperature bounds the off-normal temperature for the proposed Diablo Canyon site (see Section 6.1.3 of this SER). The staff previously determined that the methodology used to assess this off-normal condition is acceptable and that there are no consequences that affect the public health and safety so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The Diablo Canyon ISFSI Technical Specifications will ensure that the relevant conditions assumed in the previous analysis for the HI-STORM 100 system are also met for the Diablo Canyon spent fuel.

15.1.2 Accidents

The ISFSI SAR includes a discussion of potential accidents resulting from both external natural and man-induced events at the proposed facility. Natural phenomena events are discussed in Chapter 2, "Site Characteristics" of the SAR. The staff's evaluation of those events is discussed in Chapter 2 of this SER. The accident analysis review focused on the effects of the natural phenomena and human-induced events on SSCs important to safety. Analytical techniques, uncertainties, and assumptions were examined. Each event was examined to ensure that it includes: (1) a discussion of the cause of the event, (2) the means of detection of the event, (3) an analysis of the consequences and the protection provided by devices or systems designed to limit the extent of the consequences, and (4) any actions required of the operator.

The Diablo Canyon ISFSI will use the HI-STORM 100 dry cask storage system. Where applicable, the staff relied on the analyses in the HI-STORM 100 System FSAR and the related staff evaluation as documented in the HI-STORM 100 System SER.

15.1.2.1 Cask Tip-Over

The staff has previously determined that cask tip-over events need not be considered for the approved HI-STORM 100SA system, based on the cask anchorage system used and the storage pad design specifications (U.S. Nuclear Regulatory Commission, 2002a,b). Sections 3.3.2 and 4.2.1.1of the ISFSI SAR (Pacific Gas and Electric Company, 2003) describe the cask anchoring system that will be used for the Diablo Canyon ISFSI, and this design also precludes the need for consideration of cask tip-over events. The staff's evaluation of the storage pad and anchorage system design can be found in Section 5.1.3 of this SER.

15.1.2.2 Cask Drop

Due to the design features and administrative controls applied to load handling activities in the FHB/AB, a potential drop of the loaded HI-TRAC 125 Transfer Cask is only considered during movement between the FHB/AB and the CTF. Similarly, a drop of a loaded HI-STORM 100SA storage cask is only considered during transport between the CTF and the ISFSI storage pads. In its response to the staff's RAI (Pacific Gas and Electric Company, 2002), PG&E committed to design the cask transporter so it will have redundant drop protection features and will conform to the criteria of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980), American National Standards Institute (ANSI) N14.6 (American National Standards Institute, 1993), and ASME B30.9-1996 (ASME International, 1996). Based on the applicant's commitment to these design standards, transfer and storage cask drop events are not considered credible. Therefore, a lifting height limit need not be specified for the loaded casks during movements between the FHB/AB, CTF and the storage pads, provided that all of these cask transporter design requirements are met.

15.1.2.3 Flood

The applicant has not considered flooding a credible accident at the Diablo Canyon ISFSI. As discussed in Section 2.1.4, "Surface Hydrology," of this SER, PG&E demonstrated that local natural and man-made drainage systems are sufficient to prevent flooding of the ISFSI pad site and CTF.

15.1.2.4 Fire and Explosion

<u>Fire</u>

The staff reviewed the information presented in Section 8.2.5, "Fire," of the ISFSI SAR. Additional information presented in SAR Sections 4.2.3.3.2.10, "Fire;" and 4.2.3.3.2.11, "Lightning," was also considered in this review.

Locations pertaining to the proposed ISFSI that fall within the purview of 10 CFR Part 72 review are the transport route from the DCPP FHB/AB to the CTF, within the CTF, and within the cask storage area. Credible fire accidents potentially affecting SSCs important to safety at the proposed facility identified by PG&E are:

(1) An onsite cask transporter fuel tank fire;

(2) Other onsite vehicle fuel tank fires;

(3) Combustion of other local stationary fuel tanks;

(4) Combustion of other local combustible materials;

(5) Fire in the surrounding vegetation; and

(6) Fire from mineral oil from the Unit 2 transformers.

Additional information and the staff's evaluation are provided in Section 6.1.5.1 of this SER.

The cask transporter will be used to move the spent nuclear fuel in an MPC from the FHB/AB to the CTF using the HI-TRAC 125 Transfer Cask. After the MPC has been transferred to the HI-STORM 100SA storage cask at the CTF, the cask transporter will be used to move the loaded storage cask onto the storage pad. To limit the potential exposure of the HI-TRAC 125 Transfer Cask and HI-STORM 100SA storage casks to a fire attributable to the transporter diesel fuel, the fuel tank used for the transporter will be limited to a 189-L [50-gal] capacity by the ISFSI Technical Specifications.

One postulated fire scenario for the CTF or the storage pads involves the diesel-fueled cask transporter with a 189-L [50-gal] fuel tank. The tank may rupture, resulting in the spilling and ignition of all of the diesel fuel. The ability of the HI-TRAC 125 Transfer Cask and HI-STORM 100SA storage casks to provide confinement and protect the spent nuclear fuel from gross degradation as the result of a 189-L [50-gal] diesel fuel fire was previously reviewed and found to be acceptable by the staff (U.S. Nuclear Regulatory Commission, 2002a,b), and these findings also apply to the Diablo Canyon ISFSI for this analyzed event.

As described in Section 8.2.5.2 of the ISFSI SAR, administrative controls will be implemented to ensure that transient sources of fuel in volumes larger than 189 L [50 gal] will be at a sufficient distance away from the ISFSI storage pads at all times, the CTF during active MPC transfer operations, and the transport route during cask transfer. There is at least a 30.5-m [100-ft] clearance between the storage area, CTF, or the cask transport route, and any onsite stationary fuel tanks, as described in SAR Section 2.2.2.2.

In its response to NRC additional questions on supplemental blasts and explosions (Pacific Gas and Electric Company, 2003b), PG&E indicated that a 3,028-L [800-gal] gasoline tanker truck will use the transport route near the storage area to deliver fuel to the vehicle maintenance shop located approximately 610 m [2,000 ft] northeast of the storage area six times a week. The tanker truck transport route passes by the storage casks on the north side of the proposed dry storage area. To determine the potential consequences of a gasoline tanker truck fire occurring near the proposed storage facility, a bounding 7,570-L [2,000-gal] fire loading analysis was conducted to assess the potential effects on the HI-TRAC 125 Transfer Cask, which bounds the potential effects on a HI-STORM 100SA storage cask (Pacific Gas and Electric Company, 2003a). This fire loading analysis adequately demonstrated that a nonengulfing 7,570-L [2,000-gal] fuel tanker fire will not adversely affect the HI-TRAC 125 Transfer Cask or a HI-STORM 100SA storage cask at the Diablo Canyon ISFSI.

Onsite stationary fuel sources include:

(1) Three fuel tanks {946 L [250 gal] of propane, 7,571 L [2,000 gal] of No. 2 diesel, and 11,356 L [3,000 gal] of gasoline} located beside the main plant road, 366 m [1,200 ft] from the cask transport route at its nearest point; and

(2) The Unit 2 main bank transformers filled with mineral oil.

The separation distance between the three stationary fuel tanks and the transport route is 366 m [1,200 ft]. Because of the separation distance, radiation is the only mechanism through which released heat would be transferred to the cask. The surface area of a hemisphere with a 366-m [1,200-ft] radius is in excess of 836,131 m^2 [9×10^6 ft^2]. The projected area of the cask is approximately 20 m^2 [220 ft^2]. Therefore, only 0.0025 percent of the total heat energy released simultaneously from these tanks would be directed toward a single cask. This is a small amount of energy, and consequently, a fire in the transporter fuel tank would be bounding.

The potential for a fire within the CTF as the result of a cask transporter or gasoline tanker truck fuel spill was addressed in response to additional NRC questions (Pacific Gas and Electric Company, 2003c). To mitigate the potential effects of these postulated fire events, the transporter will be designed with a removable fuel tank, and the CTF opening will be located at a higher elevation than the surrounding area so any fuel spilled will flow away from the facility. Moreover, administrative controls will prohibit any transient fuel sources beyond that of the cask transporter from coming into close proximity of the CTF during transfer operations.

Vegetation surrounding the storage pad area is primarily grass with no significant brush or trees (Pacific Gas and Electric Company, 2003). A potential fire in the vegetation may be started by an offsite fire spreading onto the proposed site or by a lightning or a transmission line strike. As discussed in Section 8.2.5.2 of the SAR, "Accident Analysis" (Pacific Gas and Electric Company, 2003), no combustible materials will be stored within the security fence of the proposed facility at any time. A walk-down of the general area and the transport route will be conducted prior to any loaded cask transport to ensure that all combustible materials are controlled according to the administrative procedures. PG&E will implement a maintenance program to prevent uncontrolled growth of vegetation surrounding the storage area.

PG&E submitted an analysis of potential effects of wildfires on the HI-STORM 100SA storage casks (Holtec International, 2001a). This analysis evaluated two scenarios: (1) no wind and (2) 24-km/hr [15-mph] wind in the uphill direction. Although it is expected that facility personnel will try to suppress or control the fire quickly, it is postulated that no fire fighting activities occur. Using simulation codes FARSITE and FLAMMAP, Holtec International developed the values for the parameters necessary to describe the wildfire characteristics (namely, fire intensity, rate of spread, and flame length)(Pacific Gas and Electric Company, 2003).

There will be a minimum of a 15.2-m [50-ft] gap between the storage pads and the security fence on the north side of the proposed facility. The gap will be at least 12.2 m [40 ft] on the other three sides. The restricted area fence surrounds the area protected by the security fence and is approximately 30.5 m [100 ft] from the storage pads. Holtec International (2001a) assumed that the area within the proposed storage facility nuisance fence would be covered with either gravel or concrete. Therefore, the area surrounding the storage pads would be

covered with noncombustible materials, which will not only act as a barrier for progression of wild fires but also will not add any additional fuel to the fire.

Electrical transformers are located approximately 73 m [240 ft] from the transporter route. The mineral oil within these transformers could be ignited by lightning strike, vehicle crash, or internal electrical faults (Pacific Gas and Electric Company, 2003c). Administrative procedures will prohibit movement of the loaded transporter during inclement weather. Additionally, DCPP transition operations significantly reduce the potential for transformer mineral oil being ignited by lightning or internal electric faults. Each active transformer has a fire-suppression system that will activate in case of a fire. Administrative procedures will also prohibit use of onsite vehicles during transporter operation, negating the potential of a vehicle accident initiating a transformer fire. Moreover, even if a transformer mineral oil fire were to occur, its effect on the transfer cask during transport would be bounded by the nonengulfing 7,570-L [2,000-gal] fire-loading analysis.

The staff reviewed the information provided by the applicant regarding potential onsite fires and wildfires at the proposed facility. The staff found the applicant's analysis acceptable because:

- Through design and administrative procedures, potential fire events will be minimized for the CTF.
- The storage casks are designed to withstand a fire from 50 gallons of diesel fuel in the fuel tank of the cask transporter.
- Both the transfer and storage casks will be able to withstand a nonengulfing 7,570-L [2,000-gal] fuel fire.
- Adequate analysis was presented about potential effects of the tanker truck fire on storage casks sitting on the pads.
- The area surrounding the storage pads will be covered with noncombustible materials.

Onsite and Offsite Explosion

The staff has reviewed the information presented in SAR Sections 2.2.2.3, "Onsite Explosion Hazards"; 8.2.6, "Explosion"; and 3.3.1.6, "Fire and Explosion Protection." In addition, the staff also reviewed analyses of potential explosion events in Holtec International (2002) and PG&E Calculation No. PRA01-01, "Risk Assessment of Dry Cask/Spent Fuel Transportation Within the DCPP Owner Controlled Area," (Pacific Gas and Electric Company, 2003c). Potential sources of explosions within the proposed facility include:

(1) Detonation of a transporter or onsite vehicle fuel tank

(2) Detonation of a 3,028-L [800-gal] tanker truck while transporting fuel near the storage pad

(3) Detonation of a propane bottle transported past the ISFSI storage pad

(4) Detonation of an acetylene bottle transported past the ISFSI storage pad

(5) Explosive decompression of a compressed gas cylinder

(6) Detonation of large stationary fuel tanks in the vicinity of the transport route

(7) Detonation of the bulk hydrogen storage facility

(8) Detonation of acetylene bottles stored on the east side of the cold machine shop

Important to safety SSCs that are required to function after an explosion event include the storage casks, the transportation casks, the transporter, and the CTF. Regulatory Guide 1.91 (U.S. Nuclear Regulatory Commission, 1978) provides an acceptable methodology to estimate the minimum separation distance between an explosion source and a structure so that the peak positive incident overpressure would be less than 6.9 KPa [1 psi]. If the minimum separation distances calculated by following the suggested methodology of Regulatory Guide 1.91 are not sufficiently large to allow a conclusion that the peak positive incident overpressure would be less than 6.9 kPa [1 psi], an analysis of the frequency of hazardous materials shipment may be used to show the associated risk is sufficiently low. If the hazardous materials are shipped by more than one transportation mode, the frequency of exposure for the modes should be summed. Regulatory Guide 1.91 also states that potential explosion hazards can be screened out if, based on realistic or best estimate bases, an exposure rate less than 10^{-7} per year can be demonstrated. If conservative estimates are used, an exposure rate less than 10^{-6} per year is sufficiently low.

Regulatory Guide 1.91 sets 6.9 kPa [1 psi] as the peak positive incident overpressure below which no significant damage to the structures would be expected to result from an explosion. Explosion-induced ground motions are bounded by the earthquake criteria. Similarly, effects of explosion-generated missiles would be bounded by those associated with the air overpressure levels if the threshold air overpressure from any explosion source is kept below 6.9 kPa [1 psi], based on Regulatory Guide 1.91.

A potential explosion event can affect (1) canister transfer operation at the CTF, (2) storage casks placed on the pads, and (3) the transfer cask moved by the transporter from the FHB/AB to the proposed facility. Potential sources of explosive materials that may affect the storage casks and the canister transfer operation are (1) detonation of the transporter or onsite vehicle fuel tank, (2) detonation of a 3,028-L [800-gal] tanker truck while transporting gasoline past the ISFSI storage pads, (3) detonation of a propane bottle transported past the ISFSI storage pads, (4) detonation of an acetylene bottle transported past the ISFSI storage pads, (5) detonation of large stationary fuel tanks, and (6) an explosive decompression of a compressed gas cylinder. Other sources are far away from the proposed storage site and contain sufficiently small amounts of explosive materials such that they do not pose a credible hazard to the storage casks and canister transfer operations. A transfer cask loaded on a transporter could be affected by (1) detonation of the fuel tank of the transporter or an onsite vehicle (including the potential explosion of a parked vehicle fuel tank), (2) explosion of large stationary fuel tanks in the vicinity of the transport route, (3) explosion of the Bulk Hydrogen Storage Facility, and (4) explosion of acetylene bottles stored on the east side of the cold machine shop. Explosion of the mineral oil in the Unit 2 main bank transformers was determined to be a non-credible scenario.

Transporter and/or Onsite Vehicle Fuel Tanks

Potential sources of explosion considered for the Diablo Canyon ISFSI accident analyses include the fuel tanks of the onsite transporter or other onsite vehicles, including 3,028-L [800-gal] gasoline tanker trucks (Pacific Gas and Electric Company, 2003). The maximum capacity of the fuel tank of the onsite transporter is 189 L [50 gal] of diesel fuel. The average capacity of the fuel tank of any onsite vehicle is 76 L [20 gal](Pacific Gas and Electric Company, 2003c). A 3,028-L [800-gal] capacity gasoline tanker truck will use the onsite road near the storage pads on its way to and from the maintenance shop, located approximately 666 m [2,000 ft] northeast of the storage pads. PG&E will impose administrative controls to prevent a 15,142-L [4,000-gal] fuel truck from passing near the proposed storage facility at any time, and to also prevent it from entering the owner-controlled area at all while spent nuclear fuel is being transferred from the FHB/AB to the storage pads (Pacific Gas and Electric Company, 2003c).

Detonation of the fuel tank of a transporter and/or an onsite vehicle could potentially occur near the storage pads, CTF, and transport route. These events have been analyzed by PG&E, as they could potentially affect the storage cask, the transfer cask, or the structure of the CTF.

In its analyses, PG&E assumed that a minimum distance of 15m [50 ft] will be maintained between the source of explosion and the nearest storage cask because:

- No gasoline-powered vehicles will be allowed within the restricted area of the proposed facility; and
- A minimum distance will be maintained between the storage casks and the protected area fence at the north side of the proposed facility.

The flash point of diesel fuel is 51.7 °C [125 °F]. Based on the Fire Protection Association Handbook (National Fire Protection Association, 1997), the flash point of a liquid must be less than 37.8 °C [100 °F] to be classified as a flammable liquid. Therefore, diesel in the fuel tank of a transporter does not pose a credible explosion hazard.

Regulatory Guide 1.91 provides a methodology to estimate the exposure rate r:

$$r = n \cdot f \cdot s \tag{15-1}$$

where,

n	–	explosion rate (per mile)
f	–	frequency of shipment (per year)
s	–	exposure distance (miles)

Based on data from the National Highway Traffic Safety Administration of the U.S. Department of Transportation, a total of 6,323,000 crashes involving all types of motor vehicles took place in 2001 (U.S. Department of Transportation, 2003a). Additionally, approximately 4,450,339 million km [2,781,462 million mi] were traveled in that year by all types of vehicles. Therefore, the vehicle involvement rate would be 227 per 160 million km [100 million mi] of travel. Based on 2001 crash statistics compiled by U.S. Department of Transportation, approximately 30 percent of all vehicle crashes constitute a single-vehicle crash. Additionally, approximately 30 percent

of all single-vehicle crashes took place at a speed below 48 km/hr [30 mph]. Moreover, approximately 0.1 percent of all vehicle crashes resulted in a fire.

PG&E, through administrative controls, will prevent any vehicle from passing another within the setback distance of 52.5 m [175 ft] from the proposed facility (Pacific Gas and Electric Company, 2003c). Consequently, only a single-vehicle accident needs to be considered further. This setback distance was selected so that the resulting air overpressure from an exploding 76-L [20-gal] gasoline tank would be 6.9 kPa [1 psi]. Additionally, PG&E will use administrative controls to prevent any motor vehicles from exceeding the speed limit of 40 km/hr [25 mph] in the area of the proposed facility (Assumption 7). Therefore, the frequency of vehicle fire has been estimated by PG&E to be 3.26×10^{-10} per km [2.04×10^{-10} per mi]. Assuming conservatively that every vehicle fire leads to an explosion, the explosion rate of vehicle fire, n, would be 3.26×10^{-10} per km [2.04×10^{-10} per mi].

The exposure distance, s, is the distance along the road within the setback region of the proposed facility from which the storage casks would have the potential to receive an air overpressure greater than 6.9 kPa [1 psi]. This distance is estimated to be approximately 90 m [300 ft]. As stated by PG&E (Assumption 10), a maximum of 140 gasoline-powered vehicles would pass by the proposed facility in a day. Consequently, approximately 51,100 times in a year all items important to safety at the proposed facility would be exposed to the explosion hazard from passing gasoline-powered motor vehicles. Therefore, the annual frequency of exposure, r, is

$$r = 2.04 \times 10^{-10} \times 51100 \times \frac{300}{5280} = 5.92 \times 10^{-7} \text{ per year} \tag{15-2}$$

The staff concludes that the annual frequency of occurrence of a transporter and/or onsite vehicle fuel tank explosion was estimated in a conservative manner.

Parked Vehicle Fuel Tanks

PG&E used a probabilistic analysis to estimate the annual frequency of explosion of an on-site vehicle, parked in the power plant parking lots, that may have a potential to damage a transfer cask being hauled by the transporter on the transport route. Since the start of construction of DCPP 30 years ago, there has never been an explosion of a parked car, although one parked car caught fire. PG&E considers this an incredible scenario as, by administrative procedures, walk-downs of the parking lots would be performed looking for any explosion hazards, such as gasoline leaking from a vehicle, before a loaded transporter passes by. Additionally, administrative and physical controls would prevent movement of any vehicle within 52.5 m [175 ft] of the transporter.

PG&E conducted a search for industry information regarding the frequency of explosion of parked vehicles; however, no data have been found. Although administrative and physical controls would make an explosion of a parked car an incredible scenario; nevertheless, PG&E conducted an analysis to estimate the magnitude of the potential hazard. An analysis of gasoline-powered moving vehicles estimated the frequency of fire (and explosion) to be 3.26×10^{-10} per km [2.04×10^{-10} per mi], based on a single-vehicle crash. Since any cars parked within 52.5 m [175 ft] of the moving loaded transporter would not be allowed to

move, reduction of one order of magnitude in the explosion rate to 3.26×10^{-11} per km [2.04×10^{-11} per mi] would be reasonable.

The transporter carrying a HI-TRAC Transfer Cask will make eight trips per year from the protected area of the power plant to the proposed storage facility. Therefore, frequency, *f*, would be 8/yr. The exposure distance, *s*, is estimated to be 333 m [1,000 ft]. Assuming a maximum of 200 vehicles would be within the setback distance of 52.5 m [175 ft] at any moment while the transporter is moving, the annual frequency of exposure, *r*, is

$$r = 2.04 \times 10^{-11} \times 200 \times 8 \times \frac{1{,}000}{5{,}280} = 6.18 \times 10^{-9} \text{ per year} \qquad (15\text{-}3)$$

The staff concludes that the annual frequency of occurrence of a parked vehicle fuel tank explosion was estimated in a conservative manner.

<u>3,028-L [800-Gal] Tanker Truck While Transporting Fuel Near the Storage Pad</u>

PG&E performed a probabilistic risk analysis (Pacific Gas and Electric Company, 2003c) to estimate the annual frequency of the potential explosion hazard from the 3,028-L- [800-gal] gasoline tanker truck while passing near the proposed storage pads. Based on the U.S. Department of Transportation (2003a,b) statistics for large trucks, 429,000 crashes took place in 2001 with approximately 334,721 million km [207,686 million mi] of travel. Therefore, the involvement rate for large trucks would be 207 per 161 million km [100 million mi].

Single-vehicle accident data compiled by the U.S. Department of Transportation show that a total of 96,000 of the crashes involved a single vehicle, which is approximately 22 percent of all large truck crashes. Additionally, approximately 31 percent of these crashes took place at a speed below 48 km/hr [30 mph]. Moreover, approximately 0.5 percent of all large truck crashes resulted in fires (U.S. Department of Transportation, 2003a).

PG&E committed to prevent any vehicle from passing the tanker truck within 180 m [600 ft] of the proposed facility when the tanker truck is in motion (Assumption 8), so that only single vehicle crashes need to be considered in the analysis. The setback distance is calculated using the methodology given in Regulatory Guide 1.91, so that the air overpressure experienced by any safety-related SSCs from an accidental explosion of the gasoline tanker truck would be a maximum of 6.9 kPa [1 psi]. Additionally, administrative controls would prevent any vehicle movement at a speed greater than 40 km/hr [25 mph] within the setback region from the proposed facility (Assumption 7).

Assuming that the gasoline tanker will explode if caught on fire, PG&E estimated that the frequency of tanker explosion would be

$$207 \times 0.22 \times \frac{0.31}{100 \times 10^{6}} \times 0.005 = 7.06 \times 10^{-10} \text{ per mile} \qquad (15\text{-}4)$$

The exposure distance, *s*, is estimated to be 690 m [2,300 ft] based on a 180-m [600-ft] exclusion area from the nearest cask in the proposed facility. Assumption 5 states that the tanker truck would pass by the proposed facility six times in each week. Therefore, the annual

frequency of shipment, f, is 312. Using Regulatory Guide 1.91, the estimated exposure rate, r, is:

$$r = 7.06 \times 10^{-10} \times 312 \times \frac{2{,}300}{5{,}280} = 9.59 \times 10^{-8} \text{ per year} \tag{15-5}$$

The staff concludes that the annual frequency of occurrence of an explosion of the 3,028-L [800-gal] gasoline tanker truck while using the transport route near the proposed storage pads was estimated in a conservative manner.

Propane and Acetylene Bottles Transported Past the Storage Pad

The maintenance facility east of the proposed ISFSI uses acetylene for the cutting torch and propane to run forklifts. One acetylene bottle is the maximum required in 1 year. The forklift uses a 25.5 L [7 gal] liquefied propane bottle which is replaced at a maximum frequency of once per week. Through the use of administrative controls, PG&E will ensure that all compressed gas bottles transported past the proposed ISFSI are appropriately secured in the transporting vehicle in the upright position (Pacific Gas and Electric Company, 2003c, Assumption 19).

In analyzing this explosion event, PG&E considered that the bottle containing 25.5 L [7 gal] of liquefied propane may rupture while being transported past the proposed ISFSI, releasing the compressed gas. The propane could subsequently mix with air and the resulting vapor cloud could detonate, which could generate an air overpressure that could be damaging to the storage casks. For this event, Holtec International (2001b) and PG&E (2003) assumed that the minimum distance between the point of explosion and the storage casks would be the distance between the storage pads and the ISFSI security fence, because no combustible materials would be permitted inside the proposed ISFSI. The detonation of 26.5 L [7 gal] of propane is equivalent to 4.7 kg [10.37 lb] of trinitrotolune (TNT). At a distance of 15 m [50 ft], the resulting air overpressure would be 16.9 kPa [2.45 psi] (Holtec International, 2001b). Similar calculations performed by Holtec International for transport of the acetylene bottles, which contain smaller quantities of compressed gas, resulted in an estimated overpressure of 8.2 kPa [1.19 psi]; therefore, the postulated explosion of a propane bottle is the bounding event. PG&E asserted that because the HI-STORM 100SA storage casks are designed to perform satisfactorily under 68.9 kPa [10 psi] of air overpressure for a duration of 1 second, accidental detonation of a propane or an acetylene tank while being transported past the proposed facility would not damage the storage casks placed on the pad (Pacific Gas and Electric Company, 2003; Holtec International, 2001b). However, this overpressure level is greater than the recommended air overpressure limit of 6.9 kPa [1 psi] of Regulatory Guide 1.91; therefore, PG&E conducted a probabilistic risk analysis (Pacific Gas and Electric Company, 2003c) to estimate the annual exposure frequency of SSCs important to safety to a higher air overpressure level.

In its analysis, PG&E postulated that the motive force required for a compressed-gas bottle to fail or explode would be from a vehicle crash. Because the crashes near the proposed ISFSI are assumed to be only single-vehicle incidents, PG&E used an explosion rate, n, of 7.06×10^{-10} per mile, estimated for large truck crashes. Additionally, the frequency of bottle shipment, f, is assumed to be four times a week or 208 times a year to be conservative. The

exposure distance, s, is assumed to be 690 m [2,300 ft], the same as with the tanker truck crash. Therefore, the estimated exposure frequency, r, is

$$r = 7.06 \times 10^{-10} \times 208 \times \frac{2,300}{5,280} = 6.39 \times 10^{-8} \text{ per year} \tag{15-6}$$

Although pressurized gas bottles may also fail along the welded seam, the bottles are required to meet the current industry standards. Therefore, this mode of failure of gas bottles was not considered credible.

The staff concludes that the annual frequency of occurrence of an explosion of the propane and acetylene bottles transported past the storage pads was estimated in a conservative manner.

Compressed Gas Cylinders

Cylinders containing compressed acetylene, air, argon, helium, nitrogen, oxygen, and propane gases are stored inside the reactor-controlled area. Internal pressure of the compressed gas cylinders can be in excess of 13.8 MPa [2,000 psi]. The potential energy of the stored cylinders at such high pressures could have significant effects during a rupture because this potential energy would be released as kinetic energy that could potentially damage SSCs important to safety. PG&E postulated that these compressed gas cylinders may be damaged in a way that the valve assembly at the top of the cylinders is broken. This failure would create a hole, approximately 5 cm [2 in] in diameter, at one end of the cylinder. Gases escaping through this hole would impart a large acceleration to the cylinder body and/or the valve assembly. The cylinders and/or the valve assemblies could accelerate toward the cask systems resulting in impacts (Holtec International, 2001b).

One function of both HI-TRAC 125 Transfer Cask and HI-STORM 100SA storage casks is to prevent any missiles (e.g., gas cylinder body and valve assembly) from affecting the MPC. Based on the calculations performing by Holtec International (2001b), any missile impacting the HI-TRAC 125 Transfer Cask must penetrate a minimum of 3.8 cm [1.5 in] of steel before impacting the confinement boundary of the MPC. Similarly, any missile has to penetrate at least 5 cm [2 in] of steel before impacting the MPC for the HI-STORM 100SA storage cask neglecting the presence of the concrete overpack. Holtec International (2001b) estimated the maximum velocity of all ruptured gas cylinders using the bounding discharge coefficient so that the estimated acceleration and the resulting force are maximum, and, therefore, the depth of penetration in a steel plate would be maximum.

The maximum depth of penetration by the gas cylinder body occurs with propane gas and is equal to 0.59 cm [0.232 in]. The valve assembly produces a penetration of 0.61 cm [0.241 in]. Therefore, the maximum depth of penetration for all types of cylinders and the valve assemblies is substantially less than the steel thickness available to resist penetration. Consequently, there is reasonable assurance that no SSCs important to safety will be damaged from accidental rupture of compressed gas cylinders.

Stationary Fuel Tanks Near the Transport Route

Three large stationary fuel tanks are located approximately 360 m [1,200 ft] from the transport route at the closest point to the proposed ISFSI. These tanks include a 946-L [250-gal] propane tank, a 7,571-L [2,000-gal] diesel fuel tank, and an 11,356-L [3,000-gal] gasoline tank. These three fuel tanks are located close enough to each other so that an explosion of one tank could cause potential rupture of the other two tanks. Diesel fuel does not present an explosion hazard because of its high flash point. While a rupture and subsequent detonation of either the propane tank or the gasoline tank could potentially rupture the diesel fuel tank, the spilled diesel fuel would burn without exploding. Consequently, the stored diesel fuel would not contribute to the explosion overpressure. Therefore, this event is limited to the near-simultaneous explosion of both the propane and gasoline tanks to generate any incident air overpressure. An explosion of these tanks may potentially affect the canister transfer operations at the CTF, the storage casks placed on pads, or the loaded transfer cask en route to the CTF.

Holtec International (2001b) estimated the air overpressure from a simultaneous explosion of 946 L [250 gal] of propane and 11,356 L [3,000 gal] of gasoline. These sources are equivalent to 53.27 kg [117.33 lb] of TNT, which generates an air overpressure of 5.79 kPa [0.84 psi] at a distance of 366 m [1,200 ft], the minimum distance between the stationary fuel tanks and the transport route (Pacific Gas and Electric Company, 2003). Based on Regulatory Guide 1.91 , an air overpressure of 5.79 kPa [0.84 psi] would not cause damage to any safety-related structures. The ISFSI security fence and the CTF are further away from the storage tanks than the closest point on the transport route. Therefore, it is expected that the air overpressure at these locations will be lower than 5.79 kPa [0.84 psi].

The stationary fuel tanks are more than 805 m [0.5 mi] from the proposed storage pad location and at an elevation of approximately 61 m [200 ft] below. These tanks are located southwest of the proposed facility with prevailing southeastern wind directions. Therefore, the winds would normally take the vapor cloud south of the proposed facility. Additionally, the vapor cloud generated at the fuel tank location needs to climb the 61-m [200-ft] hill to reach the proposed facility. Moreover, there is a major cut in the hillside directly above and east of the tanks. This cut would likely channel the vapor cloud away from the proposed facility. Therefore, there is reasonable assurance that any vapor cloud generated at these stationary tanks would not pose any undue hazard to the proposed facility.

The stationary fuel tanks will be periodically filled by standard fuel tankers with a capacity between 11,356 to 15,142 L [3,000 to 4,000 gal]. During any spent fuel transfer operation, the filling of these tanks would be suspended and all vehicle movements will be administratively controlled in accordance with the Cask Transportation Evaluation Program in the Diablo Canyon ISFSI Technical Specifications. Additionally, Section 8.2.6 of the SAR states that administrative controls will be used to ensure that the air overpressure received by any safety-related structures from an explosion of a tanker truck would be less than the 6.9-kPa [1-psi] limit.

Bulk Hydrogen Storage Facility

A bulk hydrogen facility is located approximately 4.5 m [15 ft] from the transport route from where the loaded transfer casks enter and leave the FHB/AB of Unit 1 of the DCPP. This facility contains 6 hydrogen tanks with a total capacity of approximately 8,495 L [300 ft^3]. These

tanks are refilled approximately twice a month and are kept in a seismic-qualified rack enclosed in a seismic-qualified vault. The vault has a 0.3- [12-in] diameter top vent to ensure that no leaked gas builds up. The vault only opens toward the FHB/AB. The hydrogen facility is designed against excessive flow, overpressurization, and vehicle damage during refilling. Therefore, it is extremely difficult to accumulate significant quantities of loaded gas leading to an explosion.

The Electric Power Research Institute Fire Events database considers hydrogen fire to be a credible event and provides a frequency of 3.2×10^{-3} per year (Pacific Gas and Electric Company, 2003c). Therefore, the hourly frequency of fire at the bulk hydrogen facility is estimated to be $3.2 \times 10^{-3}/8760$, or 3.7×10^{-7}. Because the design of the facility prevents accumulation of leaked hydrogen gas in confined spaces, it is extremely difficult to have an explosion even in the case of a hydrogen fire. PG&E assumed that in 10 percent of the cases, a hydrogen fire would lead to an explosion in the bulk hydrogen facility and, therefore, the estimated hourly frequency of hydrogen explosion would be $3.7 \times 10^{-7} \times 0.1$, or 3.7×10^{-8}.

PG&E states that the loaded cask transporter would be in the vicinity of the hydrogen tanks for less than 1 hour during each spent fuel transfer from the FHB/AB to the storage pad (Assumption 14), and there will be eight spent fuel transfers each year (Assumption 1). To add further conservatism, PG&E assumed a yearly exposure of 10 hours. Therefore, the annual exposure frequency of the transfer cask to a potential hydrogen tank explosion would be

$$3.7 \times 10^{-8} \times 10 = 3.7 \times 10^{-7} \text{ per year} \qquad (15\text{-}7)$$

The staff concludes that the annual frequency of occurrence of an explosion of the bulk hydrogen storage facility having an impact on a loaded transfer cask was estimated in a conservative manner.

Acetylene Bottles Stored on the East Side of the Cold Machine Shop

A maximum of 10 acetylene bottles are stored on the east side of the cold machine shop near the DCPP. This facility is more than 7.5 m [25 ft] from the transporter route and is protected by concrete block walls on two sides. The third side is protected by a building. Administrative procedures ensure that these bottles are restrained in an upright position because of seismic considerations. This restraint ensures that no potential missiles, originated from an exploding bottle, would be aimed at the transporter route. Furthermore, the cold machine shop facility location allows limited access of vehicles. Additionally, administrative procedures will control any vehicle movement within 52.5 m [175 ft] of the transporter route when the transporter is hauling a loaded transfer cask. Therefore, there would be no motive force available to initiate damage to the gas bottles leading to an explosion at those times. Consequently, PG&E concluded that accidental detonation of acetylene bottles stored on the east side of the cold machine shop would not be a credible hazard to any safety-related SSC for the proposed ISFSI.

Mineral Oil from Diablo Canyon Power Plant Unit 2 Main Bank Transformers

There are six transformers on the Unit 2 side of the DCPP: three single-phase 500-kV, two three-phase 25-kV, and one three-phase 12-kV. Additionally, two spare transformers are

stored adjacent to the active transformers. The three single-phase 500-kV transformers are located approximately 240 feet from the closest point to the transport route. The other transformers are mostly shielded from the transport route by these 500-kV transformers because of the layout with respect to the transport route. Each active transformer has a fire-suppression system that will activate in case of a fire.

The mineral oil in the transformers acts as a coolant. It has a flash point of 135 °C [275 °F]. Therefore, an explosion of mineral oil does not pose a significant hazard (Holtec International, 2001b) because this is not a flammable liquid. To be classified as a flammable liquid, the flash point of the liquid should be less than 37.8 °C [100 °F] (National Fire Protection Association, 1997). Although an electrical fault may occur within one of the transformers, the resulting rupture of the transformer case may ignite and burn the mineral oil, but the mineral oil would not explode. Therefore, a potential explosion of the mineral oil at Unit 2 of DCPP was not considered a credible hazard for ISFSI operations.

Summary of Review

The potential explosion hazards that may affect the storage casks or the cask or canister transfer operations are: (1) detonation of the transporter or onsite vehicle fuel tank, (2) detonation of 3,028-L [800-gal] tanker truck while transporting gasoline past the ISFSI storage pads, (3) detonation of a propane bottle transported past the ISFSI storage pads, (4) detonation of an acetylene bottle transported past the ISFSI storage pads, (5) detonation of large stationary fuel tanks, and (6) an explosive decompression of a compressed gas cylinder. Other sources are far away from the proposed storage site and contain sufficiently small amounts of explosive materials to not pose a credible hazard to the storage casks and cask and canister transfer operations. A transfer cask loaded on a transporter could be affected by: (1) detonation of the fuel tank of a transporter or an onsite vehicle fuel tank (including the potential explosion of a parked vehicle), (2) explosion of large stationary fuel tanks in the vicinity of the transport route, (3) explosion of the Bulk Hydrogen Storage Facility, and (4) explosion of acetylene bottles stored on the east side of the cold machine shop. The Diablo Canyon ISFSI Technical Specifications will include requirements for a Cask Transportation Evaluation Program, which will specify administrative controls to prevent movement of the tanker truck and any onsite vehicles during transporter operation. Similarly, no acetylene or propane bottles will be transported during transporter operations. Decompression of compressed gas cylinders does not pose an air overpressure hazard; missiles generated by the decompression of the cylinders are the primary concern in this situation.

PG&E conducted a probabilistic risk analysis (Pacific Gas and Electric Company, 2003c) of the remaining explosion hazards that have a potential to cause damage to safety-related structures at the proposed facility. Based on the previous discussion, the annual frequency of exposure to explosion hazards of the storage casks placed on the storage pads at the ISFSI and the canister transfer operation at the CTF is:

$$P_1 = P_{onsite\ vehicle} + P_{propane/acetylene} + P_{tanker\ truck} + P_{stationary\ tanks}$$

$$or,\ P_1 = 5.92 \times 10^{-7} + 6.39 \times 10^{-8} + 9.59 \times 10^{-8} + 0 = 7.52 \times 10^{-7} \text{ per year} \qquad (15\text{-}8)$$

Similarly, the annual frequency of exposure to explosion hazards of the transfer cask while being transported by the transporter is:

$$P_2 = P_{onsite\ vehicle} + P_{parked\ car} + P_{stationary\ tanks} + P_{hydrogen} + P_{acetylene}$$
$$or,\ P_2 = 0 + 6.18 \times 10^{-9} + 0 + 3.7 \times 10^{-7} + 0 = 3.76 \times 10^{-7}\ per\ year. \quad (15\text{-}9)$$

Regulatory Guide 1.91 provides an acceptable methodology to evaluate the potential hazards by an explosion on safety-related SSCs. Regulatory Guide 1.91 also states that potential explosion hazards can be screened out if the annual exposure frequency is less than 10^{-6} and conservative estimates are used. PG&E made conservative estimates of the potential explosion hazards, so an annual frequency limit of 10^{-6} is applicable here. Therefore, the staff concludes, based on the review of information and analyses presented by PG&E, that no safety-related SSCs at the proposed facility will be subjected to explosion overpressures that exceed the 6.9 kPa [1 psi] threshold.

The staff reviewed the information provided by the applicant regarding potential hazards from an accidental onsite explosion at the proposed facility. The staff found the analysis acceptable because the applicant:

- Appropriately identified the potential sources of hazard;
- Used the Regulatory Guide 1.91 value of 6.9 kPa [1 psi] as the limiting air overpressure for all safety-related structures;
- Developed a probabilistic hazard analysis to estimate the annual frequency of exposure of safety-related structures from each potential source of explosion for those situations that do not meet stand-off zone criteria based on the 6.9-kPa [1-psi] air overpressure limit;
- Summed the annual frequency of explosion hazard from each individual source to estimate the total hazard to the proposed facility, as recommended in Regulatory Guide 1.91; and
- Used conservative assumptions to estimate the annual frequency of exposure from each source of the explosion hazard.

Based on the foregoing evaluation, the staff finds that the Diablo Canyon ISFSI SSCs will be able to maintain subcriticality, confinement, and sufficient shielding of the stored fuel for all postulated onsite explosion events.

15.1.2.5 Electrical Accident

Section 8.2.8 of the SAR evaluates the potential consequences of lightning strikes and a 500-kV transmission line drop on the HI-STORM 100SA storage casks and the HI-TRAC 125 Transfer Cask. Of the different 500-kV transmission line drop scenarios that were considered, the worst-case condition is defined by a line drop of a single conductor of one phase, which causes a single line-to-ground fault current and a voltage-induced arc at the point of contact.

Both electrical events (i.e., lightning strike and 500-kV transmission line drop) manifest themselves as electrical discharges that travel along the least resistive path through the cask to ground. Because these events originate from sources that are outside the confines of the cask, the path of least electrical resistance for the HI-STORM 100SA storage cask is the overpack, and for the HI-TRAC 125 Transfer Cask, the enclosure shell. As a result, the MPC will not be susceptible to any electrically induced damage in either case.

In the case of a lightning strike, it was satisfactorily demonstrated that the temperature increase of the HI-STORM 100SA storage cask overpack and HI-TRAC 125 Transfer Cask enclosure shell will be less than 0.6 °C [1 °F].

For the case of the 500-kV transmission line drop, it was determined that holes would be created in both the HI-STORM 100SA storage cask and HI-TRAC 125 Transfer Cask outer shells by way of material sublimation. Behind the steel outer shell of the HI-STORM 100SA storage cask is a thick concrete layer that would exhibit only localized spalling and crystallization in the immediate region where the steel outer shell sublimation occurred. The staff determined that the resulting effects on the HI-STORM 100SA storage cask decay heat removal and radiation shielding capabilities would be minimal. A hole created in the HI-TRAC 125 Transfer Cask outer shell could cause a loss of the water jacket designed to provide neutron shielding and facilitate removal of the spent nuclear fuel decay heat. As discussed in Section 8.2.11 of the SAR, a loss of the water jacket does not cause the accident radiation dose to offsite individuals to exceed the limits of 10 CFR §72.106, and the increase in fuel cladding and component material temperatures will not exceed their short-term accident temperature limits. Moreover, the MPC internal pressure will remain below the accident design limit gauge pressure of 1.38 MPa [200 psi]. Additionally, these events are even less likely to impact a loaded transfer cask, as cask transfer activities are of relatively short duration, and will generally not be conducted under the adverse conditions most likely to result in a lightning strike or transmission line drop.

15.1.2.6 Earthquake

The staff has reviewed the information presented in the following SAR sections: 8.2.1, "Earthquake"; 2.6, "Geology and Seismology"; and 3.2.3, "Seismic Design." Section 4.5 of the SAR classifies the SSCs important to safety based on the Quality Assurance (QA) Program described in Chapter 11 of the SAR. The importance to safety for each of these SSCs is further refined into three QA classification categories (i.e., Categories A, B, and C) based on the guidance contained in NUREG/CR–6407 (McConnell, et al., 1996). The Category A SSCs important to safety include the: (1) MPC; (2) fuel basket; (3) damaged fuel container; (4) transfer cask; (5) MPC lift cleats and downloader slings; (6) transfer cask impact limiters and lift links; (7) HI-STORM 100 System lifting brackets, mating device bolts, and shielding frame, and lift links; (8) cask transporter; and (9) lateral restraints (HI-TRAC 125 Transfer Cask and transporter at the CTF). The classification of Category B SSCs important to safety include the: (1) HI-STORM 100SA storage cask overpack; (2) storage pads; (3) overpack anchorage hardware; (4) CTF; (5) transfer cask horizontal lift rig and lift slings; (6) upper and lower fuel spacer columns and end plates; (7) transporter connector pins; and (8) helium fill gas. The classification of Category C SSCs important to safety includes the HI-STORM 100 System cask mating devices (except bolts and shielding frame).

A seismic event can occur at any time during any stage of a transfer or storage operation involving a cask or a canister. At a specific site, earthquake potential is often described by the annual probability of exceeding certain ground motion levels or seismic hazard curves. The design earthquake, double-design earthquake, Hosgri earthquake (HE), and long term seismic program (LTSP) earthquakes form the seismic licensing basis for the DCPP. The applicant indicated that, because both DCPP and the ISFSI sites are classified as rock and they have similar ranges of shear-wave velocities within the rock classification, and because the distance to the controlling seismic source is essentially the same, the DCPP ground motions are judged to be applicable to the ISFSI design. Section 2.1.6 of this SER provides additional information about the seismic ground motion hazard and the staff's review of the information.

In conducting analyses of transporter stability, slope stability, and ISFSI storage pad sliding, the applicant developed the ISFSI long-period (ILP) earthquake spectra. The ILP are 84th percentile spectra at damping values of 2, 4, 5, and 7 percent for the horizontal and vertical components that extended out 10s and that include near-fault effects of directivity and fling. The applicant indicates that the ILP spectra envelop the double-design earthquake spectra at 2- and 5-percent damping; the HE spectra at 4-, 5-, and 7-percent damping; and the long term seismic program earthquake spectra at 5-percent damping. The applicant further indicates that the use of ILP earthquake spectra for transporter stability, slope stability, and ISFSI storage pad sliding would provide an extra design margin by considering long-period energy. Five sets of ILP spectra-compatible time histories generated from large-magnitude earthquakes ($M > 6.7$) recorded at short distances (< 15 km [9.3 mi] from the fault) were used as input for the analyses. Based on the statements provided in the SAR, the staff concluded that the use of ILP spectra-compatible time histories to assess transporter stability, slope stability, and ISFSI storage pad sliding potential is acceptable.

Seismic Analysis of Cask Transportation on Transport Route

The transport route from the FHB/AB at the DCPP to the ISFSI storage pad is approximately 1.93-km [1.2-mi] long. Approximately one-third of the route is on bedrock, and the rest is on surficial deposits over bedrock. The route is made up of slopes with an 8.5-percent nominal grade decline and a 6-percent nominal grade incline and a 2-percent grade perpendicular to the roadway with a decline toward the hill side. The minimum roadway width is 7.92 m [26 ft]. The cask transporter carries a HI-TRAC 125 Transfer Cask in a horizontal position from the FHB/AB to the CTF for MPC transfer operation. After the MPC transfer operation is executed, the cask transporter carries the loaded overpack in a vertical orientation to the final position on the ISFSI storage pad. The cask transporter is 5.37-m [17.625-ft] wide and 7.47-m [24.5-ft] long. The applicant states that the maximum acceptable sliding movement along the roadway is limited to the cask transporter track length to ensure that the transporter will remain on the roadway after exiting a turn in the roadway. Assuming that the cask transporter travels along the middle of the roadway, the allowable lateral sliding distance is the distance between the edge of the transporter and the edge of the roadway, which is approximately 1.28 m [4.19 ft].

During transport to the ISFSI storage pad, the cask transporter protects the MPC from the effects of earthquake ground motions. The transporter stability assessment discussed in the SAR was analyzed three dimensionally. The cask transporter, the HI-STORM 100SA storage cask, the HI-TRAC 125 Transfer Cask, the MPC (including the fuel basket, fuel, and lid), and the cask lids were modeled as rigid bodies. The mass of the MPC and the contained spent nuclear fuel is lumped in a free-standing rigid cylinder. Three cases of roadway conditions were

modeled: flat surface, 6-percent grade, and 8.5-percent grade. For all cases, the ground surface was treated as a nondeformable boundary. The SAR states that a transporter stability analysis was performed for a potential transporter overturning or sliding off the roadway using only the bedrock ground acceleration associated with the ILP earthquake time histories. The maximum sliding along the roadway axis of approximately 0.77 m [2.52 ft] occurred on the portion of 8.5-percent grade roadway, and the maximum sliding transverse to the roadway axis of approximately 0.27 m [0.89 ft] occurred on the portion of the roadway with a 6-percent grade. These sliding distances are small compared to the corresponding allowable sliding distance. The analysis also demonstrated that overturning is not credible under the ILP seismic events.

The applicant indicates that peak ground accelerations at certain points along the surface of surficial deposits over bedrock of the transport route can be 1.5 to 2.0 times the amplitude of the peak ground acceleration on bedrock. PG&E did not specifically analyze the potential for overturning and sliding of the transporter on surficial deposits. The SAR points out that a significant safety margin exists to prevent a transporter from overturning or sliding off the roadway while traveling on the surficial deposits even through the ground acceleration would be amplified.

PG&E provided two analyses to address the potential accident scenario, in which an earthquake occurs while the cask is being transported on a portion of the roadway underlain by soil to the CTF or ISFSI pad. The first is a risk assessment to show that this scenario is not credible. The second is a calculation to show that transporter and cask will remain stable during an earthquake. The staff's review of these two analyses is discussed in detail in the following paragraphs. In summary, the staff agrees with the PG&E assessment that this is not a credible scenario. The staff concludes that the regulatory requirements of 10 CFR §72.90, §72.90(a), §72.98, §72.102, and §72.122 have been satisfied.

PG&E conducted a probabilistic risk assessment calculation and concluded that the annual probability of damage to the transporter while transporting spent fuel from the power plant to the CTF is 2.1×10^{-10}, which is substantially less than the 1×10^{-6} threshold criterion recommended for credible events. The PG&E probabilistic risk assessment calculation includes the 1.4×10^{-3} annual exposure probability for transport casks on the transport route (12 hr/yr) and the 1.2×10^{-7} annual exceedence probability for two times the ILP earthquake ground motions.

The staff reviewed the PG&E probabilistic risk calculation and agrees with their conclusion. Specifically, the annual exceedence probability for earthquake-induced damage of the transfer cask while in transit from the power plant to the CTF is less than 1×10^{-6} and is, therefore, not a credible hazard. The use, however, of the annual exceedence probability associated with twice the ILP earthquake ground motions is not considered to be appropriate. Twice the ILP earthquake ground motions was used by PG&E to account for possible site response amplification on those portions of the transport route underlain by soil, not as an added factor in the probability calculation. For this reason, the annual probabilities of the ILP earthquake ground motions, not the annual probabilities for twice the ILP earthquake ground motions, should be used in the probabilistic risk calculation.

The staff independently estimated the upper bound annual exceedence probability for earthquake-induced damage of the casks while in transit from the power plant to the CTF. The estimated probability would be no more than 1.4×10^{-7} per year. The calculation performed by

the staff assumed a maximum annual probability for the ILP earthquake ground motions to be less than 1×10^{-4}. When combined with the 1.4×10^{-3} probability of annual exposure of the transfer casks being in transit, an upper bound value of 1.4×10^{-7} probability of annual exposure is calculated. The exact probability depends on a number of factors, including the spectral frequency of interest and the statistical measure used (mean, median, or 84th percentile). Based on this calculation, the staff concludes, with reasonable assurance, that earthquake-induced damage of the loaded transfer cask while in transit from the power plant to the CTF is not a credible hazard.

Seismic Analysis of the Cask Transfer Facility

The staff reviewed Section 4.2.1.2 of the SAR and found that the structural analysis of the CTF demonstrates that it is designed to mitigate the effects of seismic loading as documented in Section 5.1.4.4 of this SER.

The steel structures of the CTF were analyzed to demonstrate compliance with the material allowables (Holtec International, 2001c). This analysis addressed the following major structural elements: main shell, lifting jacks, jack support platform, CTF base support block, and lifting platform. The appropriate spectral values are used to account for possible amplification of the horizontal accelerations of the stacked components. The applicant demonstrated that the factors of safety for all components and all load conditions are greater than 1.0. The adequacy of the structures has been demonstrated by the analysis results given in the SAR, as designed to satisfy the requirements of ASME Section III, Subsection NF (ASME International, 1995a).

Loads from the Holtec International structural evaluation were also used in the calculation of the necessary thickness and reinforcement for the CTF concrete (ENERCON Services Inc., 2001a). The analysis determined the required size and general reinforcing requirements to resist the loads applied to the concrete structure. The concrete structure is designed to withstand loads from both the CTF and the transporter. Using the controlling load combinations, an analysis identified shear and axial forces and moments in the reinforced concrete structural elements of the CTF. Steel reinforcement size and placement for the pad and wall were established based on these demands. The design of the concrete structure and its reinforcement are based on the requirements in American Concrete Institute (ACI) 349-97 (American Concrete Institute, 1998). Results of the analysis indicate that the available design strength of the CTF exceeds that required for the factored design loads.

Seismic Analysis of the HI-STORM 100SA Overpack Anchored on the ISFSI Storage Pad

Structural analyses of the anchored HI-STORM 100SA overpack are provided in the HI-STORM 100 System FSAR (Holtec International, 2002). The staff's evaluation of the HI-STORM 100 System FSAR is documented in the NRC HI-STORM 100 System SER (U.S. Nuclear Regulatory Commission, 2002b). The Diablo Canyon ISFSI SAR Section 4.2.3 provides a summary of the analyses performed in the HI-STORM 100 System FSAR. The loading conditions at the Diablo Canyon ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR (Holtec International, 2002). As documented in the HI-STORM 100 System SER, the structural analysis shows that the structural integrity of the HI-STORM 100 System cask system is maintained during all credible loads. Based on the results presented in the HI-STORM 100 System FSAR, the stresses in the overpack structures

during the most critical load combinations are less than the allowable stresses of ASME Boiler and Pressure Vessel Code, Section III (ASME International, 1995b) for the structures materials.

Seismic Analysis of the ISFSI Storage Pad

SAR Section 8.2.1.2.3.1, "Cask and Anchorage Seismic Analysis," summarizes seismic analyses of the cask and anchorage system performed by Holtec International. The staff's review of this analysis is summarized in Section 5.1.3.4 of this SER. Although the Diablo Canyon site-specific seismic zero period accelerations for all events are lower than those identified in Appendix B of the Holtec HI-STORM 100 Certificate of Compliance (U.S. Nuclear Regulatory Commission, 2002a), Holtec International performed a specific analysis of the cask anchoring system to be used at Diablo Canyon ISFSI (Holtec International, 2001d). The primary reason for this analysis was the difference in the number of anchor rods identified for the Diablo Canyon ISFSI anchoring system with respect to the design basis given for the HI-STORM 100SA System (Holtec International, 2002). The results indicate that the casks do not develop body decelerations that exceed the cask design basis of 45 g. The seismic events postulated for the Diablo Canyon ISFSI do not induce stresses in the preloaded anchor studs, cask flange, and shell that exceed the design-basis ASME Code limits. The interface loads transferred to the ISFSI pad embedment were established using acceptable methods.

SAR Section 8.2.1.2.3.2, "Storage Pad Seismic Analyses," identifies the analysis performed to ensure that the reinforced concrete pads and the anchored casks remain functional during all seismic conditions. Two analyses are covered in this section, a static analysis (ENERCON Services Inc., 2001b) and a nonlinear pad sliding analysis (Pacific Gas and Electric, 2001a). The static analysis was performed to determine the storage pad size and thickness required to resist the loads resulting from seismic accelerations applied to the pad and resultant loads from the cask dynamic analysis (Holtec International, 2001d). In addition to the cask loads, an inertial force was applied to the pad with reference to the zero period acceleration of the seismic event. The pad and cask vertical displacements are small and within acceptable limits. These maximum tensile stresses in the concrete are less than the tensile stress that will cause cracking in the 34.5-MPa [5,000-psi] concrete. The maximum compressive stress is significantly less that the 34.5-MPa [5,000-psi] design value. Sections throughout the pad were isolated for the HE seismic event calculations and the internal forces acting upon them were computed. The resulting internal forces for design purposes are given in Table 11 of the ENERCON calculation package (ENERCON Services Inc., 2001b). The results of the analysis were used in Calculation No. PGE–009–CALC–007 (ENERCON Services Inc., 2003a) to evaluate the concrete per the design codes and to determine the size of the steel reinforcement needed for compliance with the requirements of ACI 349-97 (American Concrete Institute, 1998). The staff has reviewed this calculation and finds that it demonstrates compliance with the requirements of ACI 349-97.

The anchorage system was designed to meet the ductile anchorage provision of the proposed Draft Appendix B for ACI 349-97. To satisfy the requirements of Appendix B of ACI 349-97, the diagonal tension shear capacity must exceed the anchor bar ductile design strength of 1.05 MN [235.63 kips]. The applicant has provided sufficient reinforcing steel to ensure the failure cone for concrete pullout intersects sufficient rebar to prevent brittle failure (ENERCON Services Inc., 2003a). The reinforcing steel in the storage pad (ENERCON Services Inc., 2003b) has been sized in accordance with the requirements of ACI 349-97.

15.1.2.7 Loss of Shielding

Section 8.2.11 of the SAR (Pacific Gas and Electric Company, 2003) evaluates the potential consequences of a loss-of-neutron shielding for the HI-TRAC 125 Transfer Cask. The potential consequences of this postulated accident were determined by assuming a loss of the water jacket and Holtite-A solid neutron shielding. The staff previously determined that the methodology used to assess this postulated accident is acceptable and the short-term fuel cladding and other component temperature limits, the MPC accident internal pressure, and the accident dose limits defined by 10 CFR §72.106 are not exceeded, so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are adhered to. The staff has confirmed that appropriate limits have been incorporated into the proposed Diablo Canyon ISFSI Technical Specifications.

Section 8.2.6.3 of the SAR specifies that the consequences of postulated explosion events analyzed for the Diablo Canyon ISFSI are enveloped by the design-basis accident conditions in the HI-STORM 100 System FSAR (Holtec International, 2002). Additionally, there is no effect on shielding, criticality, thermal, or confinement capabilities of the HI-STORM 100 System as a result of the explosion pressure load. Based on the structural and radiological evaluations presented in Chapters 3 and 11 of the HI-STORM 100 System FSAR, the applicant concludes that the MPC confinement boundary will remain intact and the shielding effectiveness of the storage and transfer casks will not be significantly affected by any potential onsite explosion.

Considering the results of the onsite explosion accident analysis evaluation presented in Section 15.1.2.4, "Fire and Explosion," of this SER, the staff finds that the maximum reduction in ISFSI radiation shielding thickness, material shielding effectiveness, or loss of temporary shielding in all possible shielding areas caused by postulated onsite explosion events, has been adequately evaluated by the applicant. Therefore, the information and analyses presented by the applicant provide reasonable assurance that the accident dose to any individual beyond the owner-controlled area will not exceed the limits specified in 10 CFR §72.106(b), and the occupational exposures from accident recovery operations will not exceed the limits specified in 10 CFR Part 20.

15.1.2.8 Adiabatic Heatup

The staff has previously determined that the methodology used to estimate the time required to reach the short-term, fuel-cladding temperature limit of spent nuclear fuel stored in the HI-STORM 100 System storage cask under adiabatic conditions is acceptable (U.S. Nuclear Regulatory Commission, 2002a,b). The HI-STORM 100 System FSAR (Holtec International, 2002, Figure 11.2.6) indicates that a total cask decay heat load of 30 kW [102,360 BTU/hr], which bounds the cask decay heat load specified for the Diablo Canyon ISFSI, will not cause the short-term cladding temperature limit for the spent nuclear fuel to be exceeded for 45 hours under adiabatic conditions. Moreover, the internal pressure limit for the MPC is not exceeded within the 45-hour timeframe for this condition.

In the event that the HI-STORM 100 System storage cask is subjected to conditions that thermally insulate its exterior (e.g., encased within soil as the result of a landslide), the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

15.1.2.9 Full Blockage of Air Inlets and Outlets

The staff previously determined that the methodology used to estimate the time required to reach the short-term, fuel-cladding temperature limit of spent nuclear fuel stored in the HI-STORM 100SA storage cask subjected to 100-percent blockage of the air inlet ducts is acceptable (U.S. Nuclear Regulatory Commission, 2002a,b). For the bounding values of decay heat load of 30 kW [102,360 BTU/hr] and insolation of 834 w/m^2 [800 g-cal/cm^2] per day {387 W/m^2 [123 BTU/hr-ft^2]}, the short-term cladding temperature limit for the spent nuclear fuel will not be exceeded for 72 hr when the HI-STORM 100SA storage cask air inlet ducts are 100-percent blocked. Moreover, the internal pressure limit for the MPC is not exceeded within the 72-hour timeframe for this condition. Furthermore, the HI-STORM 100 System CoC (U.S. Nuclear Regulatory Commission, 2002a, Appendix A) includes surveillance requirements for ensuring that the cask heat removal system is operational during storage (i.e., the air ducts are inspected every 24 hours to ensure that the ducts are free of blockages). In the event that the HI-STORM 100SA storage cask air inlet ducts are found to be partially obstructed or blocked, the previously reviewed and accepted recovery operation procedures will be implemented (U.S. Nuclear Regulatory Commission, 2002a,b).

15.1.2.10 Tornadoes and Missiles Generated by Natural Phenomena

The staff reviewed the information presented in the Diablo Canyon ISFSI SAR Sections 3.2.8, "Tornado and Wind Loadings;" 3.3.2.3.3, "Maximum Permissible Tornado Wind and Missile Load;" 4.2.3.3.2.6, "Tornado Winds and Missiles;" and 8.2.2, "Tornado." The staff also reviewed responses to requests for additional information (Pacific Gas and Electric Company, 2002; RAIs 4-3, 15-18, 15-19, 15-20, and 15-21), including the report, "Design Basis Wind and Tornado Evaluation for DCPP", (Holtec International, 2001e, Attachment 4-1) and Section 3.3, "Wind and Tornado Loadings," of the FSAR for DCPP, Units 1 and 2 (Pacific Gas and Electric Company, 2001b). This evaluation assumed that site personnel would not have any prior warning before the ISFSI SSCs are impacted by a potential design-basis tornado or a tornado missile.

The annual mean number of days with tornadoes is zero for the ISFSI site. Characteristics of the design-basis tornado and tornado missile are given in Section 3.2.1 of the ISFSI SAR. The SAR developed the characteristics of the design-basis tornado in accordance with the DCPP licensing-basis wind speed of 89 m/s [200 mph]. The proposed site is located in Region II as defined in Regulatory Guide 1.76 (U.S. Nuclear Regulatory Commission, 1974). The characteristics of the design-basis tornado for the proposed ISFSI are defined as a tornado with a maximum wind speed of 89 m/s [200 mph], a rotational speed of 70 m/s [157 mph], a translational speed of 19 m/s [43 mph], and a 5.9-kPa [0.86-psi] pressure drop at a rate of 2.5 KPa/s [0.36 psi/s].

The design-basis tornado missiles considered in the Diablo Canyon ISFSI SAR are based on Spectrum II missiles of Section 3.5.1.4, "Missiles Generated by Natural Phenomena," of NUREG–0800 (U.S. Nuclear Regulatory Commission, 1981a), the Diablo Canyon FSAR Update (Revision 14, Pacific Gas and Electric Company, 2001b), and the three 500-kV tower missiles specific to the ISFSI (Pacific Gas and Electric Company, 2003). These objects are postulated to be picked up and transported by the winds of a design-basis tornado. A list of these missiles is provided in Table 15.1.

Table 15-1. Tornado missiles considered in Diablo Canyon Independent Spent Fuel Storage Installation (ISFSI)

Missile	Mass kg [lb]	Velocity Considered m/s [mph]	
		Diablo Canyon ISFSI Safety Analysis Report	Holtec International (Region I)
Automobile	1,800 to 1,814 [3,968 to 4,000]	56 [126]	56 [126]
Utility Pole	510 [1,124] 33 cm- [13.5 in-] diameter, 10.7 m [35 ft] long, density of 688.8 kg/m^3 [43 lb/ft^3] in Diablo Canyon Power Plant Units 1 & 2 (DCPP)	16 [35]	48 [107.4]
30 cm- [12 in-] diameter Schedule 40 pipe	340 [744] 4.5 m [15 ft] long, density of 7,849 kg/m^3 [490 lb/ft^3] in DCPP	2.2 [5]	28 [62.6]
15 cm- [6 in-] diameter Schedule 40 pipe	130 [285] 4.5 m [15 ft] long, density of 7,849 kg/m^3 [490 lb/ft^3] in DCPP	3 [7]	42 [93.9]
20-cm- [8-in-] diameter solid steel cylinder	125 [276]	56 [126]	56 [126]
10 cm x 30 cm x 3.05 m [4 in x 12 in x 10 ft] board	49 [108] In DCPP Units 1 & 2, 91 kg [200 lb], density of 801 kg/m^3 [50 lb/ft^3]	89 [200]	Not Applicable
7.5 cm- [3 in] diameter, 3.05 m [10 ft-] long Schedule 40 pipe	34.5 [76] In DCPP, 4.5 m [15 ft-] long pipe with density of 7,849 kg/m^3 [490 lb/ft^3]	29.8 [66.7]	Not Applicable
500-kV insulator string	344.7 [760]	70 [157]	Not Applicable
5 cm x 5 cm x 0.32 cm [2 in x 2 in x 1/8 in] steel angle (1.5 m [5 ft] long)	3.9 [8.6]	70 [157]	Not Applicable
2.5 cm- [1 in-] diameter steel rod	4 [8] 0.9 m [3 ft-] long, density of 7,849 kg/m^3 [490 lb/ft^3] in DCPP	2.2 [5]	40 [89.5]
2.5-cm- [1-in-] diameter solid steel sphere	0.22 [0.5]	56 [126]	56 [126]

Important to safety SSCs that may be affected by design-basis tornado missiles are: (1) the CTF, (2) site transporters, (3) the transfer cask, and (4) the storage casks. These SSCs are required to function during this design-basis event.

Based on the resulting kinetic energy, PG&E's analysis assumed that an automobile at 203 km/hr [126 mph], a 500-kV insulator string at 253 km/hr [157 mph], and a 2.5 cm-[1 in-] diameter steel rod at 144 km/hr [89.5 mph] are the bounding missiles for the large, intermediate, and small missiles categories. PG&E assumed that the impact velocity of an automobile is consistent with that suggested in NUREG–0800. PG&E developed an equation to estimate the maximum horizontal missile velocity for a 322-km/h [200-mph] tornado from a 386-km/h [240-mph] Type III tornado curve using Figure 16.3.1 of Simiu and Scanlan (1986). However, the basis for the equation is not clear. This formula will produce a different result for the correlation power factor if tornados other than Type III are used.

The staff's confirmatory calculation indicated that energy imparted by the automobile is significantly larger than that of a utility pole. Therefore, any impact of a utility pole would be bounded by the automobile impact for assessing transporter stability. Holtec International (2001e) studied the effects of transporter stability while transporting a loaded transfer cask to the storage area at the proposed facility. This analysis included a large missile represented by a 1,800-kg [4,000-lb] car traveling at a speed of 56 m/s [126 mph]. The impact analysis result indicates that a loaded transporter would be displaced laterally by a distance of only 1.65 cm [0.65 in]. The transporter remains stable and does not tipover as a result of this impact.

The staff reviewed the information provided by the applicant, evaluated the analyses of potential hazards from design-basis tornadoes and tornado missiles at the proposed facility, and conducted a confirmatory analysis. The staff concludes that a tornado or tornado-generated missile would not impair the ability of the SSCs to maintain subcriticality, confinement, and sufficient shielding of the spent fuel during transfer or storage.

15.1.2.11 Accidents at Nearby Sites—Aircraft Crash Hazards

The staff reviewed the information presented in the Diablo Canyon ISFSI SAR, Section 2.2 (Pacific Gas and Electric Company, 2003). In addition, the staff reviewed information presented by PG&E in response to staff questions regarding aircraft crash hazards (Pacific Gas and Electric Company, 2003d). The purpose of this review is to ensure that the risk to the proposed facility caused by aircraft hazards has been appropriately estimated and is acceptable.

The staff reviewed the aircraft crash hazard analysis in accordance with NUREG–0800, Section 3.5.1.6, "Aircraft Hazards." The staff accepts the methodology in NUREG–0800, as applicable, for reviewing the aircraft crash probability for the proposed ISFSI. Section 3.5.1.6 of NUREG–0800, provides three screening criteria that must be satisfied to conclude, by inspection, that the aircraft hazards at a nuclear power plant are less than 1×10^{-7} per year for accidents that could result in radiological consequences greater than 10 CFR Part 100 exposure guidelines. The staff's review indicates the proposed facility site does not satisfy screening Criterion II.1(a), which states, "The plant-to-airport distance, *D* is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance, *D*, is greater than 10 statute miles, and the projected annual number of

operations is less than 1000 D^2." Based on the information given in the SAR, and the air traffic increase projected in the next 25 years by the Federal Aviation Administration (FAA), the projected annual number of operations may not satisfy Criterion II.1(a). Additionally, screening Criterion II.1(c) states, "The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern." As stated by the applicant in Section 2.2.1.1 of the SAR, air traffic to San Luis Obispo County Regional Airport passes the proposed site at a distance of 1.6 km [1 mi]. Therefore, screening Criterion II.1.(c) is also not satisfied, and in accordance with NUREG–0800 review guidance, a detailed review is needed to assess the aircraft crash hazards for the proposed site. PG&E provided its detailed analysis to estimate the annual frequency of a potential aircraft crash at the proposed ISFSI (Pacific Gas and Electric Company, 2003d). Additionally, the staff conducted its own confirmatory analysis. These analyses are discussed in the following sections.

Estimating the total probability of an aircraft crash onto the proposed ISFSI requires an evaluation of crash probabilities from several sources:

- Aircraft taking off and landing at San Luis Obispo County Regional Airport;
- Aircraft taking off and landing at other municipal airports located close to the site, such as Oceano County Airport, Camp San Luis Obispo Heliport, and Vandenberg Air Force Base;
- Aircraft flying the low-altitude flight corridor V–27 (commercial airway) and either landing at or departing from San Luis Obispo County Regional Airport, or not landing at or departing from San Luis Obispo County Regional Airport; and
- Aircraft flying military training route VR–249.

Aircraft Taking Off and Landing at San Luis Obispo County Regional Airport

San Luis Obispo County Regional Airport is approximately 19.3 km [12 mi] east of the proposed site. This airport has four runways. Only Runway 11 is equipped for instrument landing approach. The other three runways are used for visual landing. Some aircraft use Airway V–27 to align for instrument landing at the airport. Some of these aircraft come within approximately 1.6 km [1 mi] of the proposed ISFSI site at an elevation of 914 m [3,000 ft]. Based on NUREG–0800, any aircraft flying Airway V–27 for instrument landing at San Luis Obispo County Regional Airport will be in an in-flight mode. Their contribution to the overall crash hazard has been accounted for in the analysis of V-27. The commonly used approach route for visual landing at San Luis Obispo County Regional Airport passes approximately 12.8 km [8 mi] from the proposed site.

In Section 2.2.1.3 of the ISFSI SAR, "Hazards from Air Crashes," PG&E states that approximately 92,330 operations (take offs or landings) occur annually at San Luis Obispo Regional County Airport. However, while discussing local traffic on Airway V–27, PG&E stated approximately 16,000 takeoffs and landings occur annually at San Luis Obispo County Regional Airport, based on an average of data from 1998–2001, by commercial or air-taxi aircraft. Primarily turboprop aircraft with a gross weight of not more than 13,608 kg [30,000 lb] are used in these commercial flights. Additionally, private aircraft (i.e., general aviation aircraft) landed at or took off from San Luis Obispo County Regional Airport approximately 7,560 times monthly,

based on the average of data from 1998–2001. These aircraft have gross weight of less than 5,670 kg [12,500 lb]. Consequently, at least a total of approximately 106,720 landings and departures took place annually at the San Luis Obispo County Regional Airport without counting the operations by military aircraft.

PG&E concluded that no analysis would be necessary as the number of annual operations at San Luis Obispo County Regional Airport is below the number needed to have an annual crash frequency of 10^{-7} based on Criterion II.1(a) of NUREG–0800.

The staff independently verified the number of annual operations at this airport from the FAA database at <http://www.gcr1.com/5010WEB/default.htm> and another source at <http://www.airnav.com>. Based on this information, approximately 72,000 annual operations take place at the San Luis Obispo County Regional Airport, out of which approximately 16,500 operations are by commuter aircraft and approximately 55,000 operations are by general aviation aircraft, in addition to approximately 900 operations by military aircraft. Because the number of annual operations given by PG&E (i.e., 106,720 with 900 additional operations by military aircraft) is bounding, the staff used that value for further review.

The staff reviewed the information and the analysis provided by the applicant with respect to the potential hazards of aircraft taking off and landing at San Luis Obispo County Regional Airport. The staff found the hazards acceptable because adequate information has been presented to describe the potential hazards and an acceptable methodology has been used to screen the potential hazards. As the airport is approximately 19.3 km [12 mi] away from the proposed ISFSI site, the estimated annual frequency of crash onto the proposed ISFSI is insignificant using the methodology given in NUREG–0800 to analyze the crash potential of aircraft landing at or taking off from an airport. Based on the this information, the staff has concluded that aircraft taking off and landing at San Luis Obispo County Regional Airport would not pose any undue hazard to the proposed ISFSI.

Aircraft Taking Off and Landing at Oceano County Airport, Camp San Luis Obispo Heliport, and Vandenberg Air Force Base

There are several smaller municipal airports in the vicinity of the proposed site. Oceano County Airport is located 24 km [15 mi] away from the proposed site. Only general aviation aircraft with weight not more than 5,670 kg [12,500 lb] use this airport. In the Diablo Canyon ISFSI SAR, PG&E estimated annual traffic at this airport to be no more than 26,400. Both the FAA database at <http://www.gcr1.com/5010WEB/default.htm> and another source at <http://www.airnav.com> give the estimated annual number of flight operations at this airport to be approximately 10,000. Therefore, the estimated number for annual flights used in the SAR is conservative. Again, PG&E concluded that no analysis would be necessary, based on Criterion II.1(a) of NUREG–0800, as the number of annual operations at Oceano County Airport equates to an annual crash frequency of 10^{-7}, or lower. Based on the analysis methodology given in NUREG–0800, the staff estimates that the frequency of aircraft crashing onto the proposed ISFSI while taking off or landing at Oceano County Airport is insignificant.

Camp San Luis Obispo airfield, located approximately 13 km [8 mi] northeast of the proposed ISFSI site, is a heliport owned by the U.S. Army. The staff concludes that landings and takeoffs by helicopters at this heliport do not pose a credible hazard to the proposed ISFSI because of the type of aircraft and the long distance from the ISFSI, based on the U.S. Department of

Energy (DOE) Standard, "Accident Analysis for Aircraft Crash Into Hazardous Facilities." (DOE-STD-3014-96, U.S. Department of Energy, 1996).

Vandenberg Air Force Base is 56 km [35 mi] away from the proposed site. At this distance, the number of takeoffs or landings per day will not exceed the NUREG-0800 criterion that would require further analysis. Therefore, any landing or takeoff operations at Vandenberg Air Force Base will pose a negligible hazard to the proposed Diablo Canyon ISFSI.

Aircraft Flying Low-Altitude Airway Victor 27 (V-27)

A low-altitude Airway Victor 27 (V-27) passes approximately 8 km [5 mi] east of the proposed facility. Aircraft use this airway to fly between the Santa Barbara area and the Big Sur area. Aircraft using V-27 can either land at San Luis Obispo County Regional Airport or fly through to their destination without landing. The majority of the aircraft using airway V-27 fly at an en route altitude of 3,333 m [10,000 ft] above mean sea level (MSL). Occasionally, V-27 is also used by traffic approaching San Luis Obispo County Regional Airport from the south for instrument landings on Runway 11, or for instrument departures to the south from Runway 11. Some landings on Runway 29, and instrument departures to the south from Runway 29 also use V-27. Some aircraft using this approach or departure pattern pass as close as 1.6 km [1 mi] from the proposed ISFSI site at an elevation of 914 m [3,000 ft].

Aircraft Landing or Departing San Luis Obispo County Regional Airport Using V-27

PG&E used the FAA database <http://www.apo.data.faa.gov/faaatadsall.htm> to obtain information about commercial or air taxi (AT) and general aviation operations at San Luis Obispo County Regional Airport. An average of 16,000 AT operations (i.e., takeoffs or landings) took place annually during 1998–2001. Additionally, an average of 1,781 AT landings took place annually at this airport under instrument meteorological conditions.

Based on the scheduled airline flight information at San Luis Obispo County Regional Airport, PG&E estimated approximately 65 percent of the commercial traffic is departing to or approaching from the south. Therefore, approximately 1,781 × 0.65, or, 1,157 instrument landings may use the V-27 airway annually. Assuming a similar number of takeoffs using instrument conditions, approximately 2,314 flights will use Airway V-27 annually, and this number was considered in the aircraft hazards analysis for the proposed ISFSI.

The FAA database shows that approximately 7,560 landings and takeoffs by general aviation aircraft took place monthly at San Luis Obispo County Regional Airport over the 4-year period of 1998–2001. During the same period, an average of 1,430 flights, which includes local and itinerant general aviation and military flights, landed at San Luis Obispo County Regional Airport annually under instrument conditions. PG&E again assumed that approximately 65 percent of the general aviation traffic is departing to or approaching from the south. Therefore, PG&E considered that approximately 2 × (1,430 × 0.65), or 1,860 operations (takeoffs and landings) took place annually under instrument conditions.

The nearest major airway intersections (CREPE and CADAB) are approximately 18 km [11 mi] and 34 km [21 mi] from the proposed ISFSI site. Holding patterns at both of these intersections would place the aircraft even further away from the proposed ISFSI site, thus they are not considered to contribute to the overall aircraft hazard.

Since the Morro Bay Very-High Frequency Omnidirectional Range Navigation System is used for missed approaches to San Luis Obispo County Regional Airport, PG&E estimated that 5 percent of all instrument landing approaches are missed, and each aircraft remains in the holding pattern for 10 passes. Therefore, for purposes of this analysis, commercial aircraft traffic is assumed to increase by an additional 579 (2,314/2 × 0.05 × 10) annual flights and general aviation aircraft traffic by 465 [1,860/2 × 0.05 × 10] additional annual flights. In its response to additional staff questions, PG&E stated that the assumption that 5 percent of all instrument landing approaches are missed is conservative based on discussions with the personnel at the control tower of San Luis Obispo County Regional Airport regarding the specific approaches available to the airport. Additionally, discussions with pilots of commercial and private aircraft support this conclusion. San Luis Obispo County Regional Airport has limited landing facilities. Most instrument approaches are near minimum weather requirements for using the visual flying rule and result in a visual landing under an instrument flying rule approach. Essentially, zero landing misses take place under this type of approach to the airport. Runway 11 is the only runway available with a precision instrument landing system. If wind and fog results in downwind landing on Runway 11, commercial aircraft will not depart San Luis Obispo County Regional Airport.

PG&E states that aircraft approaching from the south and not during weather classified as instrument meteorological conditions will fly to the CADAB intersection and will land on Runway 29 under visual control. These aircraft do not generally use Airway V–27 while landing at San Luis Obispo County Regional Airport. However, when San Luis Obispo County Regional Airport is under instrument meteorological conditions, all aircraft arriving from the south will use Runway 11 approach, if the ceiling is below 270 to 330 m [900 to 1,100 ft], depending on the aircraft type. This approach uses V-27. However, if the ceiling is above 270 to 330 m [900 to 1,100 ft], the pilot may also use the Runway 29 approach, which does not use V–27. Consequently, a major portion of the aircraft approaching San Luis Obispo County Regional Airport from the south (PG&E has estimated it to be approximately 65 percent) do not use V–27 to land; however, PG&E has conservatively assumed that all aircraft approaching from the south use Airway V–27. The V–27 airway has a width of 12.8 km [8 statute mi] with a center approximately 8 km [5 mi] from the proposed ISFSI site. Consequently, the proposed ISFSI site is 1.6 km [1 statute mi] from the edge of V–27 airway, and an effective width equal to 16 km [10 statute mi] is used in accordance with NUREG-0800 to estimate the probability of air crashes from traffic using this airway at the ISFSI site.

PG&E assumed that the wingspan of commercial aircraft is 29.9 m [98 ft] with a skid distance of 213 m [700 ft] and cotangent of the impact angle, cot φ, equal to 10.2. Using length, width, and height of the facility as 152, 32, and 6.1 m [500, 105, and 20 ft], PG&E estimated the effective area of the facility to be 0.0580 km^2 [0.0224 mi^2] for commercial aircraft, using the formula given in DOE Standard DOE–STD–3014–96. Using a wingspan of 22.3 m [73 ft], skid distance of 213 m [700 ft], and cot φ of 10.2, PG&E estimated the effective area of the facility to be 0.0554 km^2 [0.0214 mi^2] for general aviation aircraft.

Use of 213 m [700 ft] as the skid distance by PG&E is based on the layout of the proposed facility. The proposed facility is surrounded by hills on three sides, which limits the potential skid distance by a crashing aircraft to reach to the SSCs important to safety. The fourth side is protected by a drop in the terrain with a slope greater than 1:1 (PG&E SAR, 2003, Figure 2.2-1).

PG&E assumed a crash rate of 2.5 $\times 10^{-10}$ per km [4 $\times 10^{-10}$ per mi] for commercial aircraft and 0.97 $\times 10^{-7}$ per km [1.55 $\times 10^{-7}$ per mi] for general aviation aircraft flying in this corridor. The crash rate for commercial aircraft is based on the suggested value in Section 3.5.1.6 of NUREG–0800. Additionally, PG&E used Kimura, et. al (1996) to select the crash rate for general aviation aircraft.

Based on the above parameters and using the formula given in NUREG–0800, Section 3.5.1.6, PG&E estimated the annual crash frequency onto the proposed ISFSI by commercial aircraft to be 2.59 $\times 10^{-9}$. Similarly, the annual crash frequency of general aviation aircraft is estimated by PG&E to be 7.7 $\times 10^{-7}$. Therefore, the total crash frequency by aircraft flying airway V-27 is 7.73 $\times 10^{-7}$ per year.

The staff consulted the FAA database <http://www.apo.data.faa.gov/faaatadsall.htm> to independently verify the number of annual flights by both commercial and general aviation aircraft approaching San Luis Obispo County Regional Airport during instrument meteorological conditions. The staff confirmed that the 4-year (1998–2001) average of flights during instrument meteorological conditions for both types of aircraft are acceptable. Additionally, inclusion of 2002 data would somewhat decrease the annual average for both commercial and general aviation aircraft. Therefore, the applicant's consideration of information from 1998–2001 is appropriate.

Commercial Aviation

San Luis Obispo County Regional Airport is served primarily by turboprop or smaller aircraft for the commercial or air taxi traffic. The maximum capacity of these aircraft is 41 people with a maximum gross weight of 13,608 kg [30,000 lb]. Although PG&E used the crash rate of commercial aircraft equal to 4 $\times 10^{-10}$ per mile, as suggested in Section 3.5.1.6 of NUREG–0800, the staff used a crash rate of 9.28 $\times 10^{-10}$ per km [5.801 $\times 10^{-10}$ per mi], as given in Table 2.13 of Kimura, et al. (1996) for off-airport crashes with destroyed aircraft or aircraft that sustained substantial damage to the airframe as a result of the crash. Staff considers this crash rate to be more appropriate for the type of aircraft under consideration. A search of the website <http://www.sloairport.com/flightinfo.html> shows that certified air carriers operate at this airport. A certified air carrier is an air carrier possessing a Certificate of Public Convenience and Necessity issued by the U.S. Department of Transportation in accordance with 14 CFR Part 121 to operate scheduled air services (Kimura et al., 1996). The information obtained by the staff independently from the websites <http://airnav.com> and <http://www.gcr1.com/5010WEB/default.htm> indicates that the traffic at San Luis Obispo County Regional Airport does not have any air-taxi operations, rather it has commercial operations. Therefore, the staff used the crash rate of 9.28 $\times 10^{-10}$ per km [5.801 $\times 10^{-10}$ per mi] as the crash rate appropriate for commercial aircraft operating at San Luis Obispo County Regional Airport.

Although PG&E stated that San Luis Obispo County Regional Airport is primarily serviced by turboprop or smaller aircraft for commercial traffic, PG&E used a wingspan of 29.4 m [98 ft], as suggested for air carriers in commercial aviation in Table B–16 of DOE Standard DOE–STD–3014–96. This table suggests that the wingspan for turboprop aircraft, classified as general aviation aircraft, is 22.3 m [73 ft]. Therefore, use of a higher value for wingspan will produce a larger estimate of the effective area, and therefore, it is conservative. The staff used a wingspan of 29.9 m [98 ft] for commercial aviation. Additionally, the staff considered all general aviation aircraft to be the turboprop type, which has the largest wingspan of all general aviation aircraft types.

Table B–17 of the DOE standard provides the suggested values for the mean of the cotangent of the impact angle (cot φ). For commercial aviation aircraft, the suggested value is 10.2. DOE (1996) recommends cot φ equal to 8.2 for general aviation aircraft. Mean skid distances for commercial and general aviation aircraft are 439 and 18 m [1,440 and 60 ft], as per Table B–18 of the DOE Standard. PG&E asserted that a commercial aircraft would not have enough space to skid for a distance of 439 m [1,440 ft] because of the topography surrounding the proposed ISFSI. The staff agrees with this conclusion and has used a skid distance of 213 m [700 ft] as an appropriate skid distance for commercial aircraft in the calculation. Nevertheless, the staff also used 439 m [1,440 ft] in the calculation to test the sensitivity of the skid distance parameter.

Using a wingspan of 29.9 m [98 ft], cot φ of 10.2, and a skid distance of 213 m [700 ft], the staff estimates the effective area of the proposed facility to be 0.058 km^2 [0.0224 mi^2]. Using a skid distance of 439 m [1,440 ft], however, the estimated area increases to 0.0997 km^2 [0.0385 mi^2] for commercial aircraft. Using a wingspan of 22.3 m [73 ft], cot φ of 8.2, and skid distance of 18 m [60 ft], the effective area is 0.01844 km^2 [0.00712 mi^2]. As discussed before, the effective width of the airway is 16 km [10 mi]. Based on this information, the staff estimates that the annual frequency of a crash of a commercial aircraft onto the proposed facility, using the formula given in NUREG–0800 Section 3.5.1.6, is approximately 3.8×10^{-9}, for a skid distance of 213 m [700 ft]. Assuming a skid distance of 439 m [1,440 ft], (which is not considered realistic given the topography surrounding the proposed ISFSI), the probability of a commercial aircraft crash increases to approximately 6.5×10^{-9} per year.

General Aviation

PG&E stated that the general aviation aircraft using the airport and airways near the proposed ISFSI include small single- and dual-engine aircraft, and small corporate aircraft powered by either propeller or jet. These aircraft with an average gross weight of less than 5,670 kg [12,500 lb] have a maximum capacity of eight people. Kimura, et al. (1996) provide crash rates per flight mile for single- and multi-engine reciprocating, turboprop and turbojet, rotary wing with either reciprocating or turbine engine aircraft. Because the proportion of these aircraft is not known, the staff considers the use of a crash rate of 2.48×10^{-7} per km [1.550×10^{-7} per mi] for all powered aircraft appropriate. As a part of the sensitivity analysis, the staff also used the crash rates equal to 2.416×10^{-7} per km [1.510×10^{-7} per mi] for all fixed-wing (single- and multi-engine reciprocating, turboprop and turbojet) aircraft and 5.669×10^{-7} per km [3.543×10^{-7} per mile] for all rotary-wing (reciprocating or turbine engine) aircraft. Additionally, a wingspan of 22.3 m [73 ft] has been used. As discussed before, this is a conservative estimate of the actual wingspan as the typical wingspan of a general aviation aircraft is given as 15.2 m [50 ft], except for a turboprop aircraft, which has a wingspan of 22.3 m [73 ft] (U.S. Department of Energy, 1996). Additionally, a skid distance of 18 m [60 ft] and cot φ of 8.2 have been used to estimate the effective area of the proposed facility. The effective area of the proposed facility has been estimated to be 0.01844 km^2 [0.00712 mi^2]. Therefore, the staff estimates that the annual frequency of crash of a general aviation aircraft onto the proposed ISFSI, using the formula given in NUREG–0800, Section 3.5.1.6, is approximately 2.6×10^{-7} assuming the crash rate for total powered general aviation aircraft. Assuming the crash rates for all fixed-wing and all rotary-wing aircraft, the estimated annual frequencies of crash of a general aviation aircraft onto the proposed ISFSI are approximately 2.5×10^{-7} and 5.9×10^{-7} per year, respectively.

The staff reviewed the information and analysis presented by the applicant with respect to potential hazards of aircraft flying airway V-27 to land at or depart from San Luis Obispo County Regional Airport. The staff found them acceptable because:

- Adequate information has been presented to describe the potential hazards;
- appropriate bases has been provided for the assumed crash rates for both commercial and general aviation aircraft;
- appropriate bases have been provided for the assumed number of flights of each type of aircraft in the vicinity of the proposed ISFSI using this flying corridor; and
- conservative values of crash parameters have been used to estimate the annual crash frequencies for different types of aircraft.

Aircraft Not Landing or Departing San Luis Obispo County Regional Airport Using V-27

As discussed before, V–27 is a federal airway also used by aircraft flying between the Santa Barbara and Big Sur areas. These aircraft do not land at San Luis Obispo County Regional Airport. The majority of the aircraft in V–27 fly at an altitude of 3,333 m [10,000 ft] above MSL; however, some smaller aircraft may fly at elevations as low as 1,050 m [3,500 ft]. Based on information from the FAA, PG&E estimates that mostly commercial aircraft fly in this airway at a rate of approximately 20 per day or, 7,300 flights per year. Using a crash rate of 6.4×10^{-10} per flight km [4.0×10^{-10} per flight mi] and an effective area of 0.058 km^2 [0.0224 mi^2], PG&E estimated the annual frequency of aircraft flying in this part of V–27 crashing onto the proposed facility would be 6.53×10^{-9}.

The staff estimated the annual crash frequency of aircraft in this category, assuming a skid distance of 213 m [700 ft], and a crash rate of 9.282×10^{-10} per flight km [5.801×10^{-10} per flight mi], as the contribution of this activity to the overall crash frequency is relatively minor. Using the methodology given in NUREG–0800, the staff estimated the crash frequency at the ISFSI for this category of aircraft to be approximately 9.5×10^{-9} per year. The staff reviewed the information and analysis presented by the applicant with respect to potential hazards of aircraft using Airway V–27 to transit between the Santa Barbara and Big Sur areas. The staff found the applicant's analysis acceptable because:

- Information presented to describe the potential hazards is adequate;
- an appropriate basis has been provided for the assumed crash rate, and
- an appropriate basis has been provided for the assumed number of flights of each type of aircraft in the vicinity of the proposed ISFSI.

Aircraft Flying in Military Training Route VR–249

VR–249 is a military training route. Aircraft can fly at any elevation up to 3,333 m [10,000 ft]. Flight through this route requires at least 8 km [5 mi] of visibility with a ceiling at 900 m [3,000 ft]. Aircraft using this route usually remain offshore. The ISFSI SAR indicates that these aircraft do not fly directly over the proposed ISFSI or DCPP.

A majority of the aircraft that flew through VR–249 in the period of September 2001 to September 2002 were F-18 military jets. Additionally, a limited number of F-16s, C-130s, and EA6B aircraft and some helicopters used this route. Based on the information obtained by PG&E from the Naval Air Station at Lemoore, bombs are not carried onboard a majority of the aircraft that fly VR–249, although air-to-air missiles and cannon/machine guns may be carried. The amount of explosive charges in these armaments is considered to be too small to pose a hazard to the proposed ISFSI.

The route VR–249 is used by military aircraft infrequently; approximately 50 flights annually. Additionally, aircraft fly near the proposed ISFSI area in normal flight mode, not in high-stress maneuvers. To be conservative, PG&E assumed that approximately 75 flights use this route in a year. PG&E assumed a wingspan of 33.5 m [110 ft] for an F-18 aircraft. Additionally, a skid distance of 213 m [700 ft] and cot φ of 10.2 have been assumed by PG&E. The calculated effective area of the proposed facility is 0.059 km^2 [0.0228 mi^2].

PG&E was not able to obtain specific crash information for F-18 aircraft to develop a crash rate in normal inflight mode. As a result, PG&E assumed a crash rate of 4.378×10^{-8} per km [2.736×10^{-8} per mi] for all types of aircraft flying in this corridor, based on the crash rates for F-16s developed for a separate ISFSI license application for the Private Fuel Storage Facility (Private Fuel Storage Limited Liability Company, 2002). PG&E assumed that the crash rate for F-16 aircraft could be applied to F-18 aircraft. The centerline of the route VR–249 is approximately 3.2 km [2 mi] offshore. PG&E assumed the width of the route for estimating the crash hazard as 1.6 km [1 mi]. The resulting crash hazard has been estimated to be 4.68×10^{-8} per year.

The majority of the aircraft flying through the route VR–249 are F-18s. The staff used the information given in Tables B–16 through B–18 of the DOE Standard (U.S. Department of Energy, 1996) to estimate the effective area of the proposed ISFSI, assuming all aircraft are either F–16s, F–18s, C–130, or EA–6B. Both F–16s and F–18s are high-performance small aircraft with wingspans of 10.0 m [32 ft 10 in] and 13.62 m [45 ft], respectively. The suggested value for cot φ is 10.4 and the skid distance is 136 m [447 ft] for both of these aircraft (U.S. Department of Energy, 1996). It should be noted that C–130s are transport aircraft and should be categorized as large military aircraft. The EA–6B is a twin-engine aircraft used for electronic countermeasures and is based on the airframe of A-6 aircraft. It has been categorized as a small military aircraft for estimating the effective area of the proposed ISFSI. C–130s have a wingspan of 40.4 m [132 ft 7 in]. EA–6Bs have a wingspan of 16.2 m [53 ft]. A skid distance of 112 m [368 ft] and cot φ of 9.7 have been used for C–130 aircraft. Similarly, a skid distance of 136 m [447 ft] and a cot φ of 10.4 have been used for EA–6B aircraft (U.S. Department of Energy, 1996). The estimated effective area of the proposed ISFSI is 0.0386 km^2 [0.0149 mi^2] for F–16s, 0.0396 km^2 [0.0153 mi^2] for F–18s, 0.0401 km^2 [0.0155 mi^2] for EA–6Bs, and 0.0409 km^2 [0.0158 mi^2] for C–130s. Therefore, a value of 0.0396 km^2 [0.0153 mi^2], appropriate for F–18s, has been used by the staff. Use of the effective area for any other aircraft would make an insignificant difference in the estimated annual frequency of aircraft crash onto the proposed ISFSI.

The staff searched the website <http://www.chinfo.navy.mil/navpalib/factfile/aircraft/air-fa18.html> of the U.S. Navy and found that the F–18 is a twin-engine aircraft. It is expected that the crash rate of a twin-engine, high-performance aircraft would be less than a single-engine aircraft, such as an F–16. The crash rate given in Table 4.8 of Kimura, et al. (1996) for F–16s and F–15s

(a twin-engine aircraft) indicates that the F–15 has a lower crash rate than F–16. Similarly, it is expected that the crash rate for an F–18 would be less than or equal to that for an F–16 aircraft. Therefore, in the absence of specific information, assuming an F-16 crash rate of 4.378×10^{-8} per km [2.736×10^{-8} per mi] for an F-18 is acceptable. This assumption is also considered valid because the potential crash of a military aircraft traversing route VR–249 would produce a small contribution to the overall aircraft crash hazard at the Diablo Canyon ISFSI.

Although the centerline of the route VR–249 is approximately 3.2 km [2 mi] offshore, PG&E assumed the width of the route for estimating the crash hazard as 1.6 km [1 mi]. This assumption is conservative because it places all 75 flights in a 1.6-km- [1-mi-] wide corridor centered over the proposed ISFSI site for crash hazard estimation purpose. In reality, some of the aircraft would fly further away from the proposed ISFSI site. The staff also used a width of the route equal to 1.6 km [1 mi]. Using these parameters, the estimated crash hazard of aircraft flying route VR–249 is approximately 3.1×10^{-8} per year.

The staff reviewed the data and analysis presented by the applicant with respect to the potential hazards of aircraft flights in military training route VR–249. The staff found them to be acceptable because:

- Adequate information has been presented to describe the potential hazard;
- an acceptable methodology has been used to estimate the crash potential; and
- PG&E conservatively used the crash rate of a single-engine aircraft, the F–16, which is expected to have a higher crash rate than the twin-engine F–18, which is the aircraft primarily used on this route. In addition, PG&E conducted a sensitivity analysis by doubling and tripling the crash rate used to show that the crash rate for this route has only a minor effect on the cumulative aircraft crash hazard for the proposed ISFSI.

Probability Acceptance Criterion for Aircraft Crash Hazards for the Diablo Canyon Independent Spent Fuel Storage Facility

NUREG–0800 Section 3.5.1.6, "Aircraft Hazards" provides the methodology to estimate the annual frequency of a crash of an aircraft onto a nuclear power plant. An operating nuclear power plant requires active systems to control the dynamic nuclear and thermal processes that occur in the conversion of nuclear energy into thermal power. In the event of a mishap, large amounts of thermal energy within the reactor core can be affected. Emergency cooling systems are provided as part of a reactor facility design to avoid core damage or meltdown and the release of radioactive material into the environment.

Compared to a nuclear reactor facility, an ISFSI is a passive system that does not have complex control requirements and that has contents with relatively low thermal energy. Therefore, potential fuel damage and the associated radioactive source terms from a potential accident are significantly less than those expected from a potential accident at a nuclear reactor facility. As a result, the estimated consequences from a potential accident at an ISFSI are less severe than from a potential accident at a nuclear reactor facility. Therefore, the staff concludes that a frequency of 1×10^{-6} crashes per year is an appropriate acceptance criterion for evaluating aircraft crash hazards at the proposed Diablo Canyon ISFSI.

Summary of Review and Discussion

PG&E examined past and present activities in connection with potential hazards from the crash of both civilian and military aircraft flying in the vicinity of the proposed Diablo Canyon ISFSI. The activities examined include aircraft taking off and landing at San Luis Obispo County Regional Airport, Oceano County Airport, Camp San Luis Obispo Heliport, and Vandenberg Air Force Base; aircraft flying Airway V–27; and military aircraft flying in route VR–249. The applicant provided sufficient information and used acceptable methods to evaluate the potential hazard to the proposed ISFSI from an aircraft crash. The staff reviewed the scenarios, data, information, and analyses presented by PG&E in connection with the proposed facility and also carried out independent confirmatory analyses in selected cases, as presented in the previous section of this SER. The confirmatory analyses relied on some different assumptions from those applied by PG&E.

Summarizing the staff review, the crash frequencies for aircraft are given in Table 15-2. As indicated in the discussion of aircraft hazards within this section, these frequencies are estimated on the basis of several elements that determine the overall likelihood that each specific type of aircraft operation may lead to an impact at the proposed facility. Typically, these elements include measures that reflect traffic density (e.g., flights per year), a crash rate (e.g., crashes per mile), effective target area, as well as width of the flying corridor. Other factors, such as human errors in aircraft design, fabrication, or maintenance, also influence the estimated probabilities but have not been addressed explicitly since their effects are inherently taken into account through the use of historically established crash rate data.

Table 15-2. Estimated Annual Frequency of Aircraft Crashes at the Diablo Canyon Independent Spent Fuel Storage Installation

Source	Estimated Annual Frequency (Crashes/Year)	
	PG&E	U.S. NRC
Aircraft taking off and landing at San Luis Obispo County Regional Airport (SLOC Airport)	0	~0
Aircraft taking off and landing at other nearby airports	0	-0
Aircraft flying Airway V–27 using SLOC Airport •Commercial Aviation •General Aviation	 2.59×10^{-9} 7.7×10^{-7}	 3.8×10^{-9} 2.5×10^{-7} to 5.9×10^{-7}
Aircraft flying Airway V–27 not using SLOC Airport	6.53×10^{-9}	9.5×10^{-9}
Aircraft flying military training route VR–249	4.68×10^{-8}	3.1×10^{-8}
Cumulative Aircraft Crash Hazard	8.25×10^{-7}	2.9×10^{-7} to 6.3×10^{-7}

The estimated crash frequency values determined by the staff, as listed in Table 15-2, may be different from those determined by PG&E because of the sensitivity or confirmatory calculations performed by the staff. PG&E has used more conservative values than suggested in the DOE

Standard (U.S. Department of Energy, 1996) for skid distance of a crashing general aviation aircraft. Consequently, the calculated effective area and, in turn, the estimated annual crash frequency are higher and more conservative. The values determined by PG&E have been accepted by the staff as reasonable. Based on the information presented in Table 15-2, which demonstrates that the conservative estimates of aircraft crash probabilities are below the threshold probability criterion of 1×10^{-6} crashes per year for facilities of this type, the staff concludes that the analysis of aircraft crash hazards for civilian and military aircraft and ordnance for the proposed Diablo Canyon ISFSI is acceptable.

Future Developments

PG&E estimated the projected growth of civilian flights based on the FAA long-range forecast (Federal Aviation Administration, 1999). Commercial aircraft operations include air carrier and commuter/air taxi takeoffs and landings at all United States towered and nontowered airports. Based on the FAA forecasts, the commercial aircraft operations are projected to increase from 28.6 million in 1998, to 36.6 million in 2010, and to 47.6 million in 2025. Therefore, commercial aviation operations in the United States are projected to increase by 66 percent by 2025.

The annual general aviation operations (takeoffs and landings) at all towered and nontowered airports in the United States are projected to increase from 87.4 million in 1998 to 92.8 million in 2010 and to 99.2 million in 2025 (Federal Aviation Administration, 1999). Therefore, the FAA projects an increase of general aviation traffic by 14 percent by 2025.

PG&E has discussed the long-term trend of military aviation to project the estimated aircraft crash probability for the proposed ISFSI. The FAA predicts that the military air traffic would not increase appreciably, if at all, in the foreseeable future. Based on the projections of the FAA, the number of military aircraft handled by the FAA en route to traffic control centers will remain constant at 4.2 million from 1998 through 2025.

Based on the estimated annual frequencies listed in Table 15-2 and the increase in commercial and general aviation traffic projected by the FAA, the annual frequency of aircraft crash onto the proposed ISFSI would increase to 9.40×10^{-7} by 2025, as calculated by PG&E. Applying these same growth factors to the estimated crash probability of commercial and general aviation aircraft, the staff estimates that the crash frequency will increase to 3.4×10^{-7} to 7.2×10^{-7} per year by 2025, from the range of 2.9×10^{-7} to 6.3×10^{-7} shown in Table 15-2.

Conclusion

Based on the information and analysis provided by PG&E, the staff concludes that the cumulative probability of a civilian or military aircraft crashing at or affecting the proposed ISFSI is below the threshold probability criterion of 10^{-6} per year determined to be acceptable for these types of facilities. Therefore, there is reasonable assurance that civilian or military air crashes will not pose a hazard to the proposed Diablo Canyon ISFSI.

15.1.2.12 Accidents at Nearby Sites—Missile Testing at Vandenberg Air Force Base

In its responses to staff questions, PG&E provided information regarding operations at the Vandenberg Air Force Base (Pacific Gas and Electric Company, 2003d). The base is located approximately 56 km [35 mi] south-southeast of the proposed Diablo Canyon ISFSI.

Approximately 15 to 20 missiles are tested each year at this base. Missiles are fired in directions ranging from due west to southeast. Therefore, the flight paths of these missiles do not come near the proposed ISFSI. Additionally, intercontinental ballistic missiles are tested at Vandenberg Air Force Base. They are launched from sites at the northern part of the base and typically fly due west. Typical launches for spacelift missions are carried out at sites on the southern part of the base and fly in a southerly direction. Polar orbit launches at this base are also carried out in a southerly direction. Based on the information from the Base Chief Safety Officer, the most northerly missile launch site is approximately 40 km [25 mi] south of the proposed ISFSI. Vandenberg Air Force Base is also a designated alternate landing site for space shuttles, although the base has not been used yet for that purpose. The landing approach is west to east and does not bring the shuttle within 48 km [30 mi] of the proposed ISFSI. Therefore, the planned flight paths for missile tests and space shuttles to and from Vandenberg Air Force Base are always in a direction away from the proposed Diablo Canyon ISFSI site.

Only a small fraction of the missiles tested deviate from the intended trajectories. If a missile after launch deviates from its planned flight path, the missile is destroyed before the debris path exceeds a narrow preplanned window. Therefore, the probability of missiles launched from Vandenberg Air Force Base striking any safety-related SSCs is negligibly small.

The staff reviewed the information with respect to potential hazards of missile testing at Vandenberg Air Force Base. The staff found the information acceptable because:

- Verifiable information from the U.S. Air Force was used to determine the number of missile tests carried out annually and their intended flight paths;

- intended flights paths are always away from the proposed ISFSI site; and

- The U.S. Air Force uses avoidance as one of the primary safety measures to protect facilities.

Based on the foregoing information, there is reasonable assurance that different missile tests and potential space shuttle landings at Vandenberg Air Force Base will not pose a hazard to the proposed ISFSI because (1) the selected flight paths are away from the proposed ISFSI site and (2) several low-probability events would need to occur before a missile or the space shuttle would hit the proposed ISFSI.

15.1.2.13 Leakage Through Confinement Boundary

Section 8.2.7 of the ISFSI SAR evaluates the potential consequences of leakage resulting from a confinement boundary accident. The potential consequences of this postulated accident are determined by assuming that 100 percent of the cladding for the fuel rods have ruptured and the MPC pressure boundary has been breached. The staff has previously determined that the methodology used to assess this postulated accident is acceptable and that there are no consequences that affect the public health and safety so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The Diablo Canyon ISFSI Technical Specifications will impose similar conditions and limits on spent fuel storage so that the staff's previous conclusions are applicable in this case. Moreover, NUREG–1536 (U.S. Nuclear Regulatory

Commission, 1997, Chapter 7, Section V.2) indicates that casks closed entirely by welding do not require seal monitoring. The MPC, which is the confinement system for the HI-STORM 100 System, is closed using a welded seal. As a result, the staff finds the applicant proposal not to provide monitoring of the confinement barrier for the HI-STORM 100 System acceptable because the casks will be loaded, welded, inspected, and tested in accordance with appropriate procedures. The NRC staff of the Spent Fuel Project Office has issued Interim Staff Guidance document ISG-18, which also addresses welded steel canisters, including the Holtec HI-STORM 100 MPCs. In ISG-18, the staff concludes that there is reasonable assurance that no credible leakage would occur from final closure welds of austenitic stainless steel canisters.

15.1.2.14 Loading of an Unauthorized Fuel Assembly

Section 8.2.9 of the ISFSI SAR indicates that loading of an unauthorized fuel assembly into the MPC will not occur because of the Technical Specifications and administrative procedures that will be implemented during loading operations. The Technical Specifications and administrative procedures are discussed in Chapters 10 and 16 of this SER.

15.1.2.15 Partial Blockage of Multi-Purpose Canister Vent Holes

Section 8.2.13 of the SAR evaluates the potential consequences of the partial blockage of the MPC vent holes. The potential consequences of this postulated accident were determined by assuming that only the minimum semicircular area of the vents are credited in the thermal models. The staff previously determined that the methodology used to assess this postulated accident is acceptable and partial blockage of the MPC vent holes has no effect on the structural, confinement, and thermal analyses of the MPC so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The Diablo Canyon ISFSI Technical Specifications will impose similar and appropriate limits and conditions so that the staff's previous conclusion for the Holtec HI-STORM 100 system is applicable to the Diablo Canyon ISFSI.

15.1.2.16 100-Percent Fuel Rod Rupture

Section 8.2.14 of the ISFSI SAR evaluates the potential consequences of 100-percent fuel rod rupture within the MPC. The potential consequences of this postulated accident were determined by assuming that the fission-product gases and fill gas are released from the fuel rods into the MPC cavity. The staff has previously determined that the methodology used to assess this postulated accident is acceptable and 100-percent fuel rod rupture within the MPC has no effect on the shielding, criticality, and thermal analyses of the MPC so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The Diablo Canyon ISFSI Technical Specifications will impose similar and appropriate limits and conditions so that the staff's previous conclusion for the Holtec HI-STORM 100 system is applicable to the Diablo Canyon ISFSI.

15.1.2.17 Transmission Tower Collapse

Section 8.2.16 of the ISFSI SAR addresses the potential collapse of two 500-kV transmission towers that are in the vicinity of the ISFSI storage area and the CTF. The transporter will be designed to protect the transfer cask from the direct impact of a collapsing tower. As a result,

an analysis of a transmission tower collapsing on the loaded HI-TRAC 125 Transfer Cask is not necessary.

Two tower-collapse scenarios were evaluated (Holtec International, 2001f). The first scenario was a tower collapse onto the CTF, with the tower directly impacting the lid of an MPC, which has been lowered into the HI-STORM 100 System storage overpack located within the confines of the CTF shell. The second scenario was a tower collapse onto a HI-STORM 100SA storage cask anchored to the ISFSI storage area pad.

Using an explicit finite element modeling method, PG&E determined that the maximum impact force on the MPC lid was 1.9 MN [427 kips], and for the anchored HI-STORM 100SA storage cask, 2.4 MN [534 kips]. In the case of the MPC lid, this impact force is bounded by previously evaluated tornado missile impact loads, as approved by the staff for the HI-STORM 100 system (U.S. Nuclear Regulatory Commission, 2002a,b). For the anchored HI-STORM 100SA storage cask, the impact force is predominantly oriented in the vertical direction. The horizontal component of the tower collapse impact force on the anchored HI-STORM 100SA storage cask, 0.4 MN [93 kips], is bounded by previously evaluated tornado missile impact loads approved by the staff for the HI-STORM 100 system. The vertical component of the tower collapse impact force on the anchored HI-STORM 100 SA storage cask, when converted into an equivalent gravity load, is also bounded by the previously reviewed and accepted equivalent gravity load for a cask drop (i.e., 45 g).

Even though an analysis of a collapsing tower impact with the HI-TRAC 125 Transfer Cask was not performed, the potential impact forces would be similar to those calculated for the MPC and anchored HI-STORM 100SA storage cask. Because these impact loads are bounded by previously reviewed and accepted loading conditions for the HI-STORM 100 system, the staff has determined that a separate analysis is not needed.

15.1.2.18 Nonstructural Failure of a Cask Transfer Facility Lift Jack

Section 8.2.17 of the ISFSI SAR evaluates a postulated failure of a CTF lift jack. The CTF lifting mechanism is configured with three lifting jacks, and the postulated lift jack failure evaluation assumes that only one of these will fail at any given time. If the failed mechanism cannot be repaired within 22 hours, (which corresponds to the time determined by analysis for the short-term fuel cladding temperature limits to be reached due to the diminished convective cooling efficiency of the HI-STORM 100SA storage cask when located within the CTF), the MPC will be returned to the HI-TRAC 125 Transfer Cask and the storage cask removed from the CTF so the necessary repairs can be made. This will be a requirement of the Diablo Canyon ISFSI Technical Specifications.

The design of the CTF lifting mechanism is such that the three lifting jack power screws are always loaded in tension. Because of this tension loading-only design feature, buckling failure of the lifting jack power screws, either singly or in combination, is unlikely.

15.1.2.19 Accidents Associated with Pool Facilities

The proposed ISFSI will use dry storage technology only; there will be no pool at the proposed ISFSI. Therefore, accidents associated with pool facilities are not applicable for the Diablo Canyon ISFSI.

15.1.2.20 Building Structural Failure and Collapse onto Structures, Systems, and Components

Section 4.4.5 of the ISFSI SAR evaluates the CTF for response to the design criteria identified in Chapter 3, "Principal Design Criteria" of the SAR. The CTF is designed to survive these events. (See also SER Section 15.2.2.18, "Transmission Tower Collapse"). Therefore, an accident involving structural failure of the facility is not applicable.

15.1.2.21 Hypothetical Failure of the Confinement Boundary

The HI-STORM 100 System MPC is a seal-welded pressure vessel, designed, fabricated, and tested in accordance with ASME Code requirements and acceptable alternative methods, as described in the ISFSI SAR. The MPCs have redundant welds to ensure that radioactive fuel is confined. The ISFSI SAR and HI-STORM 100 System FSAR have demonstrated that the MPC would maintain its integrity and the fuel would be adequately protected under site-specific and generic design-basis normal, off-normal, and accident conditions. As discussed in Chapter 9 of this SER, the dose (at the owner-controlled area boundary) calculated from a hypothetical failure of the confinement boundary is below the dose limit specified in 10 CFR §72.106(b).

15.2 Evaluation Findings

The applicant has provided acceptable analyses of the design and performance of SSCs important to safety under credible off-normal events and accident scenarios. The following summarizes the findings of the staff that pertain to the off-normal event and accident review.

<u>Off-Normal Events</u>

PG&E has committed to design the cask transporter so it will have redundant drop protection features and will conform to the requirements of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980), American National Standards Institute (ANSI) N14.6 (American National Standards Institute, 1993), and ASME B30.9-1996 (ASME International, 1996). The staff previously determined that a specific limit on cask lift height during transfers between the FHB/AB, CTF, and the storage pads is not necessary if these cask transporter design requirements are met (U.S. Nuclear Regulatory Commission, 2002a). As a result, an evaluation of a cask drop less than the design allowable height is not required.

The staff has previously determined that the HI-STORM 100 System storage cask provides adequate heat removal capacity under partial vent blockage conditions so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met and the environmental characteristics of the site are bounded by the corresponding design criteria (see Section 6.1.3 of this SER). The staff finds that the Diablo Canyon ISFSI Technical Specifications will impose appropriate limits and that the environmental characteristics of the ISFSI site are within the corresponding design criteria, such that the staff's previous conclusions for the Holtec HI-STORM 100 system are also applicable to the Diablo Canyon ISFSI. In addition, the Diablo Canyon ISFSI Technical Specifications include surveillance requirements for ensuring that the cask heat removal system is operational during storage (i.e., the air ducts are inspected every 24 hours to ensure that the ducts are free of blockages). In the event that the HI-STORM 100SA storage cask air inlet ducts are found to be partially blocked, the blockage will be removed within one operating shift.

The staff finds that there is reasonable assurance that important to safety functions will not be affected for the proposed cask system or the proposed ISFSI due to failure of instrumentation.

The staff finds that potential vehicular impact will not impair the ability of the SSCs to maintain subcriticality, confinement, and sufficient shielding of the stored fuel.

The staff finds that the applicant's evaluation of loss of electrical power as an off-normal event is acceptable and concludes that there is reasonable assurance that Diablo Canyon ISFSI operations can be conducted without endangering the health and safety of the public.

The staff finds that the applicant's assessment of cask transporter off-normal operation is acceptable and concludes that there is reasonable assurance that Diablo Canyon ISFSI operations can be conducted without endangering the health and safety of the public.

The staff previously determined that the HI-STORM 100SA storage and HI-TRAC 125 Transfer Casks provide adequate heat removal capacity during off-normal ambient temperature conditions so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The staff finds that the Diablo Canyon ISFSI Technical Specifications will impose similar and appropriate limits and conditions so that the staff's previous conclusion for the Holtec HI-STORM 100 system is applicable to the Diablo Canyon ISFSI.

The staff previously determined that the methodology used to assess off-normal pressure within the MPC is acceptable and that there are no consequences that affect the public health and safety so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The staff finds that the Diablo Canyon ISFSI Technical Specifications will impose similar and appropriate limits and conditions so that the staff's previous conclusion for the Holtec HI-STORM 100 system is applicable to the Diablo Canyon ISFSI.

Accidents

The staff has previously determined that cask tip-over events need not be considered if the storage cask anchorage system and the storage pad are sufficiently designed to preclude such an event (U.S. Nuclear Regulatory Commission, 2002a,b). The staff finds that the design of the storage pads and cask anchorage system for the Diablo Canyon ISFSI is acceptable, and is sufficient to prevent a cask tip-over accident for the spectrum of seismic events evaluated for the site. The staff's structural evaluation of the pad and cask anchorage system can be found in Section 5.1.3 of this SER.

PG&E has committed to design the cask transporter such that it will have redundant drop protection features and conform to the criteria of NUREG–0612 (U.S. Nuclear Regulatory Commission, 1980), ANSI N14.6 (American National Standards Institute, 1993), and ASME B30.9-1996 (ASME International, 1996). The staff previously determined that a specific limit on cask lift height during transfers between the FHB/AB, CTF, and the storage pads is not necessary if these cask transporter design requirements are met (U.S. Nuclear Regulatory Commission, 2002a). As a result, an evaluation of a cask drop is not required.

The staff finds that the information provided by the applicant is sufficient to characterize flooding as a noncredible accident at the Diablo Canyon ISFSI. As discussed in Section 2.1.4, "Surface Hydrology," of this SER, PG&E has adequately demonstrated that local natural and man-made drainage systems are sufficient to prevent flooding of the ISFSI pad site and CTF.

The staff reviewed the information provided by the applicant regarding potential fire and explosion hazards at the proposed facility. The staff finds that the design basis parameters for all credible on-site fire and explosion hazards will not be exceeded and that the SSCs will meet all subcriticality, confinement, and shielding requirements for the stored fuel.

The staff concludes that earthquake-induced damage of the spent fuel while in transit from the power plant to the CTF is not a credible hazard, based on the low probability of the event and the limited frequency and duration of the transfers.

The staff finds that the design of the CTF concrete structure and its reinforcement satisfies the applicable codes and standards for all design basis accident loads.

The staff finds that the design basis loading conditions for the Diablo Canyon ISFSI are enveloped by the loading conditions considered in the HI-STORM 100 System FSAR (Holtec International, 2002). As documented in the HI-STORM 100 System SER, the structural analysis shows that the structural integrity of the HI-STORM 100 System is maintained during all credible loads. Based on the results presented in the HI-STORM 100 System FSAR, the stresses in the storage cask and anchorage structures during the most critical load combinations are less than the allowable stresses of ASME Boiler and Pressure Vessel Code, Section III (ASME International, 1995b) for the materials to be used.

The staff finds that the Diablo Canyon ISFSI storage cask anchorage system was designed to meet the ductile anchorage provision of the proposed Draft Appendix B for ACI 349-97 for the most critical load combinations.

The staff previously determined that the methodology used to assess the loss of neutron shielding accident is acceptable and the short-term fuel cladding and other component temperature limits, the MPC accident internal pressure, and the accident dose limits defined by 10 CFR §72.106 are not exceeded so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The staff finds that the Diablo Canyon ISFSI Technical Specifications will impose similar and appropriate limits and conditions so that the staff's previous conclusions for the Holtec HI-STORM 100 system are applicable to the Diablo Canyon ISFSI. In the event that the HI-TRAC 125 Transfer Cask loses its neutron shielding, appropriate recovery operation procedures will be implemented.

The staff finds that the maximum reduction in ISFSI radiation shielding thickness, material shielding effectiveness, or loss of temporary shielding in all possible shielding areas caused by postulated on-site explosion events has been adequately evaluated by the applicant. Therefore, the information and analysis presented by the applicant provide reasonable assurance that the dose to any individual beyond the owner-controlled area will not exceed the limits specified in 10 CFR §72.106(b) and the occupational exposures from accident recovery operations will not exceed the limits specified in 10 CFR Part 20.

The HI-STORM 100 System FSAR (Holtec International, 2002, Figure 11.2.6) indicates that a total cask decay heat load of 30 kW [102,360 BTU/hr], which bounds the cask decay heat load specified for the Diablo Canyon ISFSI, will not cause the short-term cladding temperature limit for the spent nuclear fuel to be exceeded for 45 hours under adiabatic conditions. Moreover, the internal pressure limit for the MPC is not exceeded within the 45-hour timeframe for this condition. In the event that a HI-STORM 100 System storage cask is subjected to conditions that thermally insulate its exterior (e.g., encased within soil as the result of a landslide), appropriate recovery operation procedures will be implemented.

The staff previously determined that the methodology used to estimate the time required to reach the short-term, fuel-cladding temperature limit of spent nuclear fuel stored in the HI-STORM 100SA storage cask subjected to 100-percent blockage of the air inlet ducts is acceptable (U.S. Nuclear Regulatory Commission, 2002a,b). For the bounding values of decay heat load of 30 kW [102,360 BTU/hr] and insolation of 834 w/m^2 [800 g-cal/cm^2] per day (387 W/m^2 [123 BTU/hr-ft^2]), the short-term cladding temperature limit for the spent nuclear fuel will not be exceeded for 72 hr when the HI-STORM 100SA storage cask air inlet ducts are 100-percent blocked. Moreover, the internal pressure limit for the MPC is not exceeded within the 72-hour time frame for this condition. Furthermore, the Diablo Canyon ISFSI Technical Specifications includes surveillance requirements for ensuring that the cask heat removal system is operational during storage (i.e., the air ducts are inspected every 24 hours to ensure that the ducts are free of blockages). In the event that the HI-STORM 100SA storage cask air inlet ducts are found to be 100-percent blocked, appropriate recovery operation procedures will be implemented.

The staff reviewed the information provided by the applicant, evaluated the analyses of potential hazards from design-basis tornadoes and tornado missiles at the proposed facility, and conducted a confirmatory analysis. The staff concludes that a tornado or tornado-generated missile would not impair the ability of the SSCs to maintain subcriticality, confinement, and sufficient shielding of the stored fuel.

The staff finds that the applicant has adequately demonstrated that the cumulative probability of occurrence of civilian and military aircraft crashes, and ordnance accidents is below the threshold probability criterion of 1×10^{-6} crashes per year. As a result, the staff concludes that civilian and military aircraft crashes, and ordnance accidents at the Diablo Canyon ISFSI are not credible events and require no further evaluation.

The staff finds with reasonable assurance that different missile tests and potential space shuttle landings at Vandenberg Air Force Base will not pose a hazard to the proposed facility because: (1) the selected flight paths are away from the proposed ISFSI site, and (2) several low-probability events would need to occur before a missile or the space shuttle would hit the proposed ISFSI.

The staff has previously determined that the methodology used to assess leakage through the confinement boundary is acceptable and that there are no consequences that affect the public health and safety so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The staff finds that the Diablo Canyon ISFSI Technical Specifications will impose similar and appropriate limits and conditions so that the staff's previous conclusion for the Holtec HI-STORM 100 system is applicable to the Diablo Canyon ISFSI. Moreover, NUREG–1536 (U.S.

Nuclear Regulatory Commission, 1997, Chapter 7, Section V.2) indicates that casks closed entirely by welding do not require seal monitoring. The MPC, which is the confinement system for the HI-STORM 100 System, is closed using a welded seal. As a result, the staff finds the applicant's proposal not to provide monitoring of the confinement barrier for the HI-STORM 100 System acceptable because the casks will be loaded, welded, inspected, and tested in accordance with appropriate procedures.

Section 8.2.9 of the SAR (Pacific Gas and Electric Company, 2003) indicates that an unauthorized fuel assembly will not be loaded into the MPC because of the technical specifications and administrative procedures that will be implemented during loading operations. These technical specifications and administrative procedures are discussed in Chapters 10 and 16 of this SER.

The staff previously determined that the methodology used to assess partial blockage of the MPC vent holes is acceptable and this postulated accident has no effect on the structural, confinement, and thermal analyses of the MPC so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The staff finds that the Diablo Canyon ISFSI Technical Specifications will impose similar and appropriate limits and conditions so that the staff's previous conclusion for the Holtec HI-STORM 100 system is applicable to the Diablo Canyon ISFSI.

The staff previously determined that the methodology used to assess the potential consequences of a 100-percent fuel rod rupture within the MPC is acceptable and this postulated accident has no effect on the shielding, criticality, and thermal analyses of the MPC so long as the fuel specifications and loading conditions as defined in the HI-STORM 100 System CoC and SER (U.S. Nuclear Regulatory Commission, 2002a,b) are met. The staff finds that the Diablo Canyon ISFSI Technical Specifications will impose similar and appropriate limits and conditions so that the staff's previous conclusion for the Holtec HI-STORM 100 system is applicable to the Diablo Canyon ISFSI.

The staff finds that the impact loads associated with the two postulated tower-collapse scenarios are bounded by previously reviewed and accepted loading conditions (U.S. Nuclear Regulatory Commission, 2002a,b).

The design of the CTF lifting mechanism is such that the three lifting jack power screws are always loaded in tension. Because of this tension loading-only design feature, the staff find that a buckling failure of the lifting jack power screws, either singly or in combination, would not occur.

The staff finds, based on information provided by the applicant, that an accident involving structural failure of the facility is not applicable.

Based on the information provided, the staff finds that a postulated failure of the confinement boundary would result in offsite accident doses below the dose limits specified in 10 CFR §72.106(b) because the HI-STORM 100 System MPC is a seal-welded pressure vessel, designed, fabricated, and tested in accordance with the applicable codes and standards.

In summary, the PG&E analyses of off-normal and accident events demonstrate that the proposed Diablo Canyon ISFSI will be sited, designed, constructed, and operated so that during all credible off-normal and accident events, public health and safety will be adequately protected. Based on analyses submitted by the applicant and independent confirmatory analyses performed by the staff, the staff finds that the proposed ISFSI will maintain subcriticality, maintain confinement, and provide sufficient shielding for all credible off-normal events and accident scenarios consistent with the requirements of 10 CFR §72.92, §72.94, §72.98(a), §72.98(b), §72.98(c) §72.102(f), §72.106(b), §72.122(b), §72.122(c), §72.122(h), §72.122(i), §72.122(l), §72.124(a), and §72.128(a)(2).

15.3 References

AirNav, LLC. Morganville, NJ:. http://www.airnav.com. 2003.

American Concrete Institute. *Code Requirements for Nuclear Safety Related Concrete Structures.* ACI 349-97. Detroit, MI: American Concrete Institute. 1998

American National Standards Institute. *Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More.* ANSI N14.6–1993. Washington, DC: American National Standards Institute. 1993.

ASME International. *ASME Boiler and Pressure Vessel Code, Section III, Division 1.* New York, NY: ASME. 1995b.

ASME International. *ASME Boiler and Pressure Vessel Code, Section II.* New York, NY: ASME. 1995a.

ASME International. *SLINGS Safety Standard for Cable-Ways, Cranes, Derricks, Hoists, Jacks, and Slings.* ASME B30.9–1996. New York, NY: ASME. 1996.

ENERCON Services Inc. *Cask Transfer Facility (Reinforced Concrete).* PGE–009–CALC–002, Rev 1. Tulsa, OK: ENERCON Services Inc. December 17, 2001a.

ENERCON Services Inc. *ISFSI Cask Storage Pad Seismic Analysis.* PGE–009–CALC–003, Rev 2. Tulsa, OK: ENERCON Services Inc. December 14, 2001b.

ENERCON Services Inc. *ISFSI Cask Storage Pad Steel Reinforcement.* PGE-009-CALC-007. Tulsa, OK: ENERCON Services Inc. March 11, 2003a.

ENERCON Services Inc. *Embedment Support Structure.* PGE-009-CALC-001, Rev 5. Tulsa, OK: ENERCON Services Inc. March 11, 2003b.

Federal Aviation Administration. *FAA Long-Range Aerospace Forecasts Fiscal Years 2015, 2020 and 2025.* FAA–APO–99–5. Washington, DC: Federal Aviation Administration. Office of Aviation Policy and Plans. 1999.

GCR & Associates, Inc. New Orleans, LA: http://www.gcr1.com/5010WEB/default.htm. 2003.

Holtec International. *Evaluation of Site-Specific Wild Fires for the Diablo Canyon ISFSI.* HI–2012615, Marlton, NJ: Holtec International. 2001a.

Holtec International. *Evaluation of Site-Specific Blasts and Explosions for the Diablo Canyon ISFSI.* HI–2002512, Rev 2. Marlton, NJ: Holtec International. 2001b.

Holtec International. *Structural Evaluation of Diablo Canyon Cask Transfer Facility.* HI–2012626, Rev 7. Marlton, NJ: Holtec International. 2001c.

Holtec International. *Analysis of Anchored HI-STORM Casks at the Diablo Canyon ISFSI.* HI–2012618, Rev 5. Marlton, NJ: Holtec International. 2001d.

Holtec International. *Design Basis Wind and Tornado Evaluation for DCPP.* HI–2012497, Rev. 1. Marlton, NJ: Holtec International. 2001e.

Holtec International. *Evaluation of Effects of Lightning and a 500-kV Line Break on Holtec Casks.* HI–2002559. Marlton, NJ: Holtec International. 2001f.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System Revision 1 (HI-STORM 100 Cask System).* Volumes I and II, Revision 1. HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International. 2002.

Kimura, C.Y. R.E. Glaser, R.W. Mensing, T. Lin, T.A. Haley, A.B. Barto, and M.A. Stutzke. *Data Development Technical Support Document for the Aircraft Crash Risk Analysis Methodology (ACRAM) Standard.* UCRL–ID–124837. Livermore, CA: Lawrence Livermore National Laboratory. 1996.

McConnell, J.W., Jr., A.L. Ayers, Jr., and M.J. Tyacke. *Classification of Transportation Packing and Dry Spent Fuel Storage System Components According to Importance to Safety.* NUREG/CR–6407. Idaho Falls, ID: Idaho National Engineering Laboratory. 1996.

National Fire Protection Association. *Fire Protection Handbook.* 18th Edition. Quincy, MA: National Fire Protection Association. 1997.

Pacific Gas and Electric Company. *Non-Linear Seismic Sliding Analysis of the ISFSI Pad.* PG&E Calculation 52.27.100.704, Rev. 0. Avila Beach, CA: Pacific Gas and Electric Company 2001a.

Pacific Gas and Electric Company. *Diablo Canyon Power Plant Final Safety Analysis Report Update.* Rev. 14. Avila Beach, CA: Pacific Gas and Electric Company. 2001b.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Safety Analysis Report.* Amendment 2. Letter DIL-03-012, October 3, 2003. Avila Beach, CA: Pacific Gas and Electric Company. 2003.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Response to NRC Request for Additional Information for the Diablo Canyon*

Independent Fuel Storage Installation (TAC No. L23399). Avila Beach, CA: Pacific Gas and Electric Company. October 15, 2002.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Supplemental Blasts and Explosions Responses to Additional NRC Questions for the Diablo Canyon Independent Fuel Storage Installation Application (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company, 2003a. DIL-03-005, March 27, 2003.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Additional Questions Related to Supplemental Blasts and Explosions Responses to Additional NRC Questions for the Diablo Canyon Independent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company, 2003b. DIL-03-008, June 13, 2003.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Additional Questions Related to Supplemental Blasts and Explosions Responses to Additional NRC Questions for the Diablo Canyon Independent Fuel Storage Installation (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company, 2003c. DIL-03-010, July 28, 2003.

Pacific Gas and Electric Company. *Diablo Canyon Independent Spent Fuel Storage Installation: Revised and Supplemental Blasts and Explosions Responses to Additional NRC Questions for the Diablo Canyon Independent Fuel Storage Installation Application (TAC No. L23399).* Avila Beach, CA: Pacific Gas and Electric Company, 2003d. DIL-03-002, February 14, 2003.

Private Fuel Storage Limited Liability Company. Aircraft Crash Impact Crash Impact Hazard at the Private Fuel Storage Facility. Rev. 4. La Crosse, WI: Private Fuel Storage Limited Liability Company. 2000.

San Luis Obispo County Regional Airport. San Luis Obispo, CA: http://www.sloairport.com/flightinfor.html. 2003

Simiu, E. and R.H. Scanlan. *Wind Effects on Structures: An Introduction to Wind Engineersing.* Second Edition. New York, NY: John Wiley and Sons. 1986.

U.S. Department of Energy. *DOE Standard: Accident Analysis for Aircraft Crash Into Hazardous Facilities.* DOE–STD–3014–96. Washington, DC: U.S. Department of Energy. 1996.

U.S. Department of Transportation. *Traffic Safety Facts 2001: 2001 Motor Vehicle Crash Data from FARS and GES.* http://www-nrd/nhtsa.dot.gov/pdf/nrd-30/NCSA/TSFAnn/TSF2001.pdf. National Highway Traffic Safety Administration. 2003a.

U.S. Department of Transportation. *Large Truck Crash Facts 2001.* http://ai.volpe.dot.gov/Carrier Research Results/PDFs/Large Truck Crash Facts 2001.pdf. Federal Motor Carrier Safety Administration. 2003b.

U.S. Navy. United States Navy Fact File: F/18-18 Hornet. Washington, DC: http://www.chinfo.navy.mil/navpalib/factfile/aircraft/air-fa18.html. 2003.

U.S. Federal Aviation Administration. Washington, DC: http://apo.data.faa.gov/faaatadsall.htm. 2003.

U.S. Nuclear Regulatory Commission. *Design Basis Tornados for Nuclear Power Plant.* Regulatory Guide 1.76. Washington. DC: U.S. Nuclear Regulatory Commission. 1974.

U.S. Nuclear Regulatory Commission. *Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants.* Regulatory Guide 1.91. Rev. 1. Washington. DC: U.S. Nuclear Regulatory Commission. 1978.

U.S. Nuclear Regulatory Commission. *Control of Heavy Loads at Nuclear Power Plants.* NUREG-0612. Washington, DC: U.S. Nuclear Regulatory Commission. 1980.

U.S. Nuclear Regulatory Commission. "Standard Review Plan Section 3.5.1.4: Missiles Generated by Natural Phenomena." Rev. 2. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.* NUREG-0800. Washington, DC: U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. 1981a.

U.S. Nuclear Regulatory Commission. "Standard Review Plan Section 3.5.1.6 Aircraft Hazards." *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.* NUREG-0800. Washington, DC: U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. 1981b.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Dry Cask Storage Systems,* NUREG-1536. Washington, DC: U.S. Nuclear Regulatory Commission. 1997.

U.S. Nuclear Regulatory Commission. *Standard Review Plan for Spent Fuel Dry Storage Facilities—Final Report.* NUREG-1567. Washington, DC: U.S. Nuclear Regulatory Commission. 2000.

U.S. Nuclear Regulatory Commission. *10 CFR Part 72 Certificate of Compliance No. 1014, Amendment 1, for the HI-STORM 100 Cask System.* Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. July 15, 2002a.

U.S. Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* Amendment 1. Docket No. 72-1014. Washington, DC: U.S. Nuclear Regulatory Commission. 2002b.

Original

From:	Ader, Charles
To:	Flanders, Scott; Shuaibi, Mohammed
Cc:	Mrowca, Lynn; Schaperow, Jason; Hawkins, Kimberly
Subject:	FW: Interim DRAFT Spent Fuel Pool Scoping Study sent to ACRS today
Date:	Tuesday, April 23, 2013 5:41:08 PM
Attachments:	Kathy Harvey Gibson.vcf SFPSS DivDir CompiledCommentskcmaster.docx SFPSS Doc (April ACRS).pdf —— previously provided
Importance:	High

Kim previously provided you the RES response to our comments. In a discussion with Kathy Gibson yesterday, she ask if there were any concerns that we are aware of that would prevent NRO concurrence on this report. She expects the report to come to NRO by the end of the month for office concurrence and is looking for early indication of major issues.

If I missed someone that provided comments previously, please forward this e-mail.

NRO-OK

-----Original Message-----
From: Hawkins, Kimberly
Sent: Friday, April 12, 2013 12:42 PM
To: Flanders, Scott; Shuaibi, Mohammed
Subject: FW: Interim DRAFT Spent Fuel Pool Scoping Study sent to ACRS today
Importance: High

Not sure if you saw this... Responses to our comments on the study... some were addressed and resulted in revisions to the study; for others, RES provided its response.

-----Original Message-----
From: Holahan, Gary
Sent: Friday, April 12, 2013 9:47 AM
To: Hawkins, Kimberly
Subject: FW: Interim DRAFT Spent Fuel Pool Scoping Study sent to ACRS today
Importance: High

From: Tracy, Glenn
Sent: Thursday, April 11, 2013 5:37 PM
To: Holahan, Gary
Subject: FW: Interim DRAFT Spent Fuel Pool Scoping Study sent to ACRS today

From: Sheron, Brian
Sent: Thursday, April 11, 2013 4:51 PM
To: Johnson, Michael; Weber, Michael; Leeds, Eric; Wiggins, Jim; Tracy, Glenn; Haney, Catherine; Satorius, Mark; Skeen, David
Subject: FW: Interim DRAFT Spent Fuel Pool Scoping Study sent to ACRS today
Importance: High

FYI.

From: Gibson, Kathy
Sent: Thursday, April 11, 2013 4:40 PM
To: McGinty, Tim; Ader, Charles; Lombard, Mark; Skeen, David; Correia, Richard; Case, Michael; Thaggard, Mark; Miller, Chris
Cc: McIntyre, David; Burnell, Scott; Richards, Stuart; Lee, Richard; Algama, Don; Blount, Tom; Reis, Terrence; Shear, Gary; Sheron, Brian; West, Steven
Subject: Interim DRAFT Spent Fuel Pool Scoping Study sent to ACRS today
Importance: High

Refer to RES

A5-7

E/54

Gentlemen,
The interim draft SFPSS report was due to the ACRS today at noon and we met that deadline. We are scheduled for an all-day briefing of the ACRS subcommittee on Materials, Metallurgy and Reactor Fuels on the study on May 8. I encourage you and your staff to attend all or part of the briefing as you have time and interest. We will send out an agenda and slide package in advance of the meeting.

I am providing for your information two documents: (1) A list of the division director level comments that you provided in response to my request on 3/22/13 and our responses, and (2) a copy of the version of the report that was sent today to ACRS.

I believe we were able to address your comments in this version of the report. We were not able to incorporate them all directly, but we describe why the study is the way it is and added significant clarifications to add the context that you were seeking.

The study team will now go back to addressing the comments received from your staff and BCs that we were not able to get to before the ACRS deadline. Therefore, revisions to the report will continue.

This email will be forwarded by Don Algama to your staff and BCs that have been involved in the project.

It was just decided by senior management this week that the study report will be released for a 30-day public comment period from about June 10 – July 10 (ACRS full committee meeting). We are evaluating how to accommodate this development within our schedule to have the report finalized by September. As I indicated previously, the offices will have at least one more opportunity to review and concur on the report.

I appreciate your quick review and thoughtful comments on the prior version of the report. I also appreciate all the hard work and effort the team has put into responding to your comments, including late nights and some very animated conversations.
I trust that you will find this version an improvement.

ADAMs links:

View ADAMS P8 Properties
ML13101A168<https://adamsxt.nrc.gov/WorkplaceXT/integrationWebBasedCommand?_commandId=3010&objectStoreName=Main.__Library&id=current&vsId=%7b07CA2DD4-F1D3-4BD8-874F-978C0047FB66%7d&objectType=document>
Open ADAMS P8 Package (Issuance of the DRAFT "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor" for ACRS Review)<https://adamsxt.nrc.gov/WorkplaceXT/getContent?objectStoreName=Main.__Library&id=current&vsId=%7b07CA2DD4-F1D3-4BD8-874F-978C0047FB66%7d&objectType=document>

[cid:image002.jpg@01CE36CE.5AF56E80]

NRR

Comments to be addressed for ACRS SC report:

NRR

1. From a DSS perspective, we believe the report needs to be revised to clearly indicate why the study was done, why we chose the seismic response that we did, and how this compares to what would be expected at our 104 nuclear plants (or at least put in perspective that this is representative of a small subset of U.S. reactor designs). I really liked Rich's characterization in that the message is that we evaluated at the design basis and got no release. We doubled it and got no release, we tripled it and got no release so we went to four times the design basis and finally got a release for a very small number of unmitigated scenarios.

   Response: The report was revised to incorporate the following points that address this comment:
   - The study was done to confirm the results of past studies using state-of-the-art tools, as well as Fukushima insights, in a publicly available study.
   - The study will inform the Tier 3 activity by providing an updated technical basis for any regulatory action and input for the regulatory analysis.
   - The study used design, operational, and location data for a reference site for which we already had information available, a BWR Mark I with an elevated SFP. The report also considered a 1x4 pattern (required after some time after offloading) as well as sensitivity analysis for more favorable loading (1x8) and less favorable loading (checkerboard and uniform) and sensitivities for other key parameters that will provide insights for analysis of other plants.
   - The report was revised to make clearer that a low likelihood beyond design basis seismic event with and without mitigation was chosen to gain risk insights that could not be gained using a less severe seismic initiator. NRC analyzes low likelihood beyond design basis seismic events with and without mitigation to gain insights on the safety margin provided by NRC's regulatory framework.
   - The study concludes that the SFP is robust and not expected to leak, successful mitigation prevents most releases, no early fatalities are expected and individual LCF is low because effective protective actions limits individual exposure. (Note that high and low density *mitigated* moderate leak scenarios in the first week (OCP 1) resulted in releases, all other scenarios that resulted in releases were unmitigated and within the first few months (OCP 1, 2, 3) after shutdown.)

2. DSS also challenges why we are evaluating land contamination since no previous study directly discussed this issue. Considering that the Commission is currently reviewing whether to change its long-standing policy on addressing land contamination, it may be premature to evaluate this particular aspect in the report at this time.

   Response: The study included land contamination to provide inputs to a regulatory analysis. A paragraph has been added to the introduction to describe the study's relationship to the Tier 3 activities and how the study will be used in the current regulatory process. Other analyses did evaluate land contamination, including some directly (e.g., NUREG/CR-6451, NUREG-4982). Land contamination is already part of NRC's current regulatory framework including being used as input in SAMA/SAMDA analyses and is an input to regulatory/backfit analyses as part of the cost benefit analysis. Chapter 7 was revised to distinguish the safety-related individual health effects

NMSS

measures from other measures that are inputs to the cost-benefit analysis for the regulatory analysis.

NMSS

3. The results of this study in Section 11 and in other sections need to be put into context by comparison of the results against some standard such as the Quantitative Health Objectives or Qualitative Safety Goals similar to the comparison to the QHOs of NUREG-1738 results discussed on page 13. Some may argue that is comparing apples to oranges but the QSGs are based on risk to the general public of nuclear power versus other societal risks. This would give the public understandable measures to compare the results against as opposed to results without any context.

   Response: We agree that some level of comparison is appropriate. Section 7 has been rewritten, and now includes the statement, which will also be integrated into Section 11: release

   When the release frequency is considered, the latent cancer fatality risks from the events analyzed in this study are very small, in the 1E-12 to 2E-11 per year range, when using an LNT dose response model. For perspective, the Commission's safety goal policy related to the cancer fatality quantitative health objective (QHO) represents a 2E-6 per year objective for an average individual within 10 miles of the nuclear plant site (NRC, 1983). While the results of this study are scenario-specific (a beyond design-basis seismic event) and related to a single spent fuel pool, staff concludes that since these risks are several orders of magnitude below the QHO, it is unlikely that the results here would contribute significantly to a risk that would challenge the Commission's safety goal policy (NRC, 1986).

4. SRM dated July 16, 2012, documented the ACRS comment to ensure that consequences associated with expedited loading, transfer, and long-term storage need to be considered. While Enclosure 1 to the draft SFPSS indicates those areas have been included, the assessment in Appendix B compares consequence results to NUREG-1864, which does not include assessment of the consequences of expedited transfer to dry casks. Appendix C also does not address expedited transfer in the current context of the term to move all but the newest fuel out of the pool. This fact is pointed out in the SFPSS on Page 4, that the study does not address certain considerations, including expedited discharge of fuel from the pool to dry storage.

   Response: The approach for responding to the SRM was to obtain near term insights in Appendices B and C within the SFPSS project timeframe. This risk comparison template is intended to inform the Tier 3 working group considering expedited transfer of spent fuel. The Tier 3 plan includes 3 phases. Phase 1 will use this study to determine whether a significant safety enhancement could be achieved by expedited transfer and provide inputs to a regulatory analysis. More detailed treatment of these issues may be addressed as part of subsequent phases of the Tier 3 activity on expedited spent fuel transfer, as necessary.

5. Why was land contamination included on the study?

   Response. Please see response to comment #2.

6. The SFPSS should make a recommendation on whether future studies are needed or not and what they would or should entail or point to the Tier 3 effort.

Response: This scope of this study does not include making recommendations for further study. NRR will determine whether further analyses are needed to make any regulatory determinations within NRC's current regulatory framework. A paragraph has been added to the introduction to describe the study's relationship to the Tier 3 activities and how the study will be used in the regulatory process. The following statement has been added to the introduction and results sections of the report:

> Other aspects of SFP risk that have not been informed by this or past studies, may be addressed by future studies, such as the site Level 3 probabilistic risk assessment (PRA), as documented in SECY-11-0089, "Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities," dated July 7, 2011, and the associated staff requirements memorandum; or will be addressed through other inputs to the regulatory decision-making process, as needed.

7. Appendix B-why is the first table included on Page B-3? It does not include any data regarding dry cask storage.

   Response: Appendix B addresses part of the SRM (dated July 16, 2012) to compare the results of the SFPSS with past studies and consider consequences associated with loading, transfer, and long-term storage. Appendix B provides a comparison of SFPSS results to previous spent fuel pool studies and updated analyses from NUREG-1864 Dry Storage Pilot PRA. Staff will revise the introduction to Appendix B to make this clear.

NRO

8. The report needs to describe how its results could be useful in making regulatory decisions on matters including the Japan lessons-learned Tier 3 recommendation on assessment of the transfer of spent fuel to dry-cask storage and recent Commission direction on economic consequences. In responding to this comment, a fuller characterization of the purpose and usefulness of the report should be added, including an explanation of how the study's point-estimate approach is appropriate in the context described above.

   Response: NRR will determine whether further analyses are needed to make any regulatory determinations within NRC's current regulatory framework. A paragraph has been added to describe the study's relationship to the Tier 3 activities and how the study will be used in the regulatory process. Using representative point-estimates with sensitivities for important parameters is appropriate in research studies to be able to gain insights and data for regulatory decision-making in a reasonable period of time.

NRO - OK To release

   The study used design, operational, and location data for a reference site for which we already had information available, a BWR Mark I with an elevated SFP. The report also considered a 1x4 pattern (required after some time after offloading) as well as sensitivity analysis for more favorable loading (1x8) and less favorable loading (checkerboard and uniform) and sensitivities for other key parameters that will provide insights for analysis of other plants.

9. The report needs to describe the relationship between the study results and our current approach to approving nuclear power plant sites and designs. In addition to describing this approach, a column could be added to the assumptions in Chapter 2 to provide context relative to the current regulatory approach for licensing nuclear power plants and

plants' licensing bases. Accordingly, the conclusions could also be reframed to highlight the robustness of our regulatory framework for the safe operation of nuclear power plants, e.g., that mitigation strategies provide a significant reduction in release rates.

Response: NRR will use the study in making related Tier 3 regulatory determinations within NRC's current regulatory framework. A paragraph has been added to describe the study's relationship to the Tier 3 activities and how the study will be used in the regulatory process. The study's conclusions include that successful mitigation generally prevented releases. (Note that there were mitigated scenarios that resulted in releases.)

10. The Staff Requirements Memorandum (SRM) on SECY-08-0029 directed the State-of-the-Art Reactor Consequence Analyses (SOARCA) to use individual cancer fatality risk as its latent cancer health-effects metric. The study should follow the same approach by using this metric and not reporting the total number of cancer deaths. For example, Chapter 7, Table 29 reports total latent cancer fatalities per year. Also, Chapter 11, conclusion 11 states "For scenarios with large releases, significant numbers of latent cancer fatalities are predicted when using a dose-response model based on the linear-no threshold hypothesis; however, this would be a small fraction compared to cancer fatalities from all causes."

    Response: Given the uncertainty of low doses on health effects, LCFs is being removed as a quantitative metric. For clarification, SECY-08-0029 and the related SRM did not "direct" SOARCA to exclude the reporting of LCFs or other potential societal health effects. Rather, the Commission agreed to the staff's recommendation that SOARCA should report individual LCF risk. The basis for reporting individual LCF risk can be found in the Qualitative Safety Goals (QSGs). However, the QSGs also provide the basis for reporting societal health impacts, as they are an important measure of the safety of nuclear power in general. Therefore while LCFs are not quantified in the report, they are still discussed in broad terms. Societal dose as a surrogate provides a reasonable measure for societal health effects and is not subject to the uncertainty of low dose health effects. Societal dose is also an input to cost benefit analyses for backfit/regulatory analyses and SAMA/SAMDA analyses. Chapter 7 was revised to distinguish the safety-related health effects measures other measures that are inputs to the cost-benefit analysis for the regulatory analysis.

11. A memorandum to the Commission dated April 3, 2007 (OUO-SII), stated that the staff would not report land contamination/economic consequences in SOARCA because of modeling and policy issues. SRM-COMPBL-08-0002/COMGBJ-08-0003 directed the staff to develop an improved economic consequence model for the MELCOR Accident Consequence Code System (MACCS). This SRM also stated that the resulting model may be applied to the SOARCA results if so directed by the Commission. The study should follow the same approach by not reporting land contamination.

    Response. Land contamination and economic consequences results from MACCS2 models are routinely used as inputs in NRC's current regulatory framework in backfit/regulatory analyses and, in SAMA/SAMDA analyses, and have been reported in previous research studies (e.g. NUREG/CR-6451, NUREG/CR-4982). Regarding the use of MACCS2 for SAMA analyses, the ASLB has ruled that the models are adequate for the regulatory purpose (Accession No. ML11200A224).

A paragraph has been added to the introduction to describe the study's relationship to the Tier 3 activities and how the study will be used in the regulatory process. Chapter 7 was revised to distinguish the safety-related individual health effects measures from other measures that are inputs to the cost-benefit analysis for the regulatory analysis. NRR will use these measures within NRC's current regulatory framework.

Regarding the memorandum to the Commission dated April 3, 2007, current staff updated its position on MACCS models in Enclosure 9 of SECY-12-0110 stating:

> It is not obvious to current MACCS2 experts at both the NRC and Sandia National Laboratories (SNL) that rehabilitation and clean up, land contamination area, or economic models and results are excessively conservative. Economic results and some land contamination area results are controlled by user inputs and could be biased to be either conservative or nonconservative, depending on the input values selected by the user. A MACCS2 user's guide and code manual is available for reference when deciding various parameter inputs. Other land contamination areas produced by MACCS2 are influenced chiefly by the Gaussian plume and deposition modeling. Based on the 2004 benchmarking study, these values do not appear to have either a conservative or nonconservative bias.

The new economic model is not relevant to this study. It has not been completed and is not available for use at this time. Enclosure 9 of SECY-12-0110 also provides details on this project.

12. Table 3 (the last entry on page 19) includes this sentence: "Vertical spectral accelerations as high as horizontal accelerations are justified on the bases that nearby earthquakes control the ground motions spectra for this event and that the frequencies of interest for the study are frequencies near or above 10 Hz." Provide the basis for the assumption that nearby earthquakes control the estimated ground motions at the reference site.

    Response: The revised report now reads:

    > A few studies (e.g., McGuire, Silva, and Costantino, 2001; ASCE, 1999) indicate that for rock sites and frequencies near and above 10 Hz, and especially nearby seismic sources, vertical spectral accelerations may be as high as or exceed horizontal spectral accelerations. For this study, the frequencies of interest are, for the most part, frequencies near or above 10 Hz. Therefore, the assumption of equal vertical and horizontal spectral accelerations was deemed to be a reasonable starting assumption. This assumption is also supported by seismic hazard de-aggregation with the USGS (2008) model (http://earthquake.usgs.gov/hazards/apps/#deaggint) which indicates that for the seismic bin of interest (high PGA, low likelihood events) the contributors to risk would be moderate magnitude earthquakes at nearby distances.

13. Table 3 (the first entry on page 20) includes this paragraph:

The current seismic assessment uses a model and code generated by the US Geological Survey (USGS, 2008). The USGS 2008 information is being further developed and updated by a group of stakeholders, including the NRC, in a collaborative study which includes (a) the seismic source zone characterization, and (b) the ground motion attenuation models. In addition, the NRC is developing independent methods and computer codes, which will be publicly available when completed, to combine (a) and (b). Although part (a) of this updating effort has been completed in early 2012, part (b) and the computer code development are still ongoing. Therefore, this study used the earlier USGS information instead of the ongoing update program.

a. It seems that the intent of this paragraph is to reference the recently published Central and Eastern United States Seismic Source Characterization (CEUS SSC) model. Instead of saying: "The USGS 2008 information is being further developed and updated by a group of stakeholders, including the NRC, in a collaborative study," the paragraph should reference the CEUS SSC model and note that it is a new seismic source model cosponsored by EPRI, DOE, and NRC. Also, clarify that CEUS SSC is independent of the USGS 2008 model.

b. Change "ground motion attenuation models" to "ground motion prediction equations (GMPEs)" and make the distinction that the GMPE update effort was not part of the CEUS SSC model and it is an industry effort, which is still in progress.

c. Add a sentence to justify the use of the USGS 2008 model for the purposes of this scoping study, since the USGS hazard model is not endorsed by the NRC in licensing new reactors (currently the CEUS SSC model is the NRC approved starting model).

d. Add a disclaimer stating that the use of the USGS hazard is not consistent with the hazard defined in the licensing basis for new reactors.

e. This comment also applies to Section 3.1 (page 29, 2nd paragraph).

Response: The revised report will read (note that for a scoping study of this type we try, to the extent possible, to avoid references to application reviews or licensing-related activities)

> A group of stakeholders, which includes the NRC, is developing a new probabilistic seismic hazard model in a collaborative study which comprises two parts: (1) the seismic source zone characterization and (2) the ground motion attenuation models. In addition, the NRC is developing independent methods and computer codes, which will be publicly available when completed, to combine parts (1) and (2) above. Although part (1) of this updating effort has been completed (NRC, 2012b), it was not completed at the start of this scoping study. In addition, part (2) and the computer code development are still ongoing. Therefore, this study used the existing USGS (2008) model instead of the model in the ongoing program.

14. Table 3 (the first entry on page 22) includes this paragraph:

In general, for an aftershock to cause subsequent additional damage to a structure, it would have to occur much closer to the site than the main event and with characteristics, for example frequency content, that would make the structure especially vulnerable to it. The earthquake ground motion considered in the SFP scoping study is a probabilistic quantity that aggregates motions from events with various magnitudes and distances to the site. For this site, this probabilistic ground motion already tends to be controlled by relatively close events in the larger magnitude range for the credible seismic sources. . This main shock cracks the SFP studied but its structure is still stable after the earthquake and it cracks in a manner that allows for additional loading cycles at this level. Under these conditions, earthquake ground motions greater than those for the main shock would be needed to further damage the SFP. This is unlikely given that the ground motion considered is already controlled by close events with magnitudes near the credible upper magnitudes for the site.

It would be better to just state that current probabilistic seismic hazard analysis (PSHA) models do not consider aftershocks and that is why they were not considered in this study. Otherwise the statements in the above paragraph would lead to the following comments that should be clarified:

a. There is no discussion on the controlling earthquakes and the associated annual exceedance frequencies to support the statement that "[f]or this site, this probabilistic ground motion already tends to be controlled by relatively close events in the larger magnitude range for the credible seismic sources."

b. Aftershocks can be numerous and substantial (especially if the study is considering very low probability events).

c. Aftershocks could in fact be closer to the site than the main shock, and that could be significant since the report stated previously that the estimated ground motions at the reference site are controlled by nearby events.

Response: We verified that the contributing earthquakes are nearby events and the report has been modified to read:

> In general, for an aftershock to cause subsequent additional damage to a structure, it would have to occur significantly closer to the site than the main event as well as spectral accelerations at frequencies that would make the structure vulnerable to the ground motion. For this site, and for events associated with PGAs and spectral accelerations of interest for risk assessment (high PGA, low likelihood events), the main contributors to the ground motion hazard for this site are expected to be moderate magnitude nearby earthquakes (http://earthquake.usgs.gov/hazards/apps/#deaggint). The main event would crack the SFP studied but its structure would be stable after the earthquake and would crack in a manner that is expected to resist additional loading cycles at this level. Under these conditions, earthquake ground motions with damage potential greater than that for the main event would be needed to further damage the SFP. This is thought to be unlikely given that the contributors to the ground motion hazard are already nearby events.

15. Section 3.1 (page 29, 3rd paragraph) mentions the hazard estimates for a rock site. The report should discuss the implications for soil sites, as well as the implications of sites with different controlling earthquakes. Clarify how SFP characteristics vary between different operating plants and what are the implications of this variation.

    Response: The study focuses on, to the extent possible, a site-specific hazard estimate to avoid assumptions that are not realistic. The site chosen is a rock site. Consideration of the items raised would be out of the scope of the work. See also the response to Comment #1.

16. Section 3.1 (page 29, paragraphs 4 to 6) includes bullets that compare the USGS 2008 hazard estimates for the reference site with the LLNL and EPRI results. The report should clarify the purpose of these comparisons

    Response: The report has been revised to read:

    > These comparisons are provided to compare the model used in this scoping study to well-known and extensively documented information sources (LLNL model and EPRI model) that were used in past SFP risk studies.

17. Section 3.1 (page 31, Figures 4 and 5) should indicate in the figure captions that these are hard rock hazard curves.

    Response: The captions have been modified to address the comment.

18. Section 3.2 (page 33, last paragraph) includes this statement: "In addition to the PGA, ground motions at a site are also characterized by their frequency content expressed in terms of response spectra. Based on the USGS 2008 model, a uniform hazard site Ground Motion Response Spectrum (GMRS) (NRC, 2007b) was derived for the GI-199 study and used in this study." It is incorrect to combine the term uniform hazard response spectra with the term GMRS. In addition, Footnote 5 states that "the term GMRS has a specific meaning in the context of Regulatory Guide (RG) 1.208 (NRC, 2007b). In this report, the term GMRS is used more generally." The report should describe how the response spectrum for the selected site was developed. If it is not consistent with the definition of the GMRS in RG 1.208, then use a different name. Clarify whether the response spectrum for the reference site shown in Figure 7 is a uniform hazard response spectrum. In addition, do a global search for "GMRS" because it is used throughout the report.

    Response: The footnote has been deleted. After further examination, it was confirmed that the GMRS in the report is based on the guidance in Regulatory Guide 1.208 used in conjunction with USGS (2008) model. This is clearly noted in the report and repeated often. Use of a different hazard model and maybe a more detailed analysis might produce a somewhat different GMRS. We do not think that the footnote is needed because the assumptions are clearly indicated. Also, as per the response to the comment related to the use of the USGS (2008) model (comment 13) we prefer not to make references to licensing review aspects in a study of this type.

    Nevertheless, when referring to the GMRS, the text in the report will be modified to replace "site GMRS" with "reference GMRS." Also, the text at the end of Section 3.2 and after Table 5 will be modified to read:

In addition to the PGA, ground motions at a site are also characterized by their frequency content expressed in terms of response spectra. Based on the guidance in Regulatory Guide 1.208 (NRC, 2007b) used in conjunction with the USGS 2008 model, mean uniform hazard response spectra were derived to then estimate a reference ground motion response spectra (GMRS) for the GI 199 study. This reference GMRS was subsequently scaled as indicated in Section 3.3 below to obtain the input free-field ground motion response spectra used in this study.

The text at the beginning of Section 3.3 also will be modified to read.

The free-field reference GMRS for horizontal earthquake shaking for this site is based on the response spectrum and PGA used in conjunction with research assessments for GI-199, which utilized the USGS 2008 model. This reference GMRS has a zero-period spectral acceleration (PGA) of about 0.34 g.

19. In Section 3.3 (page 34, 1st and 2nd paragraphs), change "Peach Bottom" to "reference site" and do a global search for further changes because "Peach Bottom" appears in multiple places.

Response: The report will be searched for that and the change made as appropriate, which include the occasions noted in this comment. Note that the report identifies the plant on which the reference plant is based.

20. The second paragraph on page 35 includes this statement:

Vertical spectral accelerations and the vertical PGA are taken to be the same as the horizontal spectral accelerations and PGA. This is assumed on the bases that nearby earthquakes would control the ground shaking spectra for this event and that the frequencies of interest for this study are frequencies above 5 Hz (ASCE, 1999) (McGuire, Silva and Costantino, 2001).

The report should describe how controlling earthquakes were determined.

Response: The report has been revised to read:

A few studies (e.g., McGuire, Silva, and Costantino, 2001; ASCE, 1999) indicate that for rock sites and frequencies near and above 10 Hz, and especially nearby seismic sources, vertical spectral accelerations may be as high as or exceed horizontal spectral accelerations. For this study, the frequencies of interest are, for the most part, frequencies near or above 10 Hz. Therefore, the assumption of equal vertical and horizontal spectral accelerations was deemed to be a reasonable starting assumption. This assumption is also supported by seismic hazard de-aggregation with the USGS (2008) model (http://earthquake.usgs.gov/hazards/apps/#deaggint) which indicates that for the seismic bin of interest (high PGA, low likelihood events) the contributors to risk would be moderate magnitude earthquakes at nearby distances.

21. Section 3.3 (page 35, 2nd paragraph) describes other "ground motion response spectra of interest for this study." Clarify which response spectra were used in the structural

analysis described later in the report.

Response: The report has been revised to clarify this. In addition information from Section 4 will be brought to Section 3.3. The end of section 3.2 will include the following:

> These spectra are of interest for comparison purposes. The spectra in the NUREG-1150 study are also of interest because in-structure response spectra calculated for those ground motions were scaled (see Section 4), in approximation, to estimate in-structure response spectra for the input free-field ground motion used in this study.

22. Chapter 11, conclusion 5, footnote 43 gives the timeframe during which the fuel cannot be cooled by air. The Information Security Branch of NSIR should be consulted to confirm that this information is not security-related sensitive unclassified non-safeguards information, because the study is intended to be made publicly available.

    Response: The RES staff views the information as non-sensitive because it stems from the plant's response to a large seismic event (something an adversary cannot generate). Staff will confirm with NSIR and revise the report if necessary

23. Chapter 11, conclusion 6 seems to imply that the additional spent fuel pool instrumentation required by Order EA-12-051 is not effective for mitigating spent fuel pool accidents. Text should be added to this conclusion to explain its technical basis.

    Response: The report indicates that the required instrumentation is important to provide reliable indication to ensure that plant personnel can prioritize emergency actions. Further indication can affect which mitigation strategy is deployed as discussed in Chapter 2 of the report. Consideration of EA-12-051 was outside the scope of the study because it was not implemented by industry or verified by NRC at the time the plant was analyzed.

24. Chapter 11, conclusion 7 seems to imply that the additional mitigation capabilities required by Order EA-12-049 were not credited in the study. The additional mitigation capabilities required by Order EA-12-049 should be credited to improve the study's realism.

    Response: Consideration of EA-12-049 was outside the scope of the study because it was not implemented by industry or verified by NRC at the time the plant was analyzed.

25. Chapter 11, conclusion 16 states the study demonstrates that past spent fuel pool risk estimates from large seismic events are similar to this study for most consequence metrics. Text should be added to this conclusion to explain its technical basis.

    Response: Agreed and revised the conclusion to reference consequence comparison in Appendix B.

NSIR

26. Intro and Background Comments provided are repeated from the BC level review. Pg.8, Section 1.5, the report identifies that the majority of the risk from a seismic event is due to the inability of the operator to inject water into the pool for an extended period of time

(e g., days). However, this is based upon a research assumption and not a direct result of the seismic event. As such, a general comment that the research assumption of inability of mitigation efforts to commence for 48 hours is not based upon current Emergency Preparedness program capabilities which would assume that mitigation efforts commence significantly sooner rendering offsite release consequences moot. This acknowledgement of EP capabilities needs to be clearly stated early in the document and continuously throughout. If licensees presented onsite and offsite coordinated emergency response plans with the response assumptions used in this report, a reasonable assurance finding would definitely be in question.

Response: The assumptions in the study and the results of the study do not call into question a finding of reasonable assurance. Mitigation times for the study were chosen based on those assumed in SOARCA and informed by Fukushima. Section 5 3 has been revised to include a more detailed description of emergency measures in place in case of severe accidents. This section has also been revised to make clear that the truncation and assumed mitigation times were chosen by the team for purposes of the study. The report also makes clear that the initiating event chosen for analysis is well beyond design basis so a SFP failure resulting in offsite consequences is unlikely. The report also discusses the offsite response and challenges to implementing this response.

The report was clarified to explain that NRC analyzes low likelihood beyond design basis seismic events with and without mitigation to gain insights on the safety margin provided by NRC's regulatory framework. The HRA combined with reporting both mitigated and unmitigated results provides informative data to determine possible regulatory enhancements for consideration. The study corroborates the results of past studies. The study concludes that SFPs are robust and not expected to leak as a result of a seismic event, successful mitigation prevents most releases, no early fatalities are expected and individual LCF is low because effective protective actions limits individual exposure.

27. Major Assumptions Comments provided are repeated from the BC level review - Dispositioning of comment was not complete and needs to be completed as a Division Director comment. Major assumptions should include the fact that mitigation time is not indicative of the current EP environment.

    Response: See comment #26. Section 5.3 has been updated to include a more detailed description of emergency measures in place in case of severe accidents. This section has also been revised to make clear that the truncation and assumed mitigation times were chosen by the team for purposes of the study

28. Pg 60 Comments provided are repeated from the BC level review. Under "Liner Strains and Small Leakage Rates", 1st paragraph, "Maximum effective membrane liner strains from strain concentrations at the floor-walls junction are on the order of 0.037 (3.7 percent)."
    2nd paragraph,"On the basis of the reported failure criteria, this study assumed a somewhat conservative estimate for the liner failure strain from the point of view of leakage rate in order to characterize the leakage rate for a damage state with small leakage flow rate. Specifically, a liner strain at failure of 0.10 (10 percent) was assumed..." This comment was previously sent and the resolution was, "The study calculated the strains caused by the earthquake (demands). The reviewer is citing a sentence that refers to strain capacity." BC comment: clarity needs to be provided in

report as to the differences in the types of strains and the reasons/justification for the assumption which appears to be extremely conservative with respect to the design.

Response: To clarify the items raised in the comment, Section 4.4.1 is re-organized so that the part on Damage States and Relative Likelihoods will be at the beginning of section 4.4.1 (it was the last of three parts in this section). This is done to promptly inform the reader that the study treats both the induced strain (demand) and the limiting failure strains (capacity) as random variables. Although, median induced strains are less than median limiting failure strains, the uncertainty assessment shows that there is a small likelihood that the liner would tear.

The text in the second and third paragraphs of the part Liner Strains and Small Leakage Rates will be modified to read:

> An approach and failure criteria for steel liners used in reinforced concrete containments is used here to assess tearing of the SFP liner (Cherry 2001 and 1996). Failure criteria for liners without corrosion damage reported by Cherry (1996) are used in this study to estimate limiting failure strains for the stainless steel SFP liner. The approach estimates the crack width by multiplying the liner strain at failure by the width of the finite element with the maximum induced effective strain, which is approximately equal to 3.7 mm (0.15 in.) as indicated above.
>
> Since both the induced strains (demands) and failure strains (capacity) are treated as random variables, the strain at which the liner would tear, that is the condition at which the induced strain exceeds the limiting failure strain, is also random. An approach for a point estimate of that strain would be to calculate the most likely failure strain, which would be a strain greater than the estimated median induced strain (demand) of 0.37 but likely less than the median limiting failure strain (capacity) of about 0.10. Such an approach would involve a more detailed uncertainty analysis and probabilistic modeling than that used in this study, which does not seem justified given the approximations used as well as the uncertainties involved in the assessment of the flow rates through tears in the liner. This study assumed a failure strain of 0.10 (10 percent) for the liner strain at failure which is approximately equal to the assumed median failure strain

29. Pg 61 Comments provided are repeated from the BC level review. Under "Liner Strains and Small Leakage Rates", "Given the estimated width, length and depth for each localized liner tear and their number, it is still necessary to estimate the leakage rate through these tears. Estimation of this flow rate uses the following assumptions (1) the flow rate can be estimated using an equation similar to that used for flow through the concrete cracks and (2) the friction factor for that equation can be calculated on the basis of test results for leakage rates through cracks in pipes. These assumptions are not validated at this time. Therefore, considerable uncertainty exists for the resulting leakage rate estimate." This comment was previously provided and the response given was: "The assumptions referred to by the reviewer relate to the leakage rate given the estimated cracks in the liner. The initiation of cracks was calculated separately based on the strain demands and capacities." BC Comment: Response does not address comment as to why non-validated leakage rates were assumed. If the leakage rate has considerable uncertainty, the variability in the leakage rate should be stated and the assumed leakage rate needs to be justified as to why it was chosen given the

considerable uncertainty. More clarity needs to be provided on the basis for the assumed leakage rate.

Response: the paragraph is modified to read:

> Given the estimated width, length and depth for each localized liner tear and their number, it is still necessary to estimate the leakage rate through these tears. Estimation of this flow rate uses the following assumptions (1) the flow rate can be estimated using an equation similar to that used for flow through the concrete cracks and (2) the friction factor for that equation can be calculated on the basis of test results for leakage rates through cracks in pipes. These assumptions are not validated at this time. Therefore, considerable uncertainty exists for the resulting leakage rate estimate. The following paragraph addresses the process used to estimate the flow rate through these liner tears as well as sources of uncertainty for this estimation. These uncertainties may result in flow rate estimates that can vary by more than 100 %. It is noted that this damage state (small leakage rate) already is a result of binning the uncertain liner tearing into two discrete tearing conditions to cover a range of uncertainty for liner damage and associated flow rates. Assigning equal likelihood to the two highly distinct damage states acknowledges these uncertainties.

30. Pg 64 Comments provided are repeated from the BC level review. "Damage to the Reactor Building and Other Relevant SSCs" The response to the previously provided comment did not address why the HRA assumed containment failure when the SFPSS did not. The two studies should reflect the same assumptions such that mitigation efforts can be aligned between the studies. As it is, the two studies have significantly different mitigation efforts for different reasons. How can a determination be made as to how the two studies support one another with these differences? This is a fundamental question that needs to be answered/clarified within the report.

    Response: The containment in HRA is the primary containment that if failed in a reactor core damage event would make the refueling floor inaccessible for plant staff to inject or spray water into the SFP.

    The SFPSS assesses offsite consequences. It provides two bounding conditions: 10CFR50.54(hh)(2) mitigation is assumed to be successfully deployed or this mitigation is assumed to not be successfully deployed. The HRA estimates the probability of having successful mitigation for various plant damage states. These two pieces of information (i.e., consequence and probability) complement each other to inform SFP risk. The HRA provides scenario-specific likelihoods for each plant damage state (considering the state of the reactor, offsite power, etc.) The HRA combined with reporting both mitigated and unmitigated results provides informative data to gain insights on the safety margin provided by NRC's regulatory framework as well as possible regulatory enhancements for consideration.

31. Chapter 7 Comments provided are repeated from the BC level review. 1st paragraph, Doses are calculated at a great distance, e.g., 500 miles. Any health effects for small doses at such distance are speculative. As such, there is no value added to the report for this highly speculative result when considering its regulatory purposes. If not removed, then it is recommended that such health effects not be summed but rather segmented into appropriate categories and considered separately.

Response: Given the uncertainty of low doses on health effects, LCFs is being removed as a quantitative metric. See reply to comment #10 for more information. Land interdiction, displaced persons, and societal dose are reported to inform regulatory analysis under NRC's current regulatory framework. The consideration of distances beyond 50 miles is consistent with most previous research studies (See also the response to comment #43).

Individual LCF risk has been separated into appropriate categories and reported as a range based on dose truncation levels, the same as what was done in SOARCA. This SOARCA technique is preferred because it provides a range of results (that can be compared to the qualitative health objectives, for instance).

32. Pg 27 Comments provided are repeated from the BC level review. The original comment (below) as previously submitted with the disposition/response is provided. The "reviewer response" provides additional BC comment on the issue to be considered / dispositioned.

There is some confusion as to the statement that dose truncation has been implemented. The comment was not referencing the calculation of consequences with differing truncation models as has been done, but rather the summing of small doses to large numbers of people and reporting accumulated health effects while using he LNT model. At the least, the NCRP technique should be used. IT would be preferable to use the techniques of SOARCA and not report speculative dose and health effects beyond the area of regulatory interest to NRC, i.e., 50 miles. Additionally, the reporting of summed health effects, i.e., LCF is not as useful a metric as individual risk of LCF for risk communication purposes. LCF is often misinterpreted as absolute deaths, rather than an estimate of potential consequences given a conservative treatment.

Response: Given the uncertainty of low doses on health effects, LCFs is being removed as a quantitative metric. See reply to comment #10 for more information. Land interdiction, displaced persons, and societal dose are reported to inform regulatory analysis under NRC's current regulatory framework. The consideration of distances beyond 50 miles is consistent with most previous research studies (See also the response to comment #43).

Individual LCF risk has been separated into appropriate categories and reported as a range based on dose truncation levels, the same as what was done in SOARCA. This SOARCA technique is preferred because it provides a range of results (that can be compared to the qualitative health objectives, for instance).

33. Pg 150 Comments provided are repeated from the BC level review. Add an item 3 for why the latent cancer fatality risk is low because: 3. of the emergency preparedness response mitigation efforts.

Response: Section 7.2 has since been rewritten to make this point. In addition, the study concludes that SFPs are robust and not expected to leak as a result of a seismic event, successful mitigation prevents most releases, no early fatalities are expected and individual LCF is low because effective protective actions limits individual exposure.

34. Major assumption I don't agree with the assumption that offsite assistance will not arrive for 24 hours and that mitigative efforts with such equipment (e.g., fire truck) does not begin for 48 hours after the initiating event

Response: See response to comment #26. In Section 5.3, "At 24hrs" has been changed to "within 24hrs". Section 5.3 has been updated to include a more detailed description of emergency measures in place in case of severe accidents.

35. Chap 8 The HRA improved the study analysis but was unable to judge the effectiveness of offsite resources such as a fire truck. This limitation should be noted as a conservative limitation of the study.

Response: A table was added to provide an explicit list of scope and assumptions of the HRA study. Further, new text is being explored to clarify.

36. Conclusion 13 The frequencies noted appear to lack consideration of the HRA success probabilities that would, I believe, reduce the frequencies reported.

Response. The reliability of mitigation is not included as stated in Table 3 in Section 2. The conclusion will be expanded to include mitigation results. The HRA provides scenario-specific likelihoods for each plant damage state (considering the state of the reactor, offsite power, etc.) The HRA combined with reporting both mitigated and unmitigated results provides informative data to gain insights on the safety margin provided by NRC's regulatory framework as well as possible regulatory enhancements for consideration.

37. Section 8.1.2 the dose rate estimate is in error. The peak dose rate at the SFP rail is used whereas the spray would be located some distance back in a lower dose rate region. Additionally, the licensee has shielding on the floor to facilitate placement of the spray.

Response: Based on the oscillation monitors (or SFP spray nozzles) setup locations as indicated in the procedure TSG-4.1, the authors confirm that the dose rates stated in the report are correct. In addition, NRC staff walked down this strategy at PB in May 2012 with a Region 1 SRA as part of the B.5.b component of the triennial fire inspection with 2 of the individuals (Equipment Operators) assigned to carry out the strategy. At no time did they identify shielding that they anticipated using during deployment of the strategy. Additionally, the plant did not raise this as a result of their fact check of the HRA. Perhaps it is something that has been put in place since May 2012, but if so, it's newer than the snapshot of the plant that we set out to analyze. If the shielding can be confirmed and would have an impact on the results, a qualitative statement to that fact can be added to the report.

38. Section 8.1.2 the timing used in the HRA to denote when mitigation cannot be accomplished due to dose rate or steam environment, misjudges the ability of the ERO to perform the relatively simple task of attaching a fire hose to a spray in a challenging environment. For some analyses, one hour of additional time to mitigate would allow success.

Response: The high steam (or high temperature) becoming a limiting factor only occurs in small leak scenarios where the available time for response is greater than 13 hours.

Adding one or a few extra hours to the available time has little effects to HRA results. This is because in these situations time is not the dominant factor affecting human performance. Time is more important in moderate leak scenarios in which available time is 6 hours and 2.5 hours for refueling and non-refueling scenarios respectively. The radiation level is the limiting factor in these situations. Based on the SFP spray nozzles setup location indicated in TSG-4.1 the radiation level at the locations at that time is greater than 30 rem/hr. The time is firm in this criterion.

To set up the spray nozzles on the refueling floor in a moderate leak scenario where the leakage rate is greater than nozzle injection rate, based on procedure instruction the plant staff would first connect two fire hoses to two spray nozzles and inject water into the SFP, observing the change of the SFP water level (in this case the SFP water level continues lowering), attach a spray head to the spray nozzles each to change from injection mode to spray mode, ensuring the water spray into the SFP, and place a lead bag on top of the spray nozzle each to damp vibration for stable SFP spray. Completing these tasks requires some time. The 30 rem/hr is a reasonable threshold for the activities.

Furthermore the study assumptions are consistent with Appendix EE of EPRI TR-1025295 (2012) which is the technical basis for Severe Accident Management that the industry is relying on to update their Accident Management Programs.

39. Section 7.1.4 Please replace the second paragraph with the following: The staff modeled offsite response organization (ORO) decision making based upon the accident sequences, timing, radiological release, knowledge of response activities and the availability of emergency response technical support. Since actions beyond the EPZ would be taken ad hoc, there is no procedural guidance or exercise performance documentation upon which to base assumptions. However, state and local OROs have shown long standing capability and understanding of response to hypothetical radiological accidents. The accidents modeled in the SFPSS are slow to develop relative to the accident scenarios used in evaluated exercises. Additionally, there would be national level assistance to help civil authorities with protective action decision making. While alternative timing could be assumed the staff used a best estimate approach to modeling ORO decision making for protective actions beyond the EPZ.

    Response: Text has been added as requested.

40. Section 7.2 This section describes the use of dose truncation models in a manner that suggests they are a method to lower consequences rather than an alternative model. Dose truncation model use should be put in context as alternative and potentially valid health effects model

    Response: Dose truncation models provide two benefits, an alternative (and potentially valid) health effects model as well as a tool to better understand the contributions to LNT risk. Section 7.2 has since been reorganized and now is written to better represent the dose truncation models as potentially valid health effect models.

41. Fig 96 the title is confusing; is it meant to be "% of all individuals that are displaced"?

    Response: Section 7.2 has since been rewritten and the figure no longer exists.

42. General. My primary concern with this document is the fact that we are reporting significant results from a highly conservative and very low probability scenario that could be misinterpreted by the public. Accordingly, I believe that a section should be added to the document that discusses the results in the context of safety and adequate protection; i.e., do we still believe that there is adequate protection with the continued use of wet-storage and is there enough of a safety enhancement from a cost-benefit perspective to warrant moving more to the use of dry storage.

    Response: As stated in Section 1 of the report the U.S. Nuclear Regulatory Commission (NRC) has maintained that SFPs provide adequate protection of the public health and safety in either low-density or high-density storage configurations. This report does not call into question this finding. The study also does not make any determinations regarding whether there is enough of a safety enhancement from a cost-benefit perspective to warrant moving more to the use of dry storage. That is the role of NRR and the regulatory analysis. A paragraph has been added to explain the study's applicability to the Tier 3 activity and the NRC's current regulatory framework. The study corroborates the results of past studies. This study concludes that SFPs are robust and not expected to leak as a result of a seismic event, successful mitigation prevents most releases, no early fatalities are expected and individual LCF is low because effective protective actions limits individual exposure.

43. General The use of our models at great distance (i.e., up to 500 miles) becomes speculative and indicates a level of fidelity that likely exceeds their veracity. There are uncertainties in source term, dispersion modeling, and weather at distance and deposition at distance. The results are reported with excessive confidence. It would be more appropriate to provide estimates out to a distance that the analysis tools could more confidently calculate (e.g., 50 miles) and estimate qualitatively the potential impacts further away. A statement that the relocation could potentially extend to 500 miles in the worst case, would be more appropriate than reporting the results as the agency best estimate.

    Response: Though MACCS2 has been benchmarked against other Atmospheric Transport and Dispersion models up to 100 miles with favorable results, the authors acknowledge that uncertainty exists. In light of this, we have added the statement:

    > The accuracy of atmospheric transport and deposition models tend to decrease with distance, and therefore the results should be viewed with caution.

    In addition, the figures showing land contamination and displaced individuals at specific distances have been replaced with tables that more generally report these consequences at 0-50, 0-100, and 0-500 miles, which is largely consistent with most past research studies.

44. Section 7.3.2 DD Comment: I am providing this comment to give the answer to the "disposition" question. Please reconsider original comment with this additional information:

    after reading this I cannot determine whether contaminated food is included in consequence data or not... it should not be, no one is going to eat contaminated food in the US after this accident.

The basis for stating that no contaminated food will be consumed simply comes from the knowledge of public and civil authority reaction to actual and hypothetical radiological incidents. In repeated exercises public officials have decided to condemn a regional crop rather than parse contamination levels. Public reaction to contaminated food would also be extreme and anything even remotely associated with the contaminated area would be eschewed. There is no technical document establishing this outcome, it is just the nature of current society as alternative food sources would be widely available. It cannot be said the "no contaminated food would be consumed"" as very low levels of radioactivity currently exist in food currently, but the point is that no significant amount of contaminated food would be consumed. Pursuit of dose consequences through this exposure pathway seems inappropriate.

Response: Latent cancer fatalities are no longer being reported, and MACCS2 does not treat this pathway in individual LCF risk, and therefore the report no longer reports any type of LCF metric from ingestion.

RES/DE

This report provides the methodology and results of a limited-scope consequence study to update the best-estimate consequences expected from the application of a postulated beyond-design-basis earthquake (with an estimated frequency of occurrence of one event in 61,000 years) to a selected U.S. Mark I boiling-water reactor spent fuel pool. The primary objective of the study is to provide updated and publicly available consequence estimates of a representative, postulated spent fuel pool severe accident under high-density and low-density loading conditions. These estimates can then inform ongoing discussions as to whether action should be taken to require operators of U.S. nuclear power plants to expedite movement of fuel from the spent fuel pool to onsite, dry cask storage.

I would delete the last sentence and replace it with this:

These estimates can be used to confirm that the current industry strategy favoring high density fuel storage in spent fuel pools remains adequately safe and whether a change in strategy towards low density fuel storage in spent fuel pools might represent a significant safety improvement.

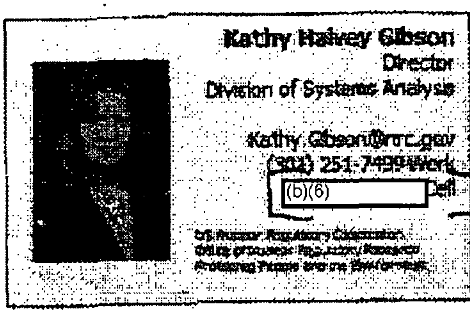
Response: We did not change the wording as suggested, but we did revise the wording to say "The study will be used to inform regulatory decision-making regarding whether expedited transfer of spent fuel from spent fuel pools to casks is justified." Additionally, a paragraph has been added to the report to describe the study's relationship to the Tier 3 activities and how the study will be used in the current regulatory process.

RES

Ader, Charles

From: Gibson, Kathy
Sent: Wednesday, November 09, 2011 3:14 PM
To: Correia, Richard; Coe, Doug; Case, Michael; Richards, Stuart
Cc: Scott, Michael; Lee, Richard; Sheron, Brian; Holian, Brian; Ader, Charles; Ruland, William
Subject: RE: Status of Spent Fuel Pool Scoping Study Interim Deliverables and Location in ADAMS
Attachments: Kathy Halvey Gibson.vcf

For you information, status and location of interim deliverables in ADAMS outlined by the PM below.



Ex 6

From: Lee, Richard
Sent: Wednesday, November 09, 2011 1:43 PM
To: Gibson, Kathy
Cc: Scott, Michael
Subject: FW: Status of Spent Fuel Pool Scoping Study Interim Deliverables and Location in ADAMS

Kathy:

Please inform other RES Division Directors on the status of this activity as you see fits.

Thanks,
Richard

From: Wagner, Katie
Sent: Tuesday, November 08, 2011 4:14 PM
To: Lee, Richard
Cc: Esmaili, Hossein; Helton, Donald; Nosek, Andrew; Murphy, Andrew; Pires, Jose
Subject: Status of Spent Fuel Pool Scoping Study Interim Deliverables and Location in ADAMS

Releasable

FSME

Good Afternoon Richard,

Non-Pub

As follows is the status of the interim deliverables due at the end of October 2011 (which may be viewed in ADAMS Package ML113110789) as part of the Spent Fuel Pool Scoping Study:

- A draft report documenting seismic and structural assessment activities to date as well as a draft report on seismic initiators provided by DE staff. Work toward developing the initial damage states necessary for initiation of the accident progression analysis is still ongoing.

- Some information has been obtained from a June 2011 DOE Science Council briefing about onsite radiation fields at Fukushima and a contract at ORNL has been initiated (a kick-off meeting was held on Oct. 31) to obtain a report regarding the specific potential radiation fields at the Peach Bottom site. This task is being done by DSA/FSCB staff.

A5-8 E/1

- A draft interim report discussing probabilistic considerations has been completed by DRA/PRAB staff.

- A draft report describing MELCOR SFP modeling and capabilities familiarization activities has been completed by DSA/FSCB staff.

- A draft interim report discussing offsite release and consequence modeling considerations has been completed by DSA/AAB. This includes discussion on the scope, modeling decisions, and input data for EP, dose, and economic parameters, as well as other needed emergency and long-term phase input parameters. Preliminary discussions with NSIR on this project have taken place, and more detailed discussions on emergency preparedness and response are planned for November.

- A draft communication plan has been completed by DSA/FSCB staff.

The above reports fulfill most of the requirements laid out for Phase 2 in the July 2011 SFPSS plan. Work in November will focus on:

- completion of the ongoing preliminary structural assessments and emergency preparedness scoping;
- conduct of the ORNL task on accessibility related to radiation fields
- continued MELCOR model familiarization and development
- event tree model development
- supporting routine briefings for other office staff and RES management
- supporting emergent briefings (e.g., the NRR LT)

Thank you,

Katie Wagner
General Engineer
U.S. Nuclear Regulatory Commission
(301) 251.7917
Katie.Wagner@nrc.gov

Package Name: Status of Interim Deliverables for the Spent Fuel Pool Scoping Study (SFPSS)

Accession Number: ML113110789

Package Properties

Paste into Package | Unfile | View | Profile | Open | Checkout | Refresh | Close

	Name	Accession Number	Official Record?	Availability	Document Date	ADAMS Date Added
OK to release	Status of Spent Fuel Pool Scoping Study Interim Deliverables - E-Mail	ML113120404	No	Non-Publicly Available		Nov 8, 2011 2:24 PM
	Structural Assessments Input for the Spent Fuel Pool Scoping Study	ML113110742	No	Non-Publicly Available		Nov 7, 2011 2:15 PM
	Offsite Release and Consequence Modeling Considerations Related to the Spent Fuel Pool Scoping Study	ML113110807	No	Non-Publicly Available		Nov 7, 2011 2:21 PM
OK to release	Draft Communication Plan for the Spent Fuel Pool Scoping Study - October 2011	ML113110857	No	Non-Publicly Available		Nov 7, 2011 2:26 PM
	Probabilistic Considerations Related to the Spent Fuel Pool Scoping Study - Draft Report	ML113110913	No	Non-Publicly Available		Nov 7, 2011 2:31 PM
	Seismic Initiators Input for the Spent Fuel Pool Scoping Study	ML113110929	No	Non-Publicly Available		Nov 7, 2011 2:34 PM
	MELCOR Accident Progression Modeling Approach for the Spent Fuel Pool Scoping Study - Draft Report 11/2/2011	ML113120395	No	Non-Publicly Available		Nov 8, 2011 2:18 PM

There are 7 Documents in this Package

Page 1 of 1

Reorder

Close

AS-9

Powell, Eric

RES

From: Helton, Donald
Sent: Wednesday, March 07, 2012 7:49 AM
To: Ader, Charles
Cc: Powell, Eric; Mrowca, Lynn; Wagner, Katie
Subject: ACRS Slides on SFPSS - Official Use Only
Attachments: ACRS_Subcomm _Mar6_FINAL_ver2_05Mar2012.pptx

Charlie:

Per your request when we talked in the hallway yesterday afternoon, the slides we used yesterday to brief the ACRS on the RES SFP Scoping Study are attached. Note that these slides are Official Use Only (it was a closed briefing). We will be presenting on the project (methods/models only) at both the RIC and an April ACRS full committee meeting.

If you want anything else, don't hesitate to ask.

Don

Don Helton
Division of Risk Analysis
NRC Office of Nuclear Regulatory Research
Physical address: 21 Church Street, CSB4-C9, Rockville, MD 20850
Postal address: US NRC / MS CSB4-C7M / Washington, DC 20555
Ph: 301 251-7594

A5-10

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Spent Fuel Pool (SFP) Scoping Study

Katie Wagner
Office of Nuclear Regulatory Research

Briefing for ACRS
March 6, 2012

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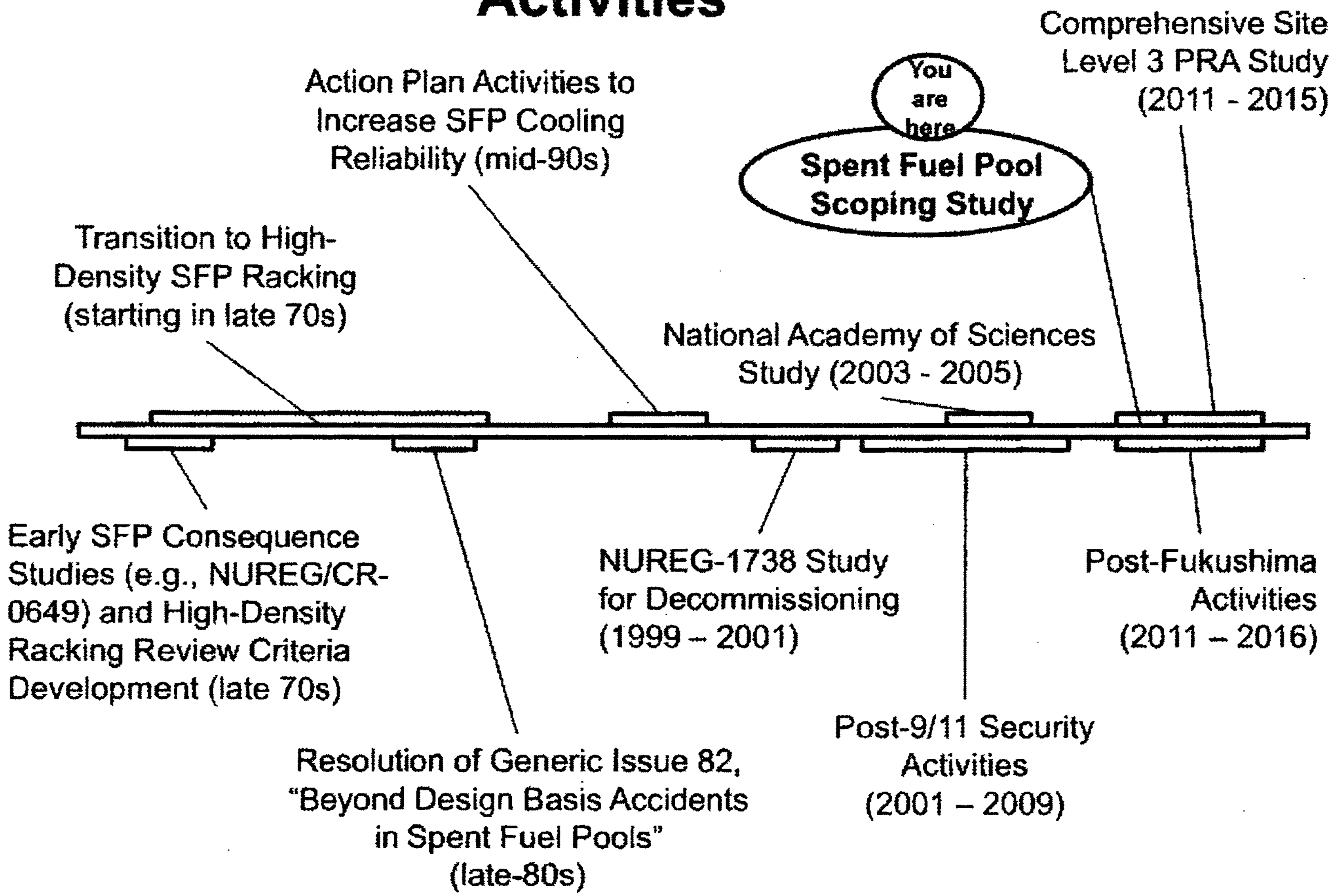
Timeline of Major SFP-related Activities



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Background on SFP Risk

- SFP risk is low, due to the low frequency of events that could damage the thick reinforced pool walls
  - Frequency of fuel uncovery; 6E-7 to 2E-6/yr – NUREG-1738
  - Consequences have been assessed to be large due to the potential for heat-up of all the fuel in the pool
  - Heat-up of the fuel in the pool can lead to "zirconium fire" initiation and propagation
  - Large inventory of Cs-137
- The above has prompted questions as to whether older fuel should be moved to casks

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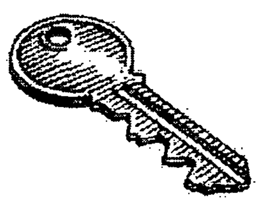
Motivation for SFP Seismic Study

SFP Risk involves:

- SFP Seismic Hazards
- Dry Cask Storage Risk *(e.g., NUREG-1864)*
- Cask Drop Hazards for SFPs *(e.g., NUREG-1738)*
- Repackaging For Transportation
- Fuel Storage Infrastructure *(e.g., 2010 EPRI study)*
- Worker Dose *(e.g., 2010 EPRI study)*
- Emergency Preparedness *(e.g., NUREG-1738)*
- Part 50, 72 & 73 Regulatory Requirements
- Multi-Unit Risk *(e.g., SECY-11-0089 project)*
- Design/Operation Differences Between Sites
- Boraflex Degradation & Inadvertent Criticality
- Protection Against Malevolent Acts *(e.g., post-9/11 security assessments)*
- Other SFP Hazards *(e.g., NUREG-1353)*
- Actions in Response to Japan Events *(e.g., Near-Term Task Force Recommendation 7)*



SFP
Seismic
Hazard

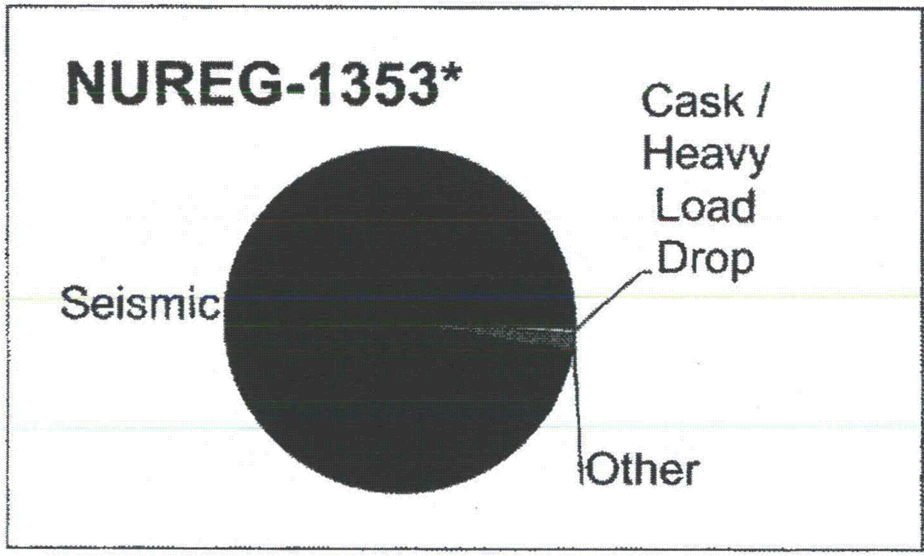
Past studies have indicated that SFP seismic hazard is an important piece of overall spent fuel risk.

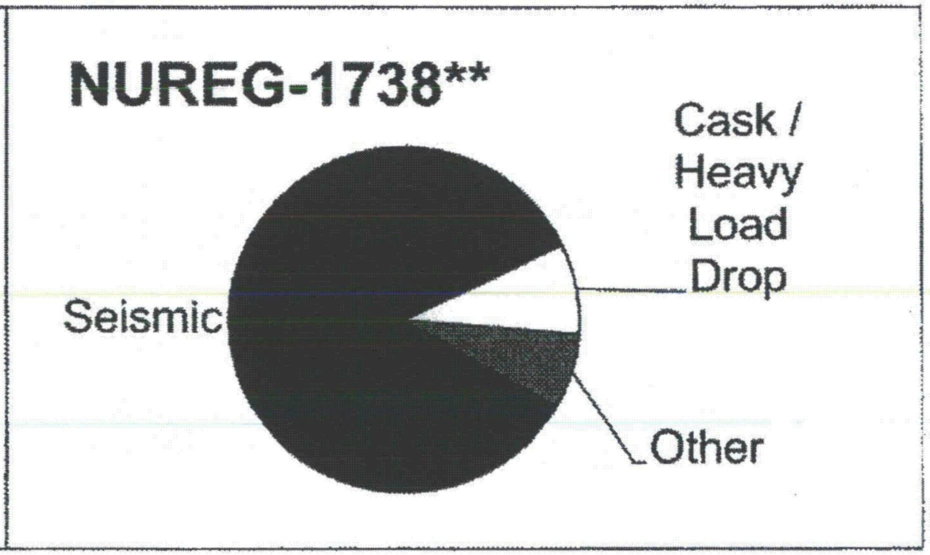
For this reason, SFP seismic hazard is the logical place to start in probing the continued applicability of past studies and developing insights for the current spent fuel storage situation.

Depending on the results gained from the study, additional work might be necessary to obtain a more holistic answer.

Motivation for Seismic Study

Annual frequency of SFP fuel uncovery as reported in previous SFP risk studies





*BWR, best estimate results

**Based on Livermore hazard curves which generally more closely match the updated USGS curves for the studied plant

Past SFP risk studies indicate that seismic hazard is the most prominent contributor to SFP fuel uncovery. While these studies have known limitations, this is sufficient motivation to focus on this class of hazards in the SFPSS.

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Overview of Spent Fuel Pool Scoping Study (SFPSS)

- Focus: reexamination of the potential advantages associated with moving older fuel stored in the SFP to dry cask storage in an expedited manner
- Emphasis is given to acquiring timely results for ongoing deliberations and external stakeholder interest. The project is using:
  - Available information / methods
  - A representative operating cycle for a BWR Mark I (Peach Bottom)
  - Past studies to narrow scope
- Plan finalized: July 2011 (ML111570370)
- Study to be completed by: June 2012

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Summary of Unique Aspects of the SFP Scoping Study

- Specific examination of the pool loading configuration (high-density vs. low-density) for **contemporary** SFP loading and requirements
- **Updated** SFP beyond design basis accident (BDBA) consequence estimates (first for operating reactor since 1989)
- **Site-specific** (and thus more realistic) SFP BDBA consequence estimates (not a focus of previous studies)
- First public capturing of SFP BDBA consequences considering **50.54(hh)(2)** equipment
- First capturing of SFP BDBA **seismic /structural** response since late 1980s
- First publicly available NRC study utilizing **MELCOR** (our tool of choice) for SFP analysis

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Technical Approach

- Two conditions to be considered:
  - Representative of the current situation for the selected site (i.e., high-density loading and a relatively full SFP)
  - Representative of expedited movement of older fuel to a dry cask storage facility (i.e., low-density loading)
- Elements of the study include
  - Seismic and structural assessments based on available information to define initial and boundary conditions
  - SCALE analysis of reactor building dose rates
  - MELCOR accident progression analysis (effectiveness of mitigation, fission product release, etc.)
  - Emergency planning assessment
  - MACCS2 offsite consequence analysis (land contamination and health effects)
  - Probabilistic considerations

Seismic and Structural Methods and Preliminary Results

Jose Pires

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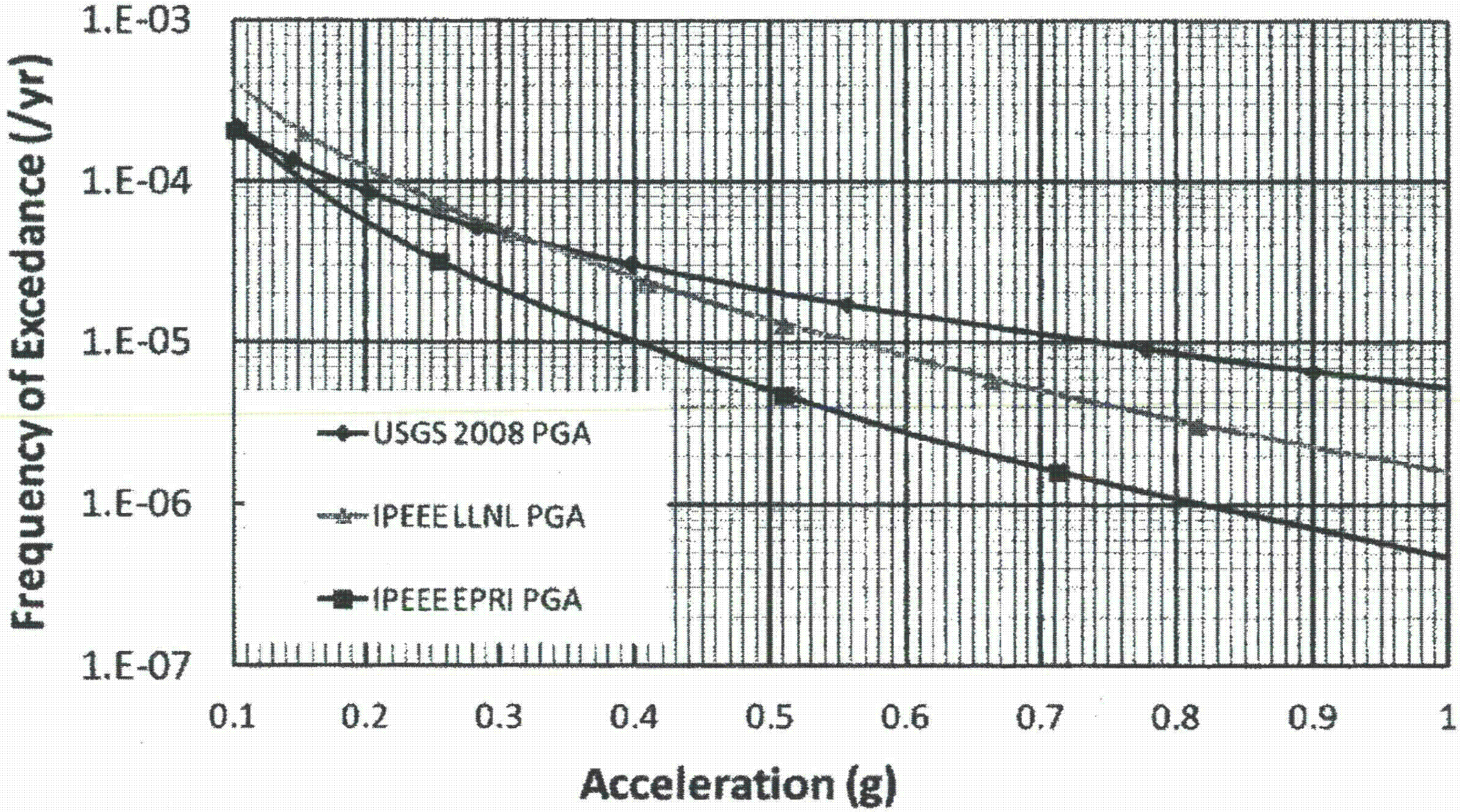
Prescribed Seismic Scenario

- Seismic event: 0.5 g to 1.0 g peak ground acceleration (PGA)
  - Challenging but very low frequency of occurrence (one event in 61,000 years)
    - PGA 6 times greater than that for the SSE and beyond the seismic design basis for Eastern US plants
  - USGS hazard models (2008) being used as starting model for ground motion response spectra
- Review of past studies indicates that less severe events would not challenge the SFP

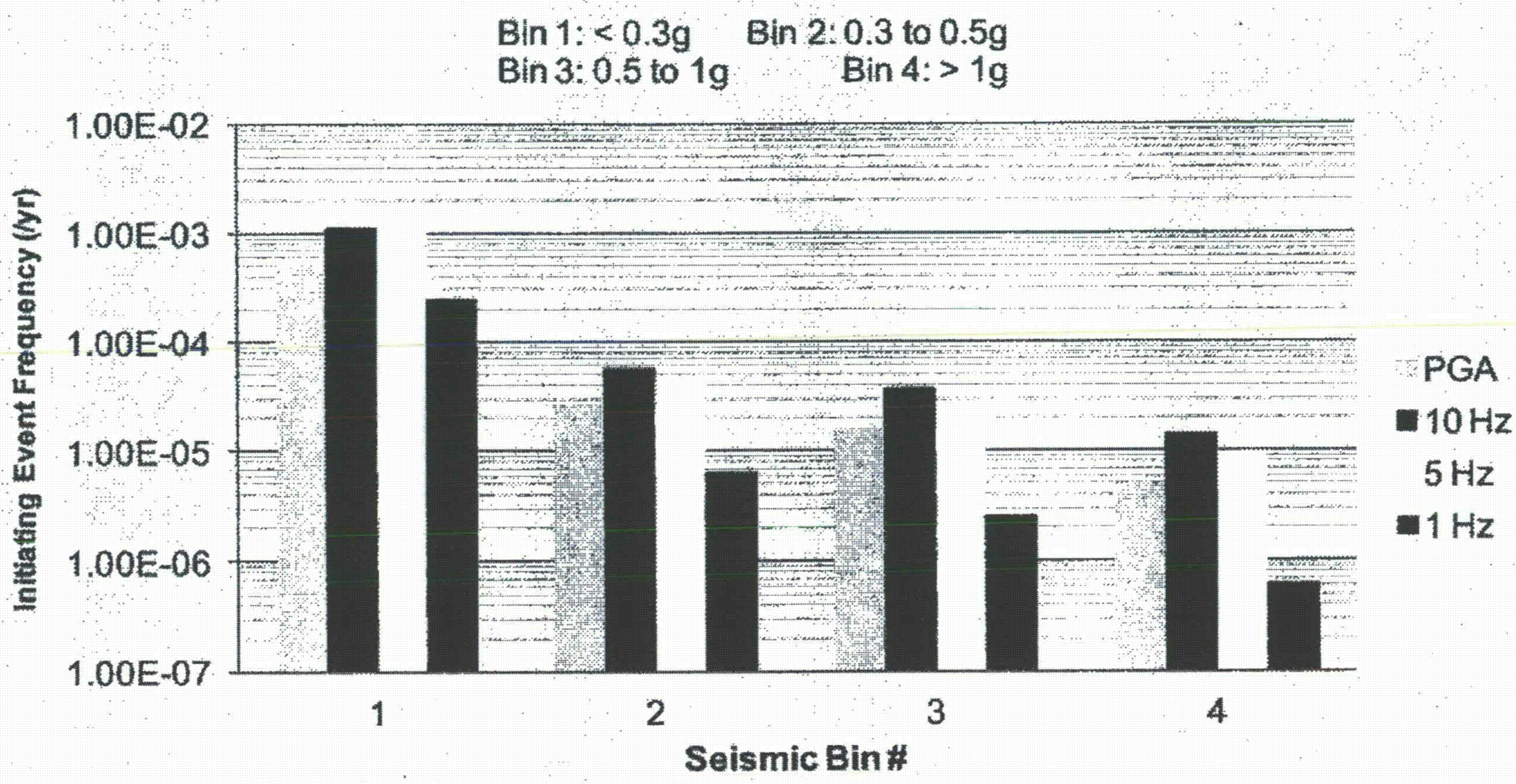
Site Seismic Hazard

PGA exceedance probabilities from various models



Initiating Event (IE) Frequency

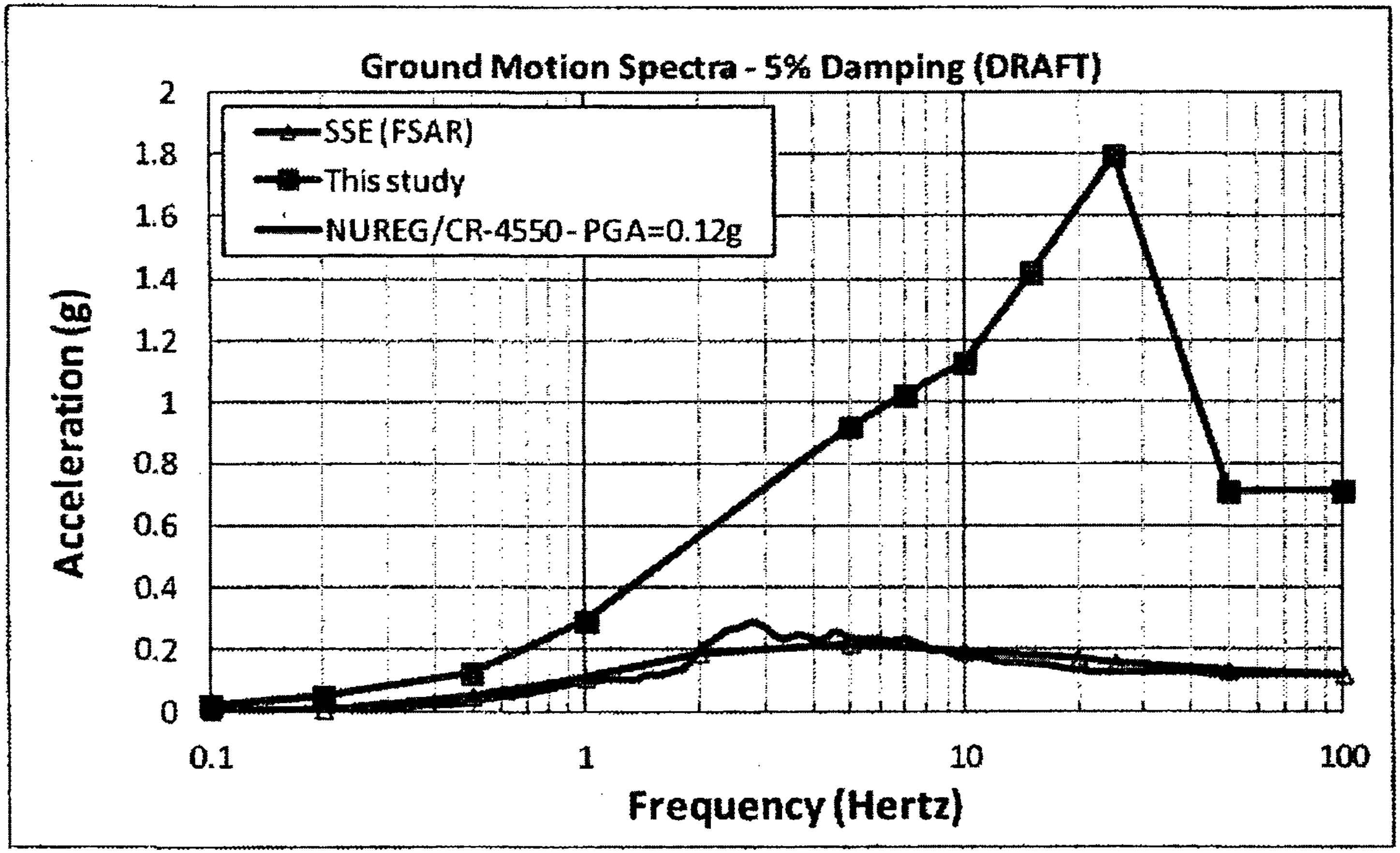
Seismic IE Frequency Comparison
(Based on USGS 2008)



Seismic Input

- Objective: to provide initial seismic assessment
  - Site Ground Motion Response Spectrum (GMRS)
- Rock site
  - OBE is 0.05g
  - SSE is 0.12g
- Challenging damage is not expected below seismic hazard Bin 3 – 0.5g to 1.0g
  - Level of 0.71g used
- USGS Hazard Assessments (2008) used as starting hazard level to obtain GMRS shape (Similar to GI-199 Resolution)
  - Site GMRS scaled to obtain GMRS for Bin 3
- Site GMRS rich in high frequencies (5 to 25 Hz)

Seismic Input



Comparison of ground motion spectra: this study, SSE and spectrum for the NUREG-1150 PRA (scaled to the SSE PGA)

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Structural Input

- Objective: to determine starting point for subsequent accident progression analysis
- Approach:
  - Generally follows approach used for GI-82 (NUREG/CR-5176)
    - Enhanced to address specific study aspects (Finite Element Modeling)
  - Uses in-structure response spectra (accelerations) calculated for the NUREG-1150 study (NUREG/CR-4550, Vol. 4, Part 3)
    - Scaled for increased PGA (from 3xSSE to about 6xSSE)
    - Scaled to account for high frequency content in the site GMRS
  - Uses 3D nonlinear finite element analysis of the SFP structure and its supports (subjected to equivalent static loads) to calculate:
    - displacements, concrete and reinforcement strains and stresses, structural distortion, and liner strains

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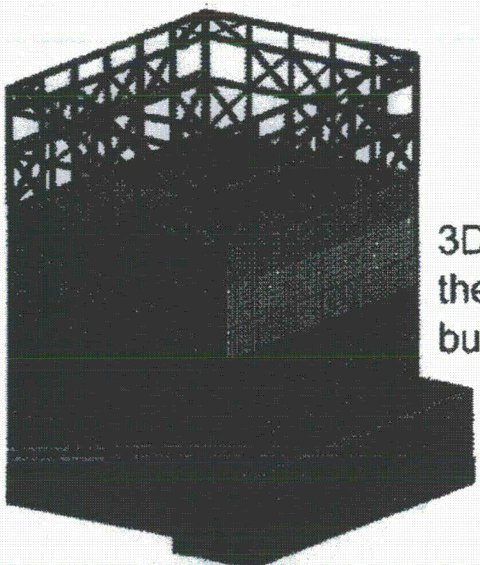
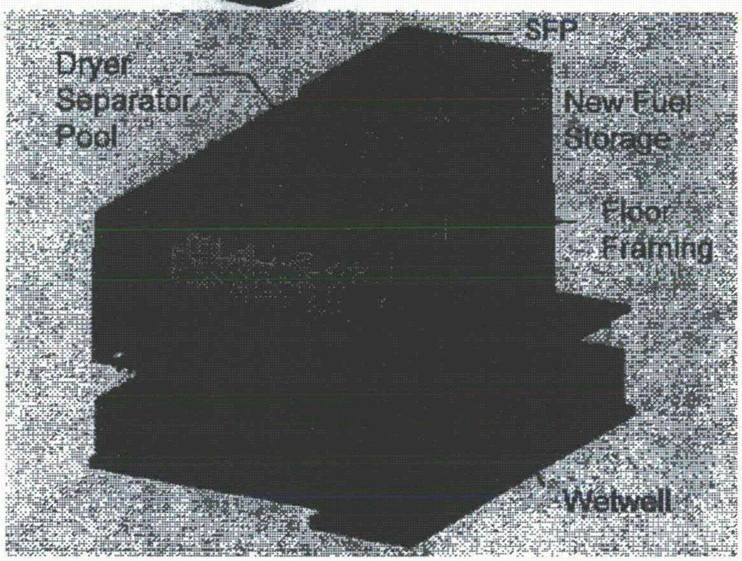
Structural Input

- Analysis of the SFP structure subjected to equivalent static seismic loads induced by the site GMRS provides input for the estimation of the damage states for the SFP
- Loads considered
  - Dead loads -- weight of the pool structure (mostly concrete), spent fuel racks and assemblies, and water
  - Seismic loads -- body forces from the accelerations of the pool structure, hydrodynamic impulsive pressures and reaction forces from spent fuel racks
  - Combined vertical and horizontal ground shaking
- Finite element programs used:
  - ANSYS to estimate and verify seismic loads (especially the hydrodynamic impulsive loads and sloshing amplitudes)
  - LSDYNA for the nonlinear analyses

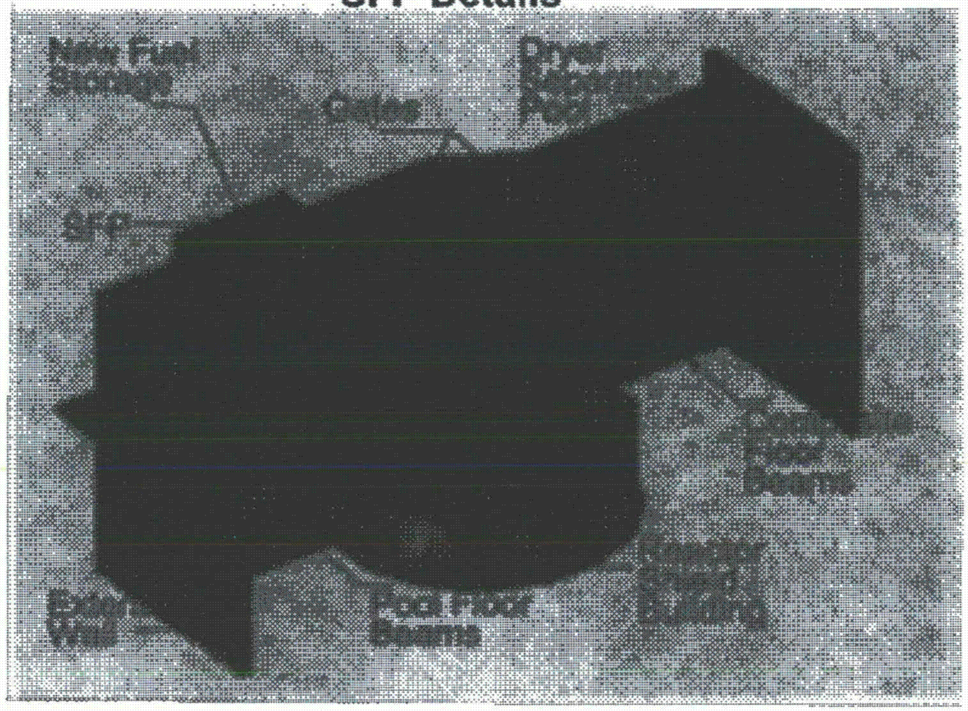
Reactor Building and SFP



3D model of the reactor building



SFP Details



Used to generate 3D finite element models of the SFP structure and its supports

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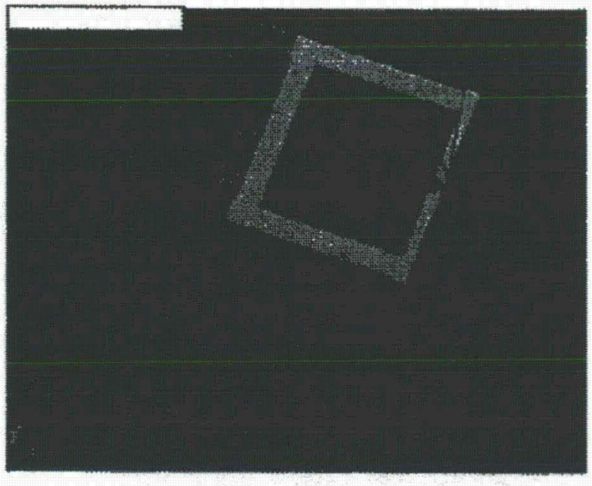
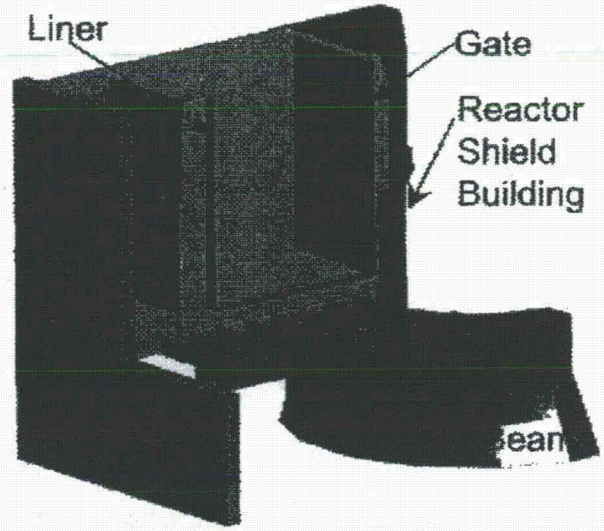
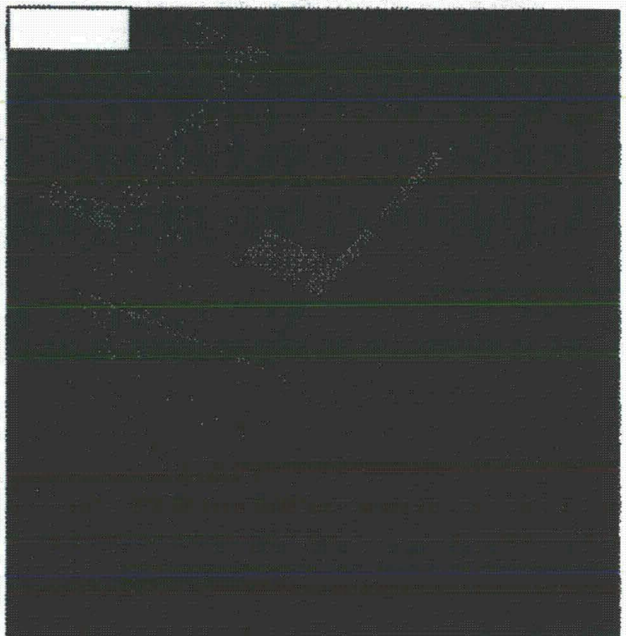
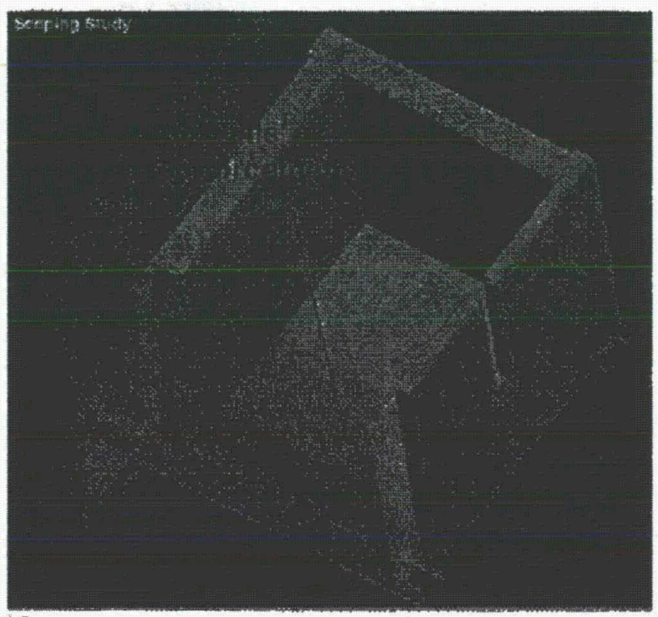
Finite Element Model of SFP





All these components included in the model





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Structural Input

- Approach (cont'd)
  - Simpler approaches to assess damage to:
    - Penetrations, support systems, other SSCs necessary for accident mitigation (e.g., building housing a portable diesel pump), other structures
- Approximations / assumptions
  - Effects of ground motion incoherency on high frequency components of floor spectra was not considered (possible conservatism)
  - Floor spectra do not account for coupling of SFP components to building (possible conservatism)
    - Hydrodynamic pressures based on scaled floor response spectra
    - Dynamic time-history analyses of the whole reactor building including the SFP were not done at this stage
  - Seismic loads from spent fuel racks and assemblies approximated
    - May need adjustment based on the analysis reports from the licensee submitted at the time of the license amendment for high density racking

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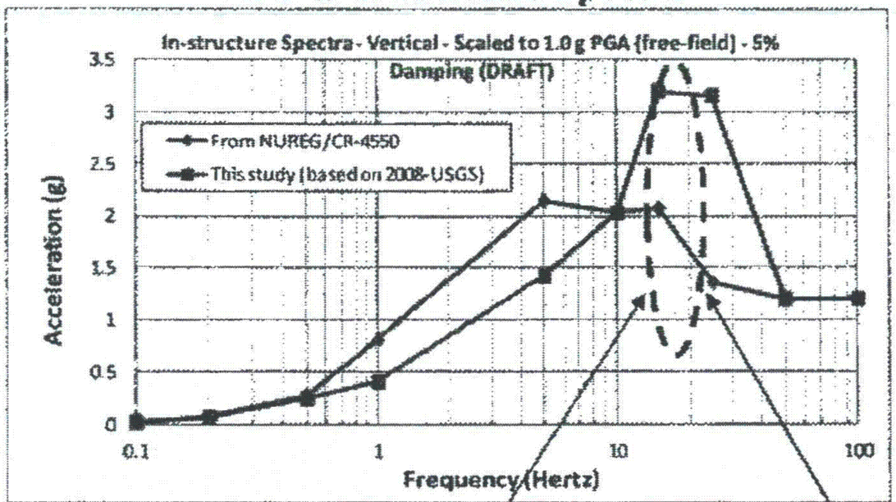
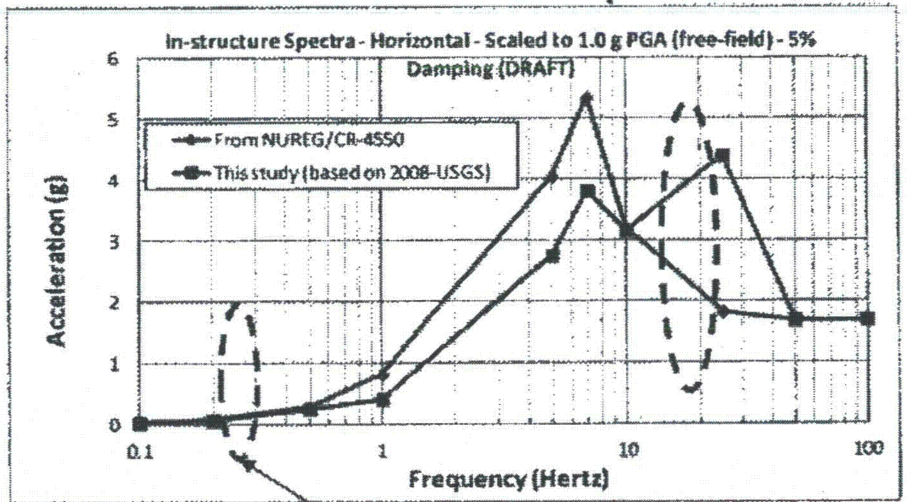
SFP Structural Analysis

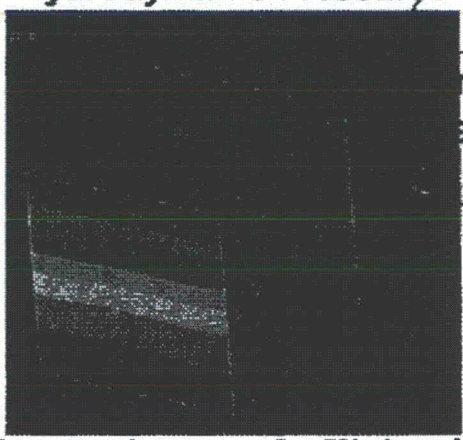
Vertical In-structure Spectra



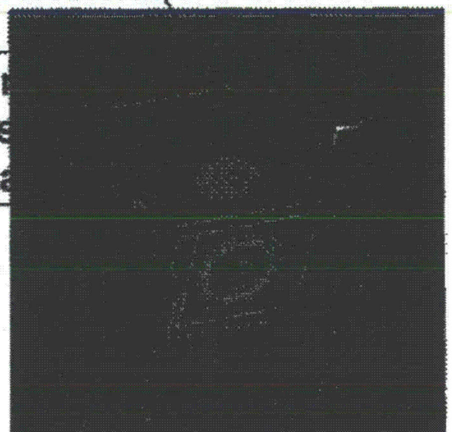
Horizontal In-structure Spectra



Hydrodynamic Pressures

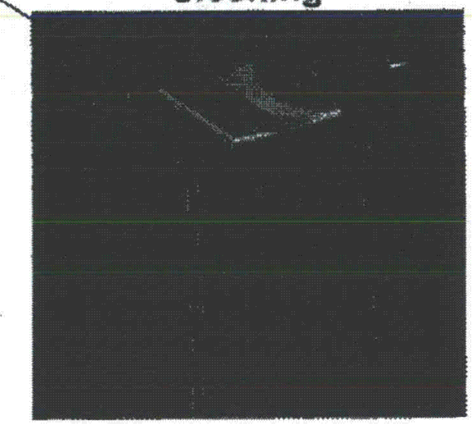


Accelerations in SFP Floor



results a
a in this s
prelimina

Sloshing



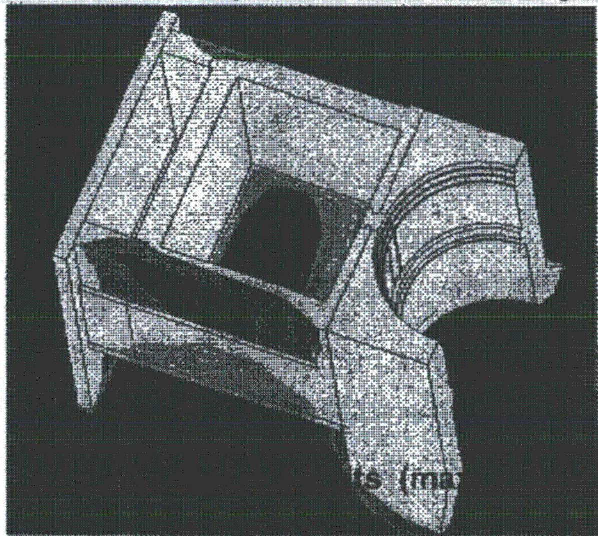
- Spectra shown are for 5% damping and 1.0g PGA (event PGA is 0.71 g)
- Analysis used 10% damping for the reactor building and 5% for the SFP. For sloshing lower damping was used.
- Analysis conducted with ANSYS – Hydrodynamic pressures used to verify inputs used in the finite element analysis

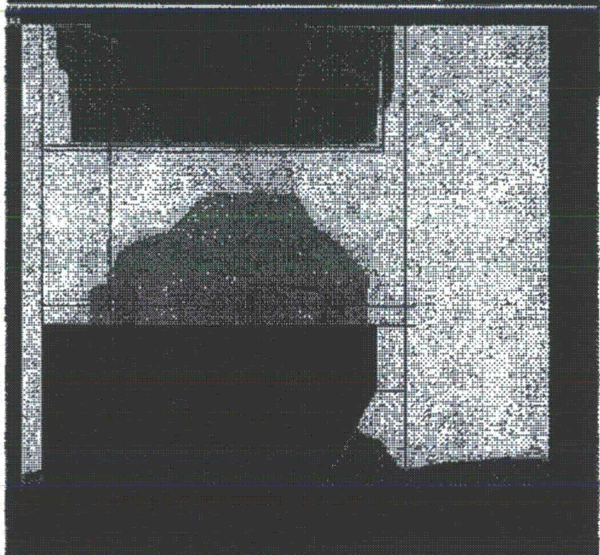
Preliminary Results

- Concrete cracking – Results indicate a through-the-wall concrete crack at the junction of the spent fuel pool walls to the pool floor
  - Crack thickness varies along this junction
- Liner strains – Maximum liner strains at the junction of the walls and floor
  - Analysis for the base finite element model (element size ~ 6 to 8 inches) shows strains in the range
    - 1.5E-4 to 1.9E-3 (0.0019)
    - Yield strain: about 1.2E-3 (0.0012)
  - Detailed liner analysis done to assess strain concentrations

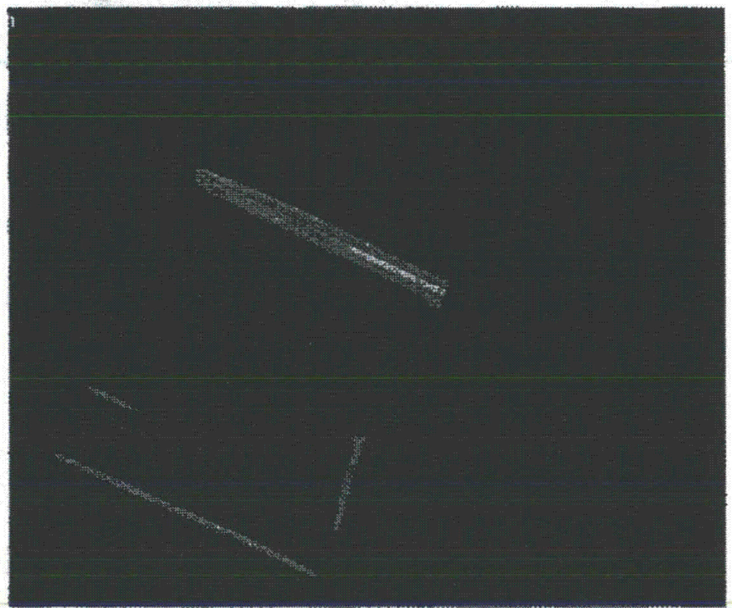
Preliminary Results

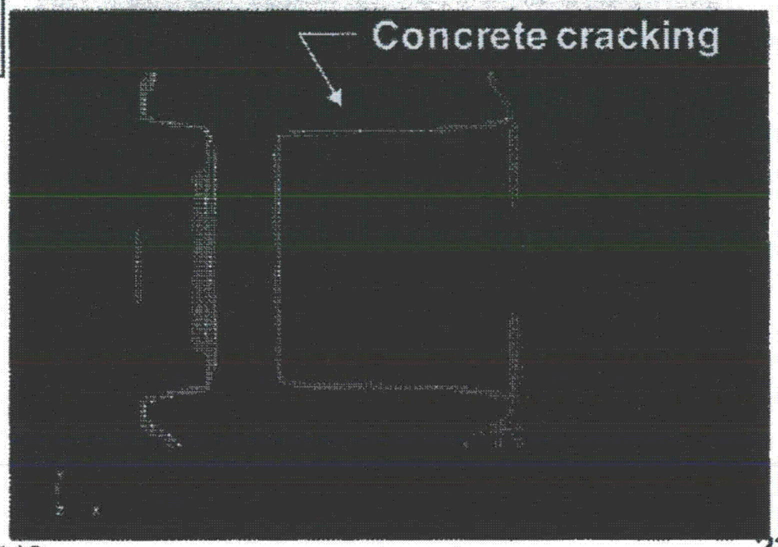
Load Case: (100%V + 40%H)





Preliminary results for the base model and analysis





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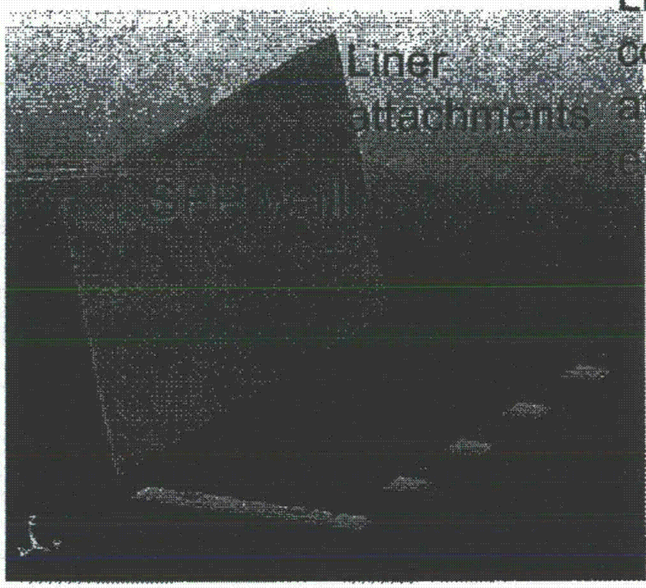
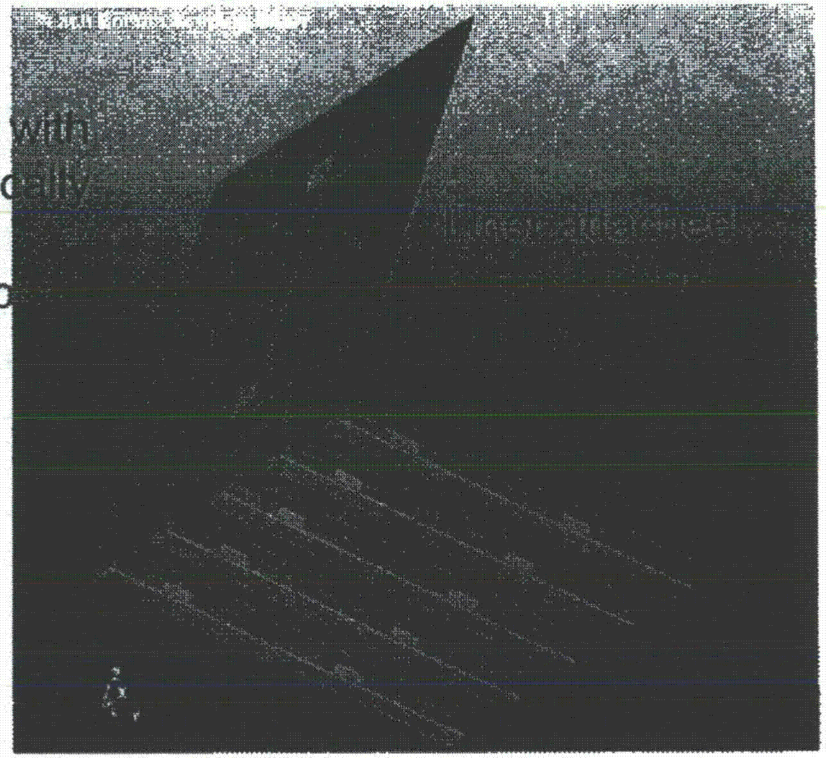
Preliminary Results

- Liner strain concentrations
  - Size of liner elements in the base model ~ 8 inches (~20 cm)
    - Base model does fully capture liner strain concentrations at the junction
  - Constructed detailed model of liner and liner attachment details with elements as small as ~ 0.15 inches (~ 0.37 cm)
  - Embedded this model in the base model to get strain concentrations

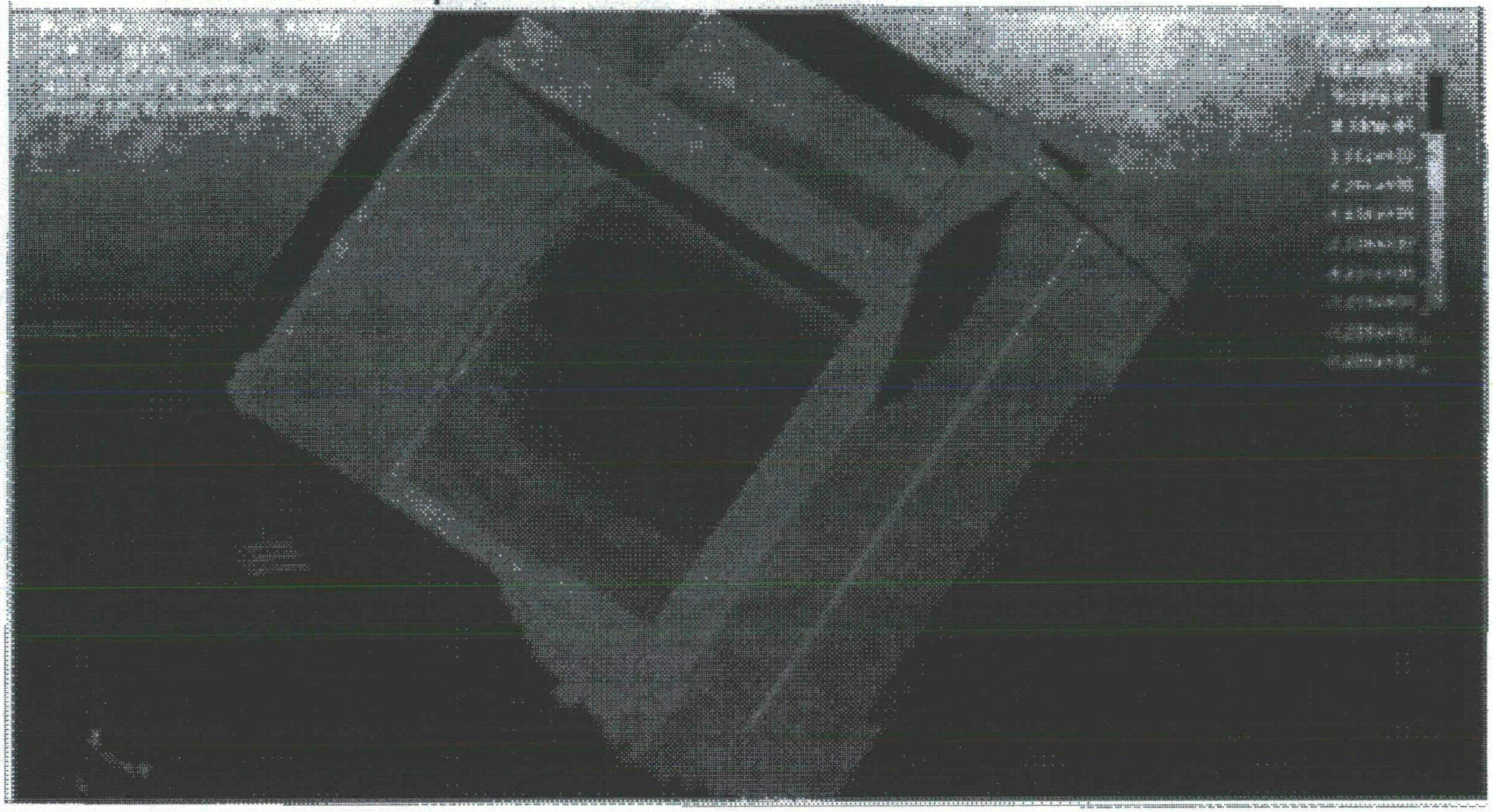


Liner in contact with concrete and locally attached to embedded shap



Preliminary Results

- Assessment of liner strain concentrations
  - Calculated displacements with detailed liner insert



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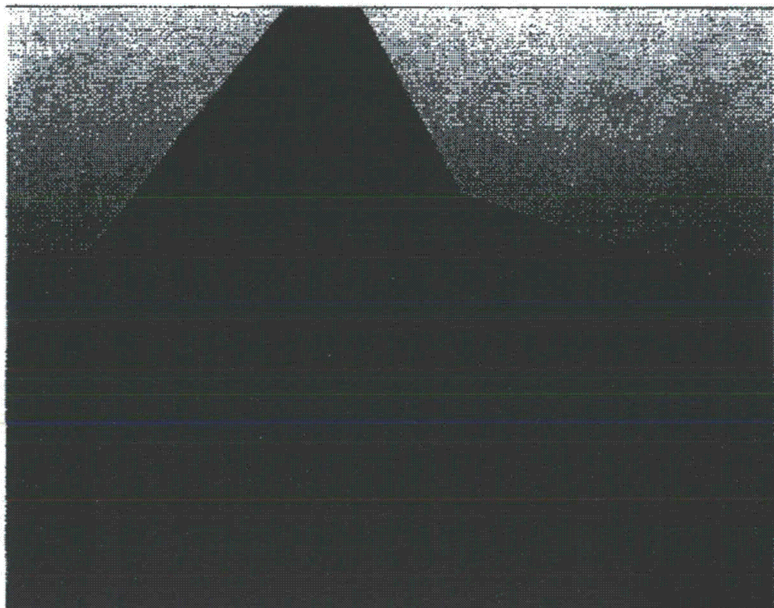
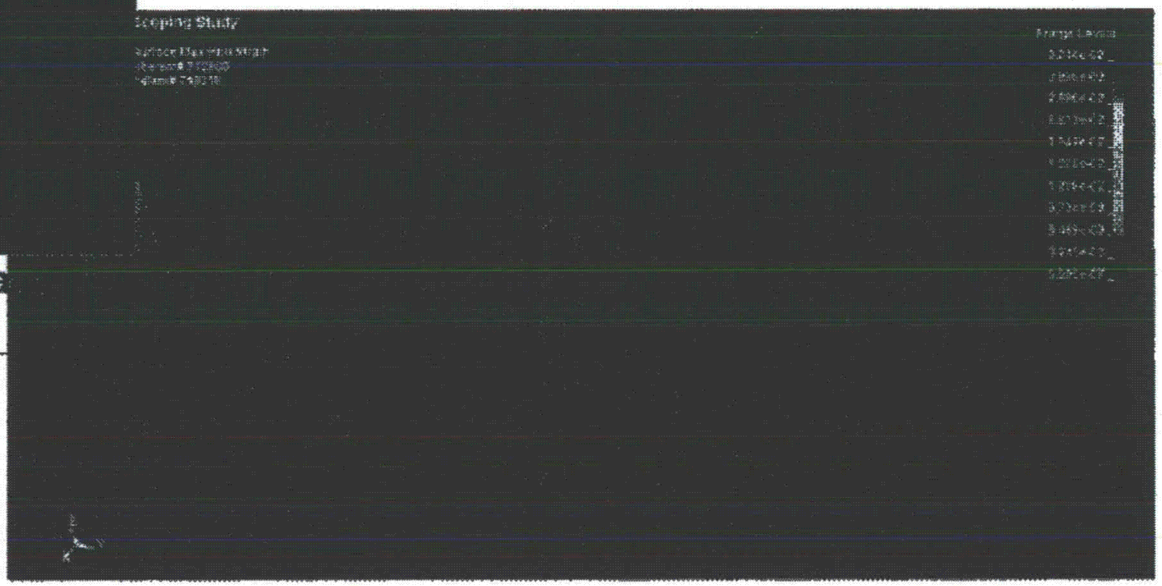
Preliminary Results

- Assessment of liner strain concentrations
  - Contours of maximum principal strains (mid shell surface)



- Strains concentrate at the backup plates with maximum less than the expected failure strain
- Strains below yield away from the backup plates (liner is not attached in these regions)

Prelimina results



Preliminary Results

Damage States (SFP structure)

- Most likely case – no liner tearing and so no water leaking except from sloshing
- Small relative likelihood of:
  - Extensive liner tearing with drainage controlled by size of concrete cracking
  - Liner tearing localized at the backup plates

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Preliminary Results

- Penetrations, other structures and equipment
  - Penetrations
    - The analysis shows that the pool is stiff and the distortions small
    - Damage to penetrations is not expected
  - Reactor building
    - Fragility estimates for the NUREG-1150 study (in NUREG/CR-4550) indicate a median fragility of about 1.6 g
    - Natural frequencies of the reactor building are not in the high frequency region of the GMRS used for the site (except for the vertical mode which does not control the building fragility)
    - On these bases, it is concluded that the reactor building would survive this event
  - Building structure above the spent fuel pool and overhead crane
    - Building structure above the pool is not vulnerable to earthquakes (it has low mass and was designed to carry heavy crane loads of 125 tons)
    - Overhead crane, even if were to collapse, would not fall inside the pool
    - On these bases, damage to the pool is not expected from this part of the building from the ground shaking

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Preliminary Results

- Penetrations, other structures and equipment
  - Building housing B5B equipment
    - Water treatment building, which is not a seismic Category I structure
    - Fragility of older buildings in the Central and Eastern United States (CEUS) not especially detailed to resist earthquakes is expected to be less than that of Seismic Category I buildings
    - Estimated median fragility for the turbine building (NUREG 1150 PRA) is about 0.5 g (not Category I)
    - Building collapse would not necessarily prevent access to equipment
    - Absent other information suggest considering equal likelihood of access and non-access to the equipment for this event
- Spent fuel racks and assemblies
  - The condition of spent fuel racks for the event was not calculated
  - Sliding, rocking and possibly contact between racks might occur for this event
    - Would not necessarily imply damage to the racks and assemblies (accelerations less than those expected in cask accidents) but would result in some re-arrangement of relative rack positions and clearances

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Status of Seismic and Structural Inputs

- Finalizing quantification of intermediate damage state and relative likelihood of damage states
  - State with no leakage is the most likely condition (preliminary result)
- Checking potential for leakage from the reactor at refueling conditions
- Verifying all results and documenting methods, assumptions, analyses and results in report

Scenario Delineation

Don Helton

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Scenario delineation

- Two SFP loading conditions to be considered:
  - Representative of the current situation for the selected site (i.e., high-density loading and a relatively full SFP)
  - Representative of expedited movement of older fuel to a dry cask storage facility (i.e., low-density loading)
- Successful deployment of mitigation and unsuccessful deployment of mitigation considered for each scenario
- Three SFP damage states considered
- Seismically-induced rupture to reactor piping considered
- Operating cycle is dissected in to 5 phases

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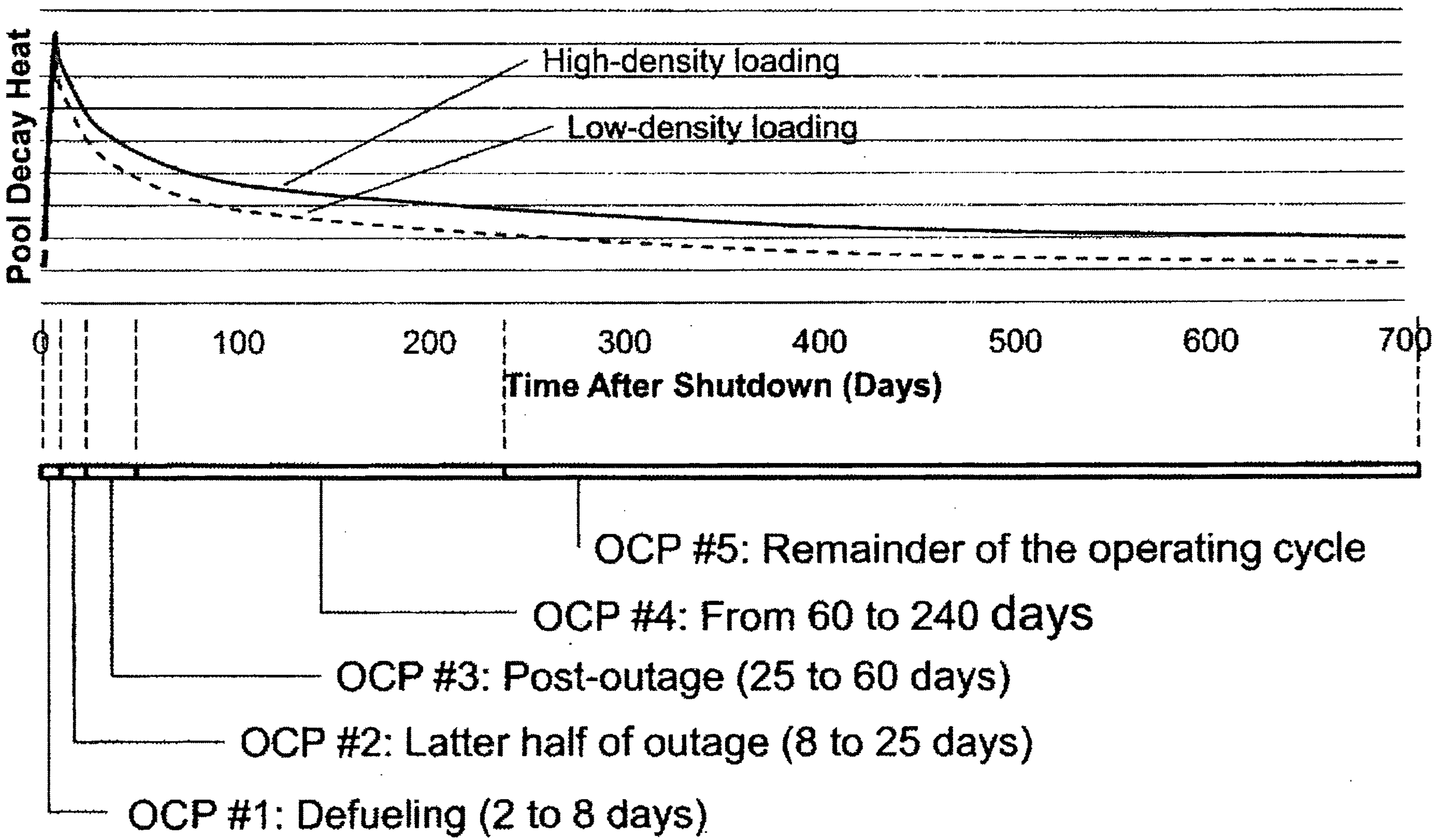
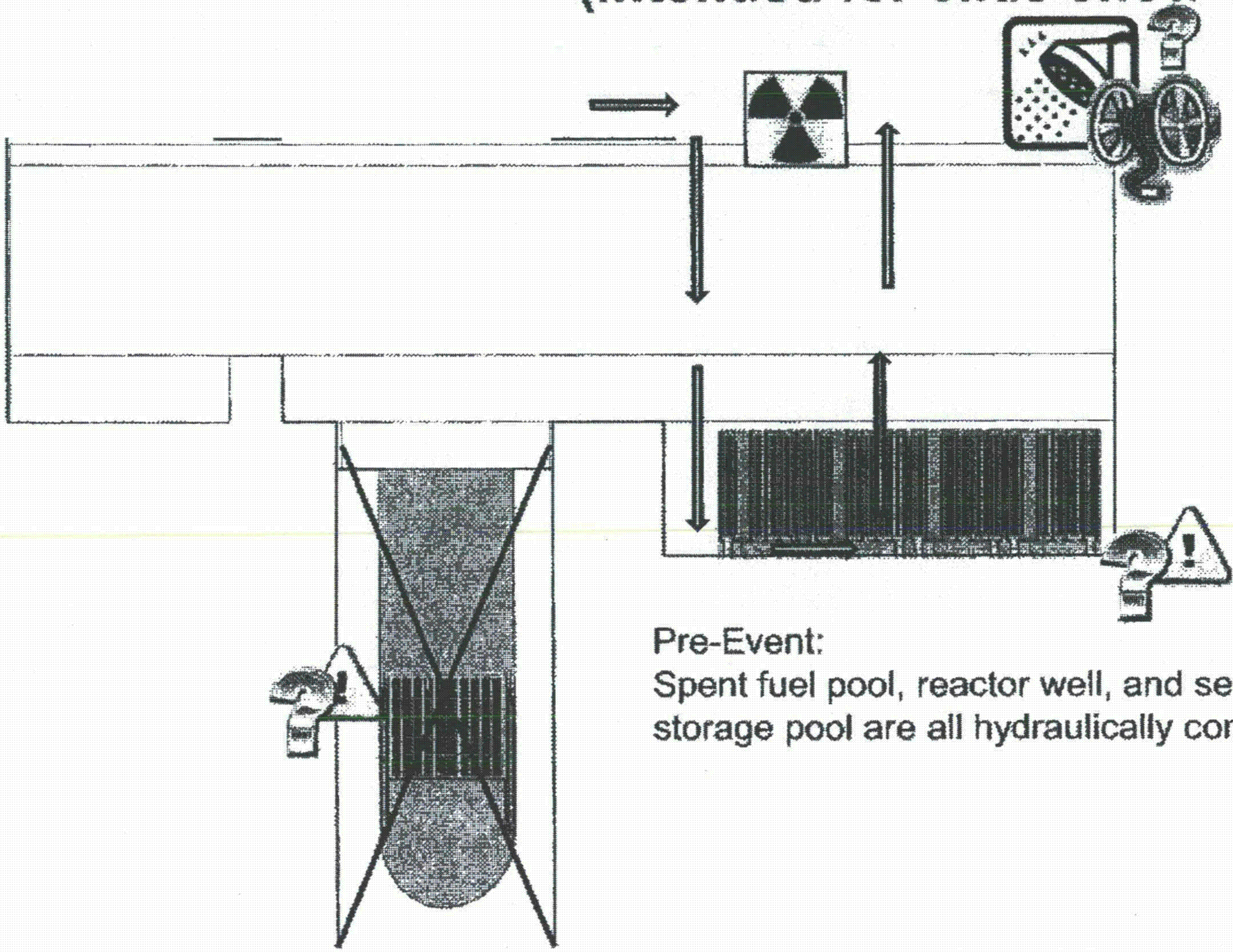
Illustration of Pool Decay Heat and Operating Cycle Phases (OCPs)



OCP Details

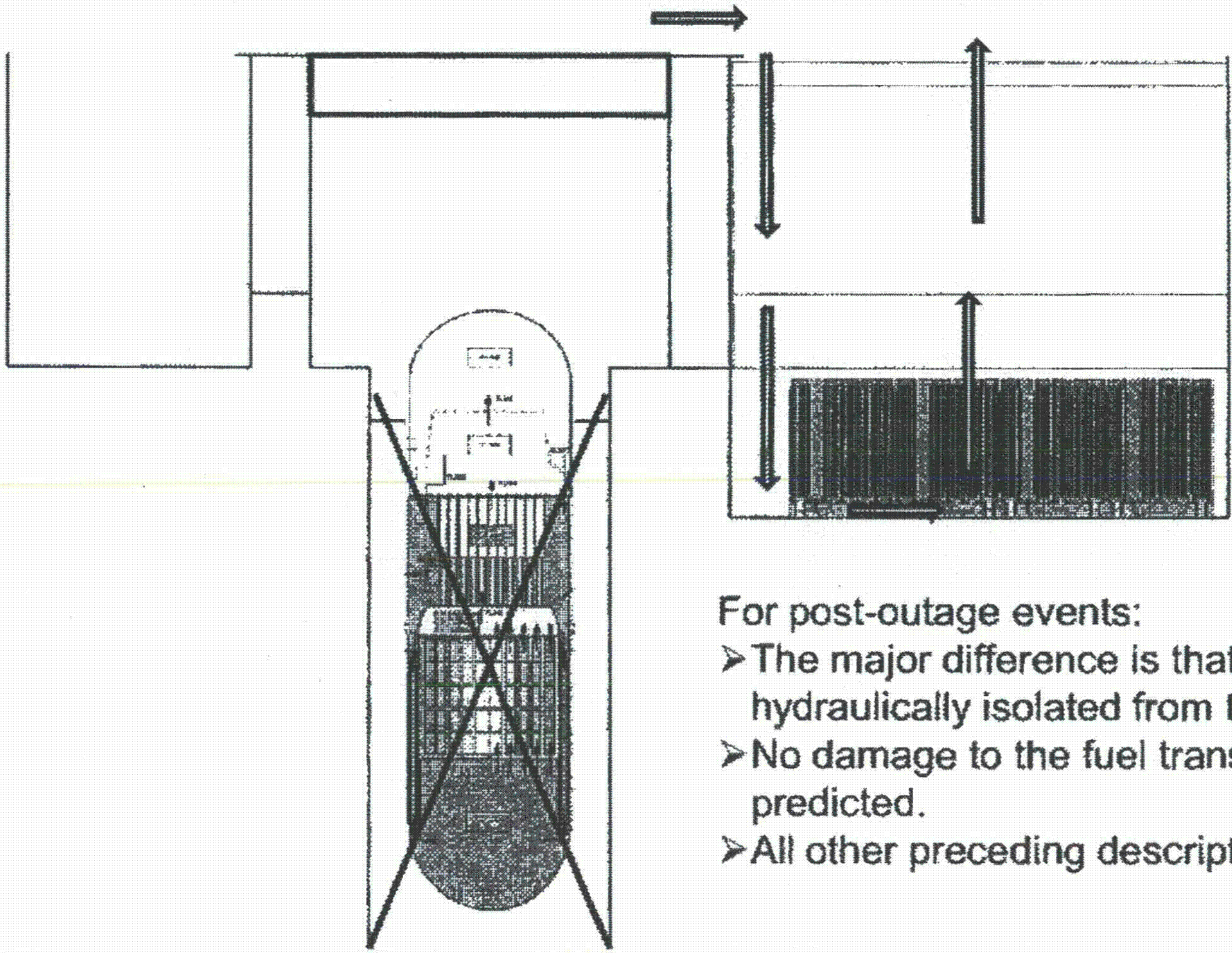
OCP #	Time window / (Time of evaluation) [days]	Fraction of operating cycle	Pool-reactor configur-ation	Spent fuel configuration for high-density loading
1	2 – 8 (5)	0.01	Refueling	Non-dispersed
2	8 – 25 (13)	0.02		
3	25 – 60 (37)	0.05	Un-connected	Dispersed
4	60 – 240 (107)	0.26		
5	240 – 700 & 0 – 2 (383)	0.66		

SFP Cartoon – During Outage
(Intended for slide show viewing)



Pre-Event:
Spent fuel pool, reactor well, and separator/dryer storage pool are all hydraulically connected.

SFP Cartoon – During Outage
(Intended for slide show viewing)



For post-outage events:
- The major difference is that the SFP is hydraulically isolated from the reactor.
- No damage to the fuel transfer canal gate is predicted.
- All other preceding descriptions remain valid.

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SCALE Analysis

- SCALE computer suite used to estimate refueling floor "shine" dose rates from fuel in SFP prior to radioactive release (to provide additional context)
  - Not used to directly affect mitigation assumptions since 50.54(hh)(2) includes consideration of deploying capabilities in high-radiation environments
- Leveraging a capability Oak Ridge National Labs used to inform Fukushima response – "low hanging fruit"
- Results indicate:
  - Worst-case locations exceed 25 rem/hour when SFP water level is 1-2 feet above top of fuel
  - Time since discharge has a limited effect on dose rates
- Additional analysis ongoing:
  - A single lifted assembly (event occurs during fuel handling)
  - Lower-density pool loading

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Mitigation Assumptions

- Re-arrangement of fuel to favorable configuration (for high-density loading) assumed to occur at the end of the outage
  - Represents a compromise between pre-configuring (which we believe the site did last outage) and waiting the full exemption time (which is a non-public value) or longer
- For scenarios not including mitigative actions:
  - No operator action is considered
- For scenarios including mitigative actions:
  - Diagnosis is assumed to take until SFP level drops 5 feet + 30 minutes for observation/decision-making (recall assumption of no AC power)
  - Capacities / timings follow underlying endorsed guidance in NEI-06-12, Revision 2
    - Except where an additional 3 hour delay is permitted above and beyond the 2 hour delay

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Mitigation Assumptions (2)

- Mitigation parameters:
  - 200 gpm spray (delivered; uniformly throughout SFP) or 500 gpm makeup
  - Higher site-specific flow rates not credited based on uncertainty in pump speed, pool coverage, etc.
  - Commences 2 hours <u>after diagnosis</u>
  - Mode is determined based on SFP water level at time of deployment
    - Later switching of mode is not considered based on modeling simplification and complications of initiating event
  - Once deployed, equipment runs indefinitely
    - Represents successful arrival of offsite support or deployment of other onsite assets
  - <u>Effectiveness is determined by MELCOR</u>

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Sample End-States
OCP #3 – Post-outage @ 37 days

Preliminary Results	Scenario Characteristics			Radioactive Release Commences Prior to 72 hours?	
Case #	SFP Leakage Rate?	Reactor Leakage?	Mitigation?	High-Density Loading	Low-Density Loading
1	None	N/A	Yes	No	No
2			No	No	No
3	Small		Yes	Still developing boundary conditions	
4			No		
5	Moderate		Yes	No	No
6			No	Yes	Yes

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Other Issues Not Addressed in Defining Scenarios

- Full core offload outages for vessel inspections
- Presence of new fuel in the SFP as source of zirconium (for a short period of time)
- Multi-unit effects
  - Only addressed until reactor/SFP become hydraulically decoupled
- Assembly in the process of being moved at time of event
- Inadvertent criticality events
- Recovery of offsite power
- Other long-term recovery actions

➢ The intent is to address as many uncertainties as practical via sensitivity studies

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Accident Progression Methods and Preliminary Results

Hossein Esmaili

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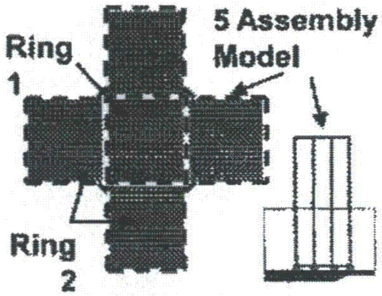
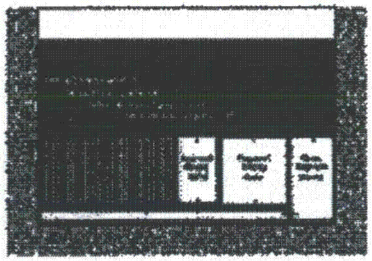
Use of MELCOR for SFP Analysis

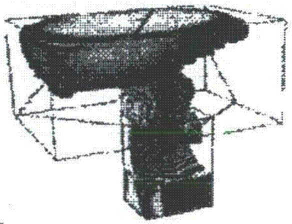
Analysis



MELCOR Separate Effects Analysis

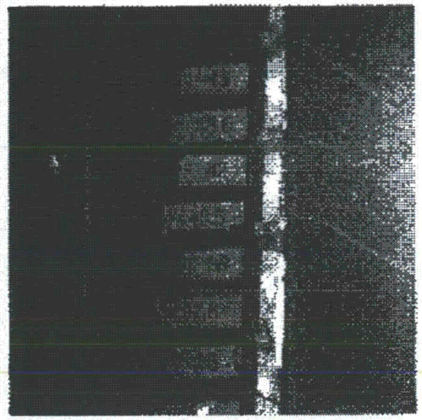


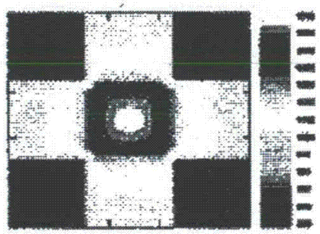
MELCOR Whole-Pool (i.e. Integrated) Analysis



Computational Fluid Dynamics

Experimental studies

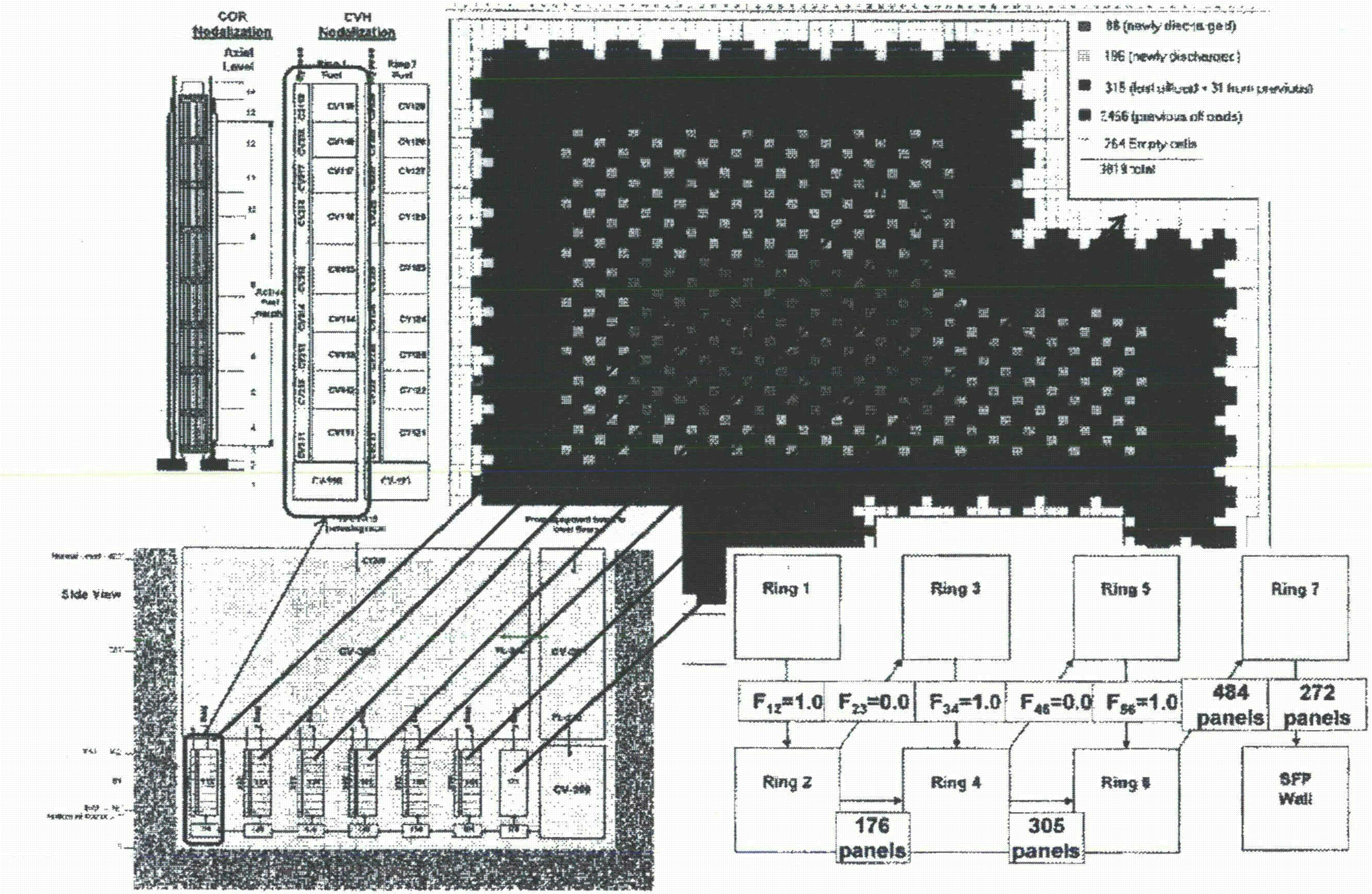




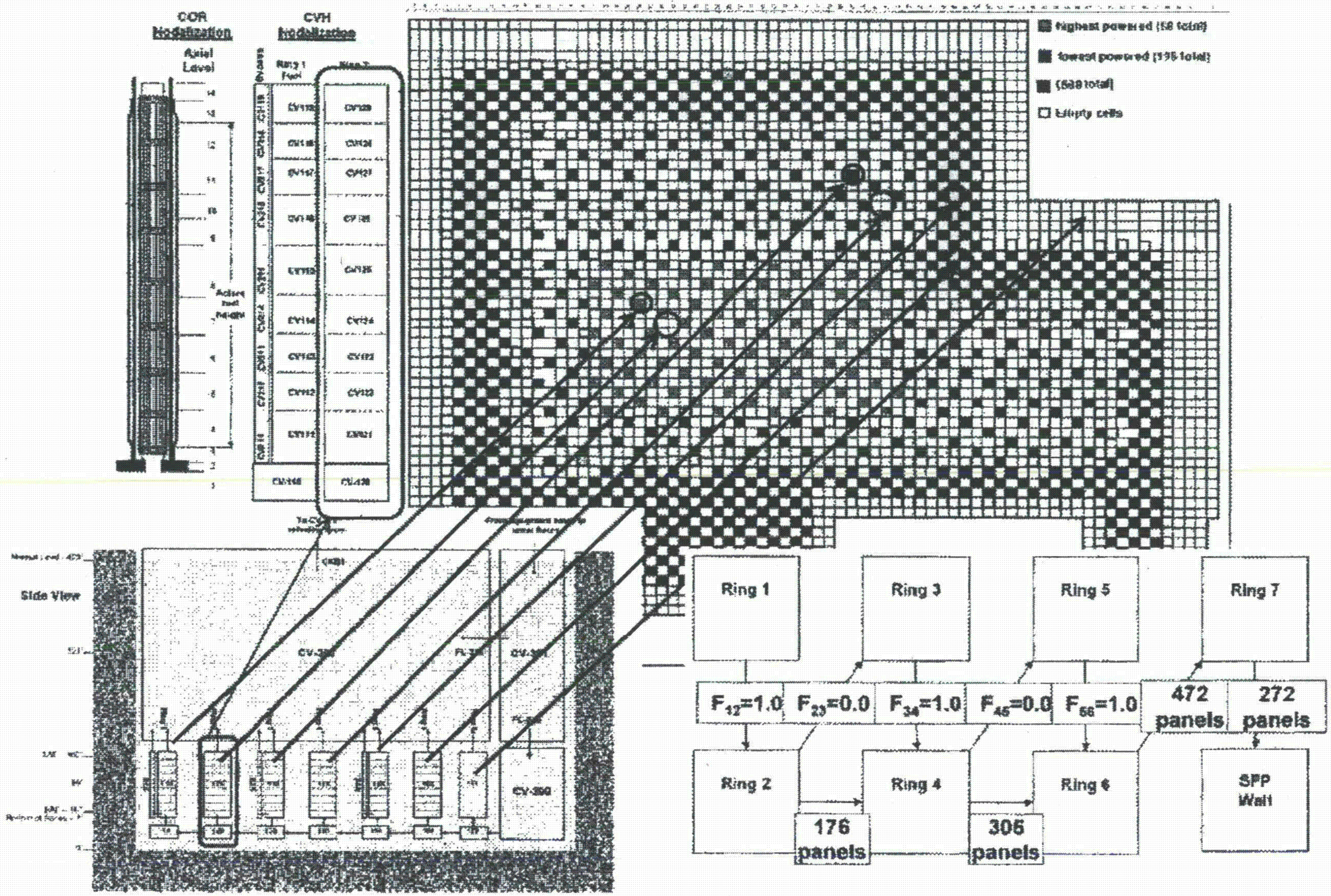
COBRA-SFS Analysis

~~OUO – Internal Use Only~~

High-Density Post-Outage SFP MELCOR Model



Low-Density Post-Outage SFP MELCOR Model



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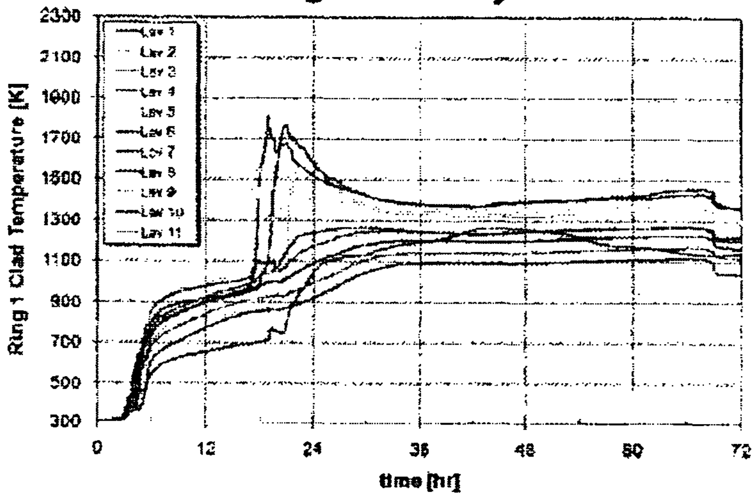
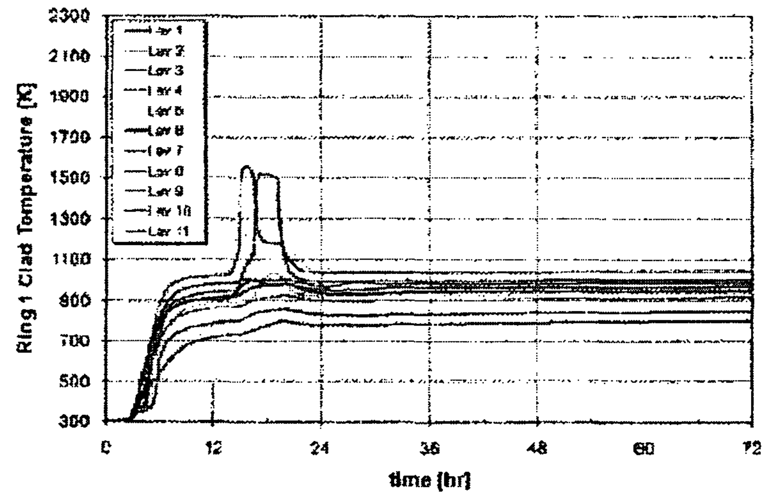
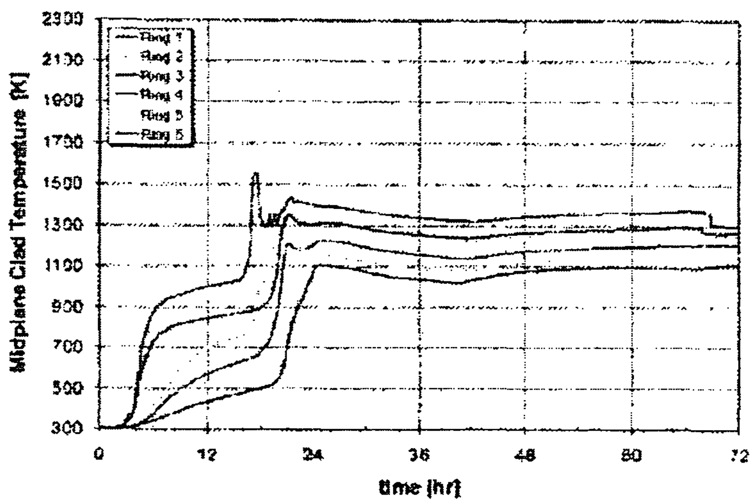
OCP 3 Preliminary Result (Moderate Bottom Hole - Unmitigated)

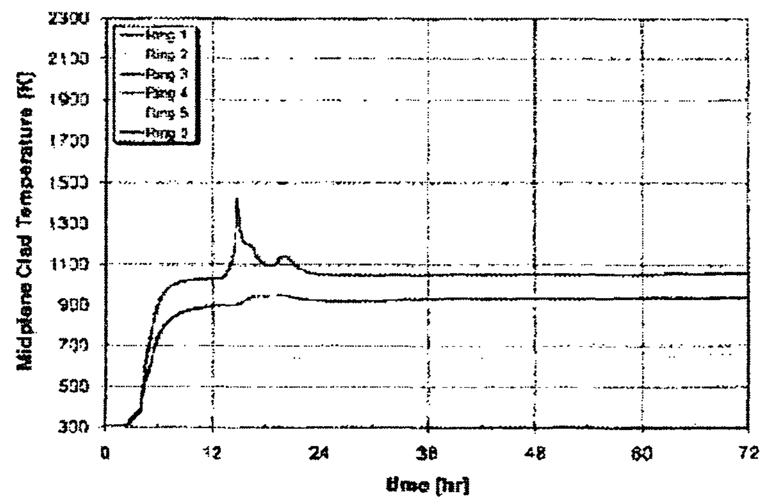
High Density



Low Density







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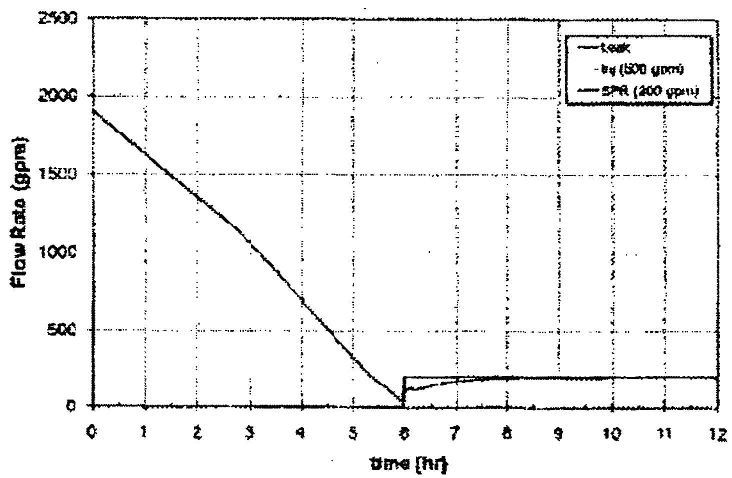
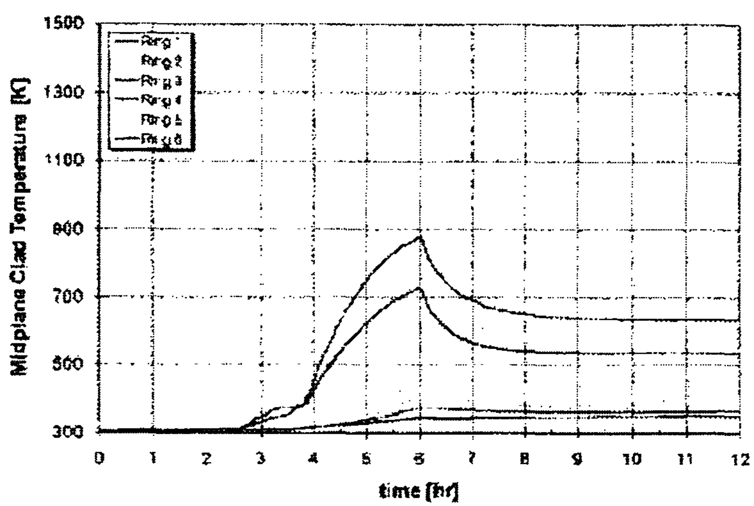
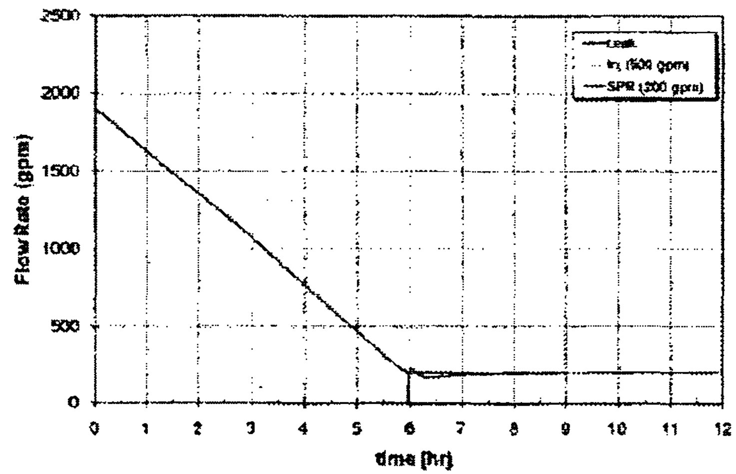
OCP 3 Preliminary Result (Moderate Bottom Hole - mitigated)

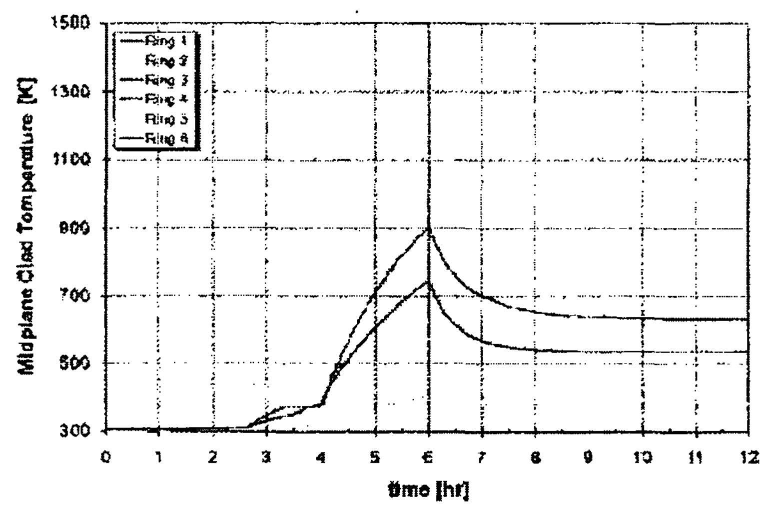
High Density





Low Density





OCP 3/4 Preliminary Results

	OCP3			
	High Loading (2.57 MW)		Low Loading (2.15 MW)	
	No mitigation	Mitigated	No mitigation	Mitigated
Low water level signal (hr)	0.47	0.47	0.47	0.47
Water Level at top of Racks (hr)	2.61	2.61	2.61	2.61
End of mitigation deployment (hr)	-	2.97	-	2.97
Water level at racks base plate (hr)	5.3	5.3	5.8	5.8
Time of Release (hr)	16.77	-	14.47	-
Time of spray actutation (hr)	-	5.98	-	5.98
Peak cladding temperature (K)	1823	876	1557	898

	OCP4			
	High Loading		Low Loading	
	No mitigation	Mitigated	No mitigation	Mitigated
Low water level signal (hr)	0.47	0.47	0.47	0.47
Water Level at top of Racks (hr)	2.61	2.61	2.61	2.61
End of mitigation deployment (hr)	-	2.97	-	2.97
Water level at racks base plate (hr)	5.3	5.3	5.8	5.8
Time of Release (hr)	-	-	-	-
Time of spray actutation (hr)	-	5.98	-	5.98
Peak cladding temperature (K)	943	720	896	730

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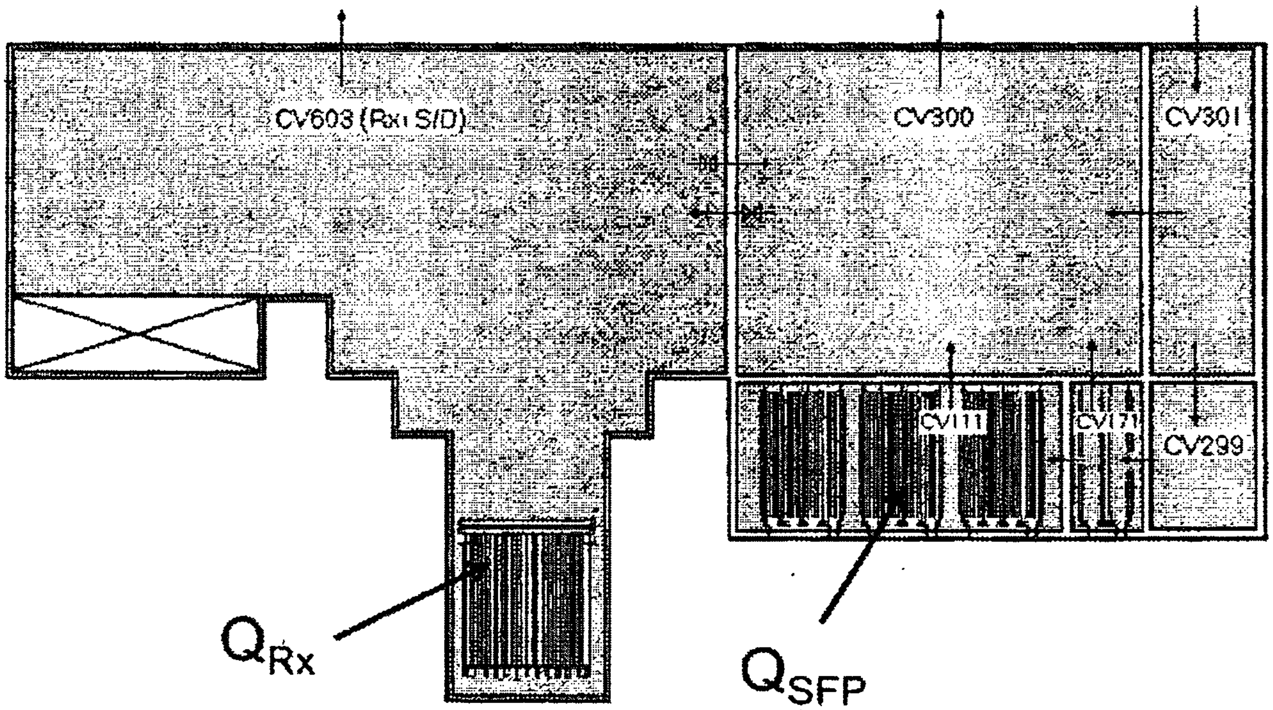
Simplified Boil-off Model

- Addition of a single control volume for the Separated/Dryer (S/D) pool and Reactor (Rx)
- Simplified SFP pool model (1 CV for empty racks, 1 CV for fueled assemblies)
- Flow between Rx+S/D to SFP controlled by valve (closed when level < SFP gate)
- Decay heat for the SFP and Rx are user specified energy source (no core model)



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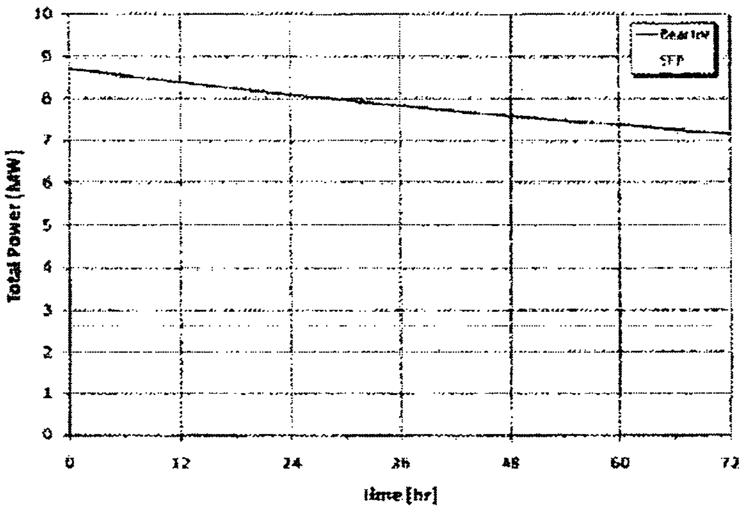
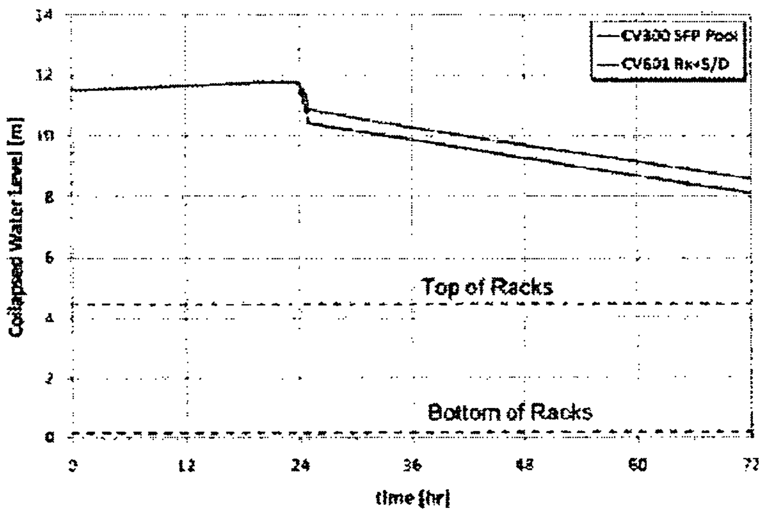
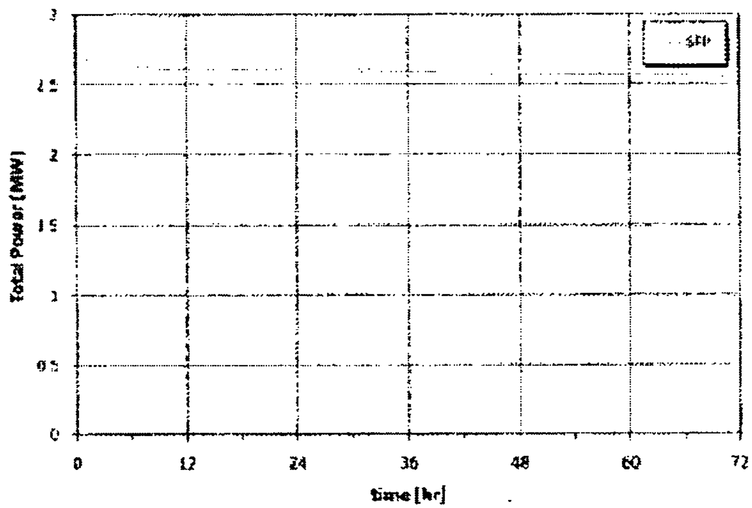
Demonstration Boil-off Results

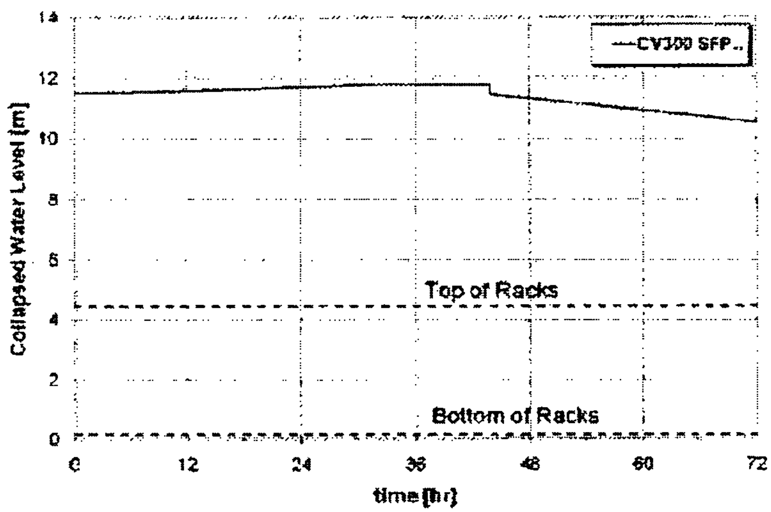
OCP1 (high density)





OCP3 (high density)





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Consequence Analysis Methods

AJ Nosek

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Offsite Consequence Modeling

- MACCS2 code will be used
  - Input: Accident source term, weather, population and economic data, protective measures
  - Output: Consequences (e.g. contamination, health effects) from atmospheric release
- Modeling will leverage best practices from draft NUREG-1935 (SOARCA)
- Population and economic data updated for 2011

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MACCS2 Modeling: Code Modules

- ATMOS
  - Atmospheric transport and dispersion
  - Source term from MELCOR and ORIGEN calculations

- EARLY (1 week)
  - Early consequences
    - Altered by countermeasures such as sheltering, evacuation, and relocation
  - Early doses also contribute to long-term consequences

- CHRONC (50 years)
  - Long-term consequences
    - Altered by countermeasures such as decontamination, interdiction, and condemnation

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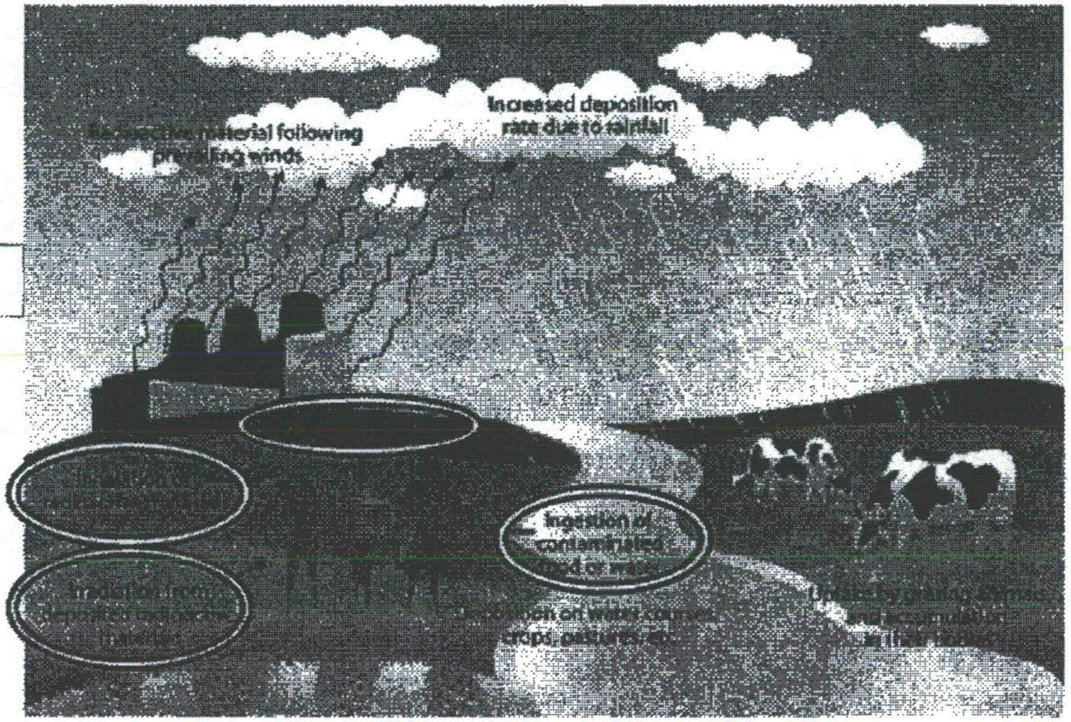
MACCS2 Modeling: Atmospheric Release and Exposure Pathways

MACCS2 models the radioactive release to the atmosphere (e.g. plume rise, dispersion, dry and wet deposition)



MACCS2 estimates the health effects from: inhalation, cloudshine, groundshine, skin deposition, and ingestion (e.g. water, milk, meat, crops)

MACCS2 Modeling: Atmospheric Transport and Dispersion (ATD)

- Dispersion based on Gaussian plume model (with provisions for meander and surface roughness effects)
  - Phenomena not treated in detail in this model are: Irregular terrain, spatial variations in wind field, temporal variations in wind direction
- Meteorological data required
  - Wind direction and speed, Pasquill stability category, precipitation, (seasonal [PM,AM]) mixing layer height, and boundary weather
- Multiple weather sequences (accounts for uncertainty in weather conditions at the time of the accident)

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MACCS2 Modeling: Protective Measures

- Emergency phase response
  - Evacuation
    - 360 degree, 10-mile radius (Pennsylvania specific)
    - 7 Cohorts will represent different groups of the public
    - Road network route considerations
    - Seismic-specific considerations
    - Start of evacuation will depend on sequence
  - Sheltering
  - Relocation: two triggers
  - KI ingestion
- Long-term phase response
  - Relocation
  - Decontamination, condemnation, and interdiction (from both food/water and land)
  - Trade-offs exist between health effects and economic costs

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MACCS2 Modeling: Dose and Health Effects

- Stochastic health effects (e.g. latent cancer fatalities)
  - Three dose response models
    - Linear, no threshold (LNT) hypothesis
    - Linear, low-dose truncation
      - 620 mrem/yr (U.S. average dose)
      - 5 rem/yr or 10 rem lifetime (HPS position)
  - Risk reduction factor of 2 for low dose and dose rates to account for low-LET radiation (DDREF)
- Deterministic health effects (e.g. early fatalities)
  - Values informed from the joint NRC/CEC (Commission of the European Communities) expert elicitation study
- Federal Guidance Report 13
  - Dose conversion factors: pathway, organ, and nuclide specific
  - Includes 825 nuclides and acute capabilities
  - Incorporates ICRP-68/72 Standard and BEIR V
  - Most current federal guidance published by EPA

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Consequence Reporting

- Health Effects
  - Primary Plan: Report the conditional risk of early fatalities and latent cancer fatalities as related to distance from the site. (Ideal for informing individual members of the public)

- Land Contamination
  - Primary Plan: Report total land contamination for the site region above a specified dose level (e.g., the habitability criterion for the selected site of 500 mrem/year)
  - Additional Plan: Report land contamination above a specified activity level (e.g. Cs-137 Bq/m^2)

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Consequence Reporting

When is land considered "contaminated"?

When (after the accident) should we evaluate land contamination?

Radiation Dose Limit to Members of the Public				
Annual Dose (mrem)	Standard/Regulation/Guide	Timeframe	Applicability	Notes
???	EPA/NRC long-term cleanup standard	long-term	severe accidents	currently in rulemaking
25	CFR 20 subpart E	long-term	decommissioning	unrestricted use of land
100	CFR 20 subpart D	operation	licensed operation	
100	ICRP standard	long-term	severe accidents	with "optimization"
100	DHS Protective Action Guide	long-term	severe accidents	with "optimization"
500	Pennsylvania Code Title 25 § 219.51	after emergency	severe accidents	
2000	EPA Protective Action Guide	after emergency	severe accidents	for first year (500 mrem thereafter)

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Concluding Remarks and Questions

Katie Wagner

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Coordination and Communication

- Internal:
  - Working with Japan Lessons Learned Project Directorate Steering Committee on path forward
  - Input from program offices
  - Briefings for Senior Management and Commissioners
- Interactions with licensee
- ACRS:
  - Informal discussion
  - Closed March Subcommittee meeting
  - Planned open April Full Committee meeting (methods only)
- External:
  - Mentioning of study in Congressional testimony, senior management presentations, etc.
  - RIC 2012 presentation
  - A communication plan has been drafted (currently working w/OPA)

SFPSS Project Team

- Katie Wagner – Overall project lead
- Hossein Esmaili – Accident progression lead
- Don Helton – Boundary conditions and probabilistic aspects lead
- Andy Murphy – Seismic analysis lead
- AJ Nosek – Offsite consequence lead
- Jose Pires – Structural analysis lead

Powell, Eric

From: Helton, Donald
Sent: Thursday, May 31, 2012 7:56 AM
To: Powell, Eric
Cc: Wagner, Katie
Subject: RE: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

RES

Eric,

In using the terminology I've used in the SFPSS report, I'm trying to stick to the following convention:

Security-related events = loss of large area of the plant (explosions or malevolent aircraft impact) + other land-based or water-based attacks + insider threat --> by direction, not considered in PRAs (for now) due to issues in estimating initiating event frequencies (note that these events are outside the scope of SOARCA, SFPSS, and the Vogtle Level 3 PRA – they are handled via deterministic programs and requirements)

Inadvertent aircraft impact = NPP near an airport has some low likelihood of being struck by a malfunctioning aircraft during initial ascent or descent, and is a safety-related (as opposed to security-related) hazard --> this accident has been considered in past PRAs (including both past agency SFP PRAs; it was not a selected sequence for either SOARCA or SFPSS; while for Vogtle it is TBD whether it would be screened in or out but is, at least theoretically, within scope)

Does that make sense? Does it suggest any change to the report to alleviate your concern?

Best,
Don

From: Powell, Eric
Sent: Wednesday, May 30, 2012 5:32 PM
To: Helton, Donald
Cc: Wagner, Katie
Subject: RE: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

NRO

Don,

Thanks for your thorough review of my comments.

Numbers 1, 3, & 4 below are addressed by your response. However, I have a question about #2 to make sure I understand what you are saying. Specifically, using the language "aircraft crash (or impact)" is done because of previous SFP PRA studies? I can understand the sabotage aspect of your comment, but from my perspective saying loss of large area of the plant due to fire or explosions covers 50.54(hh)(2) and AIA scenarios. However, if other losses, caused by something other than an aircraft crash, have never really been analyzed in PRA space I can understand the difficulty that would raise and the constraints that limited resources plays into this effort (SFPSS).

Like I said, from my perspective saying aircraft crashes is very specific and loss of large areas is more inclusive to other threats.

Thanks,
Eric

A5-11

From: Helton, Donald
Sent: Tuesday, May 29, 2012 11:00 AM
To: Powell, Eric
Cc: Wagner, Katie
Subject: RE: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

RES

Eric,

I've gone through all your comments, and there were a few that seemed to warrant a response, which I've provided below. If there are any of these that you believe need to be added to the "unresolved" list, please let me know.

Thanks for finding the time to review the report!
Don

1) Q: Is this [not addressing many multi-unit aspects] really appropriate given the events at Fukushima? More of a philosophical question, but I would say that it is not appropriate given how such an event would affect multiple units on a site. .. [From later Section] EThis appears to address my earlier comment. However, I would like to note that looking at only a single unit event, given a threat which affects more than a single unit, is a dated approach and inconsistent with the lessons that have been learned since the accident at Fukushima.
A: Nonetheless, that is the state-of-practice for this project and almost all others. Do you see this as substantively different then the way new reactor licensing (e.g., SAMDAs) are currently conducted?

2) Q: While this [referring to the mentioning of inadvertent aircraft crashes on page 5] is true, 50.54(hh)(2) looks at losses of large areas from any number of threats. Why not use language similar? Is it the sabotage aspect? ..... Why [not include security-related events in SFPSS]? With the NTT Task Force recommendation of making the 50.54(hh)(2) equipment more robust and required (or whatever language they use), why not look at sabotage events in a general manner. Instead of saying aircraft crashes, we could say losses of large areas of the plant due to any number of threats. I think it would be more flexible and comprehensive this way.
A: The first instance is specifically referring to the safety-related consideration of an inadvertent aircraft crash, which was studied in the past SFP PRAs. Intentional aircraft crashes are not considered in SFPSS because (i) their effect has already been addressed via the SFP security assessments, (ii) the agency's position on not quantifying risk from security-related events due to the inability to quantify the initiating event frequency, and (iii) the need to reduce scope to manage schedule/resources.

3) Q: Based on the guidance in NEI 06-12, the site will have sufficient fuel for the pumping source to operate for 12 hours without off-site support. This means that the 50.54(hh)(2) equipment relied upon to mitigate an event is only required to be able to operate for that amount of time. This is a potential issue, because if the study is assuming off-site support doesn't arrive until 24 hours after the event (and implemented sometime after) then for at least 12 hours the plant will not be able to provide makeup or spray to the SFP.
A: As we discussed in the meeting last week, this situation represents an intermediate state between the mitigated and unmitigated scenarios.

4) Q: Revision 3 [to NEI-06-12] was issued in September 2009.
A: Correct, but it only applies to new reactors. The endorsed revision for operating reactors is Revision 2.

From: Wagner, Katie
Sent: Tuesday, May 22, 2012 4:44 PM
To: Esmaili, Hossein; Helton, Donald; Murphy, Andrew; Nosek, Andrew; Pires, Jose
Subject: FW: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

FSME

Forwarding

From: Powell, Eric
Sent: Tuesday, May 22, 2012 4:43 PM
To: Wagner, Katie
Cc: Ader, Charles; Mrowca, Lynn; Weerakkody, Sunil
Subject: RE: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

NRO

Katie,

In the attached document you will find my comments.

Thanks,
Eric

From: Wagner, Katie
Sent: Tuesday, May 15, 2012 7:42 PM
To: Barto, Andrew; Sullivan, Randy; Schrader, Eric; Jones, Steve; Mitman, Jeffrey; Bowman, Eric; Witt, Kevin; Tegeler, Bret; Powell, Eric
Cc: Gibson, Kathy; Scott, Michael; Poole, Brooke; Lewis, Robert; Ruland, William; Giitter, Joseph; McGinty, Tim; Ader, Charles; Bergman, Thomas; Skeen, David; Evans, Michele; Clifford, James; Lee, Richard; Coyne, Kevin; Hogan, Rosemary; Santiago, Patricia; Wood, Kent; Hansell, Samuel; Ennis, Rick; Esmalli, Hossein; Helton, Donald; Murphy, Andrew; Nosek, Andrew; Pires, Jose
Subject: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

~~RES~~ FSME dup.

Attachment contains OUO – Sensitive Internal Information

All,

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- This is a working draft which has not been formally reviewed by the SFPSS Team or technical writing editors yet due to the tight schedule involved. The formatting of the document has not been finalized at this time. Equations which did not show up in the main .pdf document are attached in a MS Word file.
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- RES and NRR are still determining the path and schedule to obtain public comment on the report and providing it to the Commission.
- Division Directors have been cc'ed on this email at the request of RES Division Management.

Please let me know if you have any questions or comments.

Thanks,

Katie Wagner
General Engineer
U.S. Nuclear Regulatory Commission
(301) 251-7917
Katie.Wagner@nrc.gov

Schaperow, Jason

From: Schaperow, Jason
Sent: Tuesday, June 05, 2012 1:46 PM
To: Santiago, Patricia
Subject: RE: DRAFT SFPSS report for review by EOB on June 4th
Attachments: RE: SFPSS meeting

NRO

The analysis does not represent the Peach Bottom Atomic Power Station and is too simplified and conservative to be useful for regulatory purposes. My preliminary comments (in the attached email) need to be resolved by revising the analysis and not by adding qualitative discussion to the report. Because no one has resolved my preliminary comments, the value of additional review is unclear.

From: Santiago, Patricia
Sent: Tuesday, June 05, 2012 12:50 PM
To: Schaperow, Jason
Subject: RE: DRAFT SFPSS report for review by EOB on June 4th

RES

Can you tell me? The whole thing if you like but it is 177 pages so we should focus on what we can review to assist them. AJ analyses will not be done until Randy provides some EP input.
thanks

From: Schaperow, Jason
Sent: Tuesday, June 05, 2012 12:40 PM
To: Santiago, Patricia
Cc: Nosek, Andrew
Subject: RE: DRAFT SFPSS report for review by EOB on June 4th

NRO

Thanks. What part(s) of this report do you want me to review?

From: Santiago, Patricia
Sent: Tuesday, June 05, 2012 12:35 PM
To: Schaperow, Jason
Cc: Nosek, Andrew
Subject: FW: DRAFT SFPSS report for review by EOB on June 4th

RES

You did get it but this is what AJ gave me
And let's set a time to discuss any input to AJ/Katie
Thanks!

From: Wagner, Katie
Sent: Tuesday, May 29, 2012 5:33 PM
To: Lee, Richard; Coyne, Kevin; Hogan, Rosemary; Santiago, Patricia
Cc: Schaperow, Jason; Madni, Imtiaz; Esmaili, Hossein; Helton, Donald; Murphy, Andrew; Nosek, Andrew; Pires, Jose
Subject: DRAFT SFPSS report for review by EOB on June 4th

FSME

Good Afternoon Richard, Kevin, Rosemary, and Pat,

Attached is the draft SFPSS report for your review, which is about 200 pages long. We ask that you complete your review by EOB on Monday, June 4th. Here are a few notes about this document:

- The SFPSS team is still working to resolve comments and have not reviewed each other's sections in detail. However, the team is comfortable sending this draft forward for review at this time.
- The equations do not show up in this version of the document, so an equation list is attached for your information.

A5-127

- Some MACCS2 runs have not been completed at this time due to the pending implementation of new evacuation models developed by NSIR in cooperation with Sandia staff. The team does not expect most of the results to change dramatically once those are complete and I will send you the final offsite consequences chapter once that chapter is complete.
- The concurrence package is currently in the process of coming together. In the package, the report will be an enclosure to a memo with the following characteristics (I will write the memo and send it to you in the next day or two):
    - Will be from B. Sheron to E. Leeds.
    - Content:
        - Discuss that the SFPSS is a RES product that was executed as laid out in the July 2011 project plan.
        - Discuss that the SFPSS is for NRR consideration as part of the Tier 3 Japan Lessons Learned item.
        - Will contain a placeholder for a few main conclusions.
- Some comments which the team has not resolved have been added to a tracking list, which is attached.
- OPA sent extensive comments on the abstract and executive summary of the report on the afternoon of Thursday, May 24th. The SFPSS team has not processed these comments yet.
- I received comments from QTE on the afternoon of Tuesday, May 29th. The SFPSS team has not processed these comments yet either.

Note: Jason and Imtiaz have been cc'ed since they are acting for Pat this week.

Thanks,

Katie Wagner
General Engineer
U.S. Nuclear Regulatory Commission
(301) 251.7917
Katie.Wagner@nrc.gov

Schaperow, Jason

NRC dup.

From: Schaperow, Jason
Sent: Wednesday, May 23, 2012 8:25 AM
To: Helton, Donald
Cc: Wagner, Katie
Subject: RE: SFPSS meeting

My issues are as follows:

• Seismic initiator - The analysis does not include a concurrent reactor accident.
• Arrangement of fuel - Peach Bottom uses a 1x8 arrangement of fuel, not the 1x4 arrangement assumed in the study.
• Pool damage - Fukushima shows that an earthquake would not make a hole in a spent fuel pool.
• Mitigation - Peach Bottom-specific mitigation measures are not credited.
• Mitigation - Makeup and spray are likely, because the spent fuel pool is an open system and there is a long time available until draindown and fuel damage. Also, offsite equipment began arriving at Fukushima within about 8 hours (INPO report of November 2011).
• Mitigation - The operators are likely to make openings in the reactor building to aid in spent fuel pool cooling and to prevent a buildup of hydrogen from a concurrent reactor accident.
• Mitigation - For one of the "mitigated" cases, the analysis assumes makeup when spray is needed (and available) to prevent fuel overheating.
• Mitigation - The "unmitigated" cases include some B.5.b mitigation, namely, arranging the fuel in a favorable pattern for cooling.
• Release from clad-pellet gap - The assumed release of cesium (magnitude of 0.05, chemical form CsOH) is conservative.
• Release from fuel pellet - The modeling was validated using in-pile tests for reactor accidents, which is not prototypical of spent fuel pool accidents which progress more slowly and have lower fuel temperatures.
• Hydrogen combustion - A single node is used for the area between the refueling floor the reactor building roof. Simple parametric modeling is used for determining whether there will be a burn.
• Public evacuation - Assuming that we can evacuate tens and even hundreds of thousands of people but we cannot get a couple of people up to the spent fuel pool with a fire hose seems illogical.
• Public evacuation - NRC recommended a 50-mile evacuation for Fukushima.
• Public evacuation - MELCOR and MACCS analysis was used for developing evacuation and relocation assumptions, instead of RASCAL.
• Results - The consequence/risk results presented in the study assume the probability of mitigation is zero.

NRO

Schaperow, Jason

From: Schaperow, Jason
Sent: Tuesday, June 05, 2012 1:46 PM
To: Santiago, Patricia
Subject: RE: DRAFT SFPSS report for review by EOB on June 4th
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ıject: FW: DRAFT SFPSS report for review by EOB on June 4th

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FSME

Wagner, Katie
Tuesday, May 29, 2012 5:33 PM
, Richard; Coyne, Kevin; Hogan, Rosemary; Santiago, Patricia
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AS-13

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Note: Jason and Imtiaz have been cc'ed since they are acting for Pat this week.

Thanks,

Katie Wagner
General Engineer
U.S. Nuclear Regulatory Commission
(301) 251.7917
Katie.Wagner@nrc.gov

Mrowca, Lynn

From:	Powell, Eric
Sent:	Tuesday, August 28, 2012 11:27 AM
To:	Mrowca, Lynn
Subject:	FW: UPDATE #3: HRA associated with RES Spent Fuel Pool Scoping Study (SFPSS)
Attachments:	SFPSS HRA Collaboration Group.docx

From: Wagner, Katie
Releasable FSME
Sent: Thursday, July 12, 2012 8:10 AM
To: Barto, Andrew; Powell, Eric; Tegeler, Bret; Jones, Steve; Mitman, Jeffrey; Bowman, Eric; Wood, Kent; Ennis, Rick; Sullivan, Randy; Schrader, Eric; Witt, Kevin; Bloom, Steven
Cc: Helton, Donald; Esmaili, Hossein; Murphy, Andrew; Nosek, Andrew; Pires, Jose
Subject: FW: UPDATE #3: HRA associated with RES Spent Fuel Pool Scoping Study (SFPSS)

SSPSS Other-Office Working Group and Others,

Below is the third update on the recently-added HRA to keep you in the loop.

Also, this afternoon (4-5pm EST) SFPSS team members will be briefing Regions I and IV at the division level (via VTC (image only) and teleconference (sound)) with the other Regions invited to participate via teleconference. The briefing slides will be extremely similar to those used for the Cmr. Magwood briefing on July 2nd.

Thanks,

Katie Wagner
General Engineer
U.S. Nuclear Regulatory Commission
(301) 251.7917
Katie.Wagner@nrc.gov

RES

From: Coe, Doug
Sent: Tuesday, July 10, 2012 12:39 PM
To: Ruland, William; Clifford, James; Roberts, Darrell; Lewis, Robert; Thaggard, Mark; Evans, Michele; Giitter, Joseph; Lee, Samson; Wilson, Peter; Monninger, John; Weaver, Doug; Ader, Charles; Lombard, Mark; Gibson, Kathy; Scott, Michael
Cc: Correia, Richard; Peters, Sean; Uhle, Jennifer; Chang, James; Wagner, Katie; Weerakkody, Sunil; Fleger, Stephen; Witt, Kevin; Ennis, Rick; Kahler, Robert; Sullivan, Randy; Helton, Donald; Mitman, Jeffrey; Zoulis, Antonios; Cahill, Christopher; Taylor, Robert; Coyne, Kevin; Keighley, Elizabeth
Subject: UPDATE #3: HRA associated with RES Spent Fuel Pool Scoping Study (SFPSS)

To: Division Directors supporting/engaged on subject HRA for SFPSS initiative

1. The staff has revised the project plan based upon comments received to date. We expect to send the project plan out for concurrence this week, in parallel with management briefings on the plan. We expect that once the plan is sufficiently vetted, we will ask the RES FO to ticket it and thereby provide a definitive tasking and schedule to be completed, for planning purposes.

2. *Our RES PM James Chang (RES), along with Jeff Mitman (NRR), Anthony Zoulis(NRR), and Chris Cahill (RI) will visit Peach Bottom on July 12 to gather specifics for the analysis*. The visit will consist of accident mitigation walkthroughs and walkdowns of the main control room, water treatment building, fire header, cooling tower basin, B5b equipment storage location, spent fuel pool cooling pump, refueling floor, and the spent fuel pool

   As a reminder, the enhancements to mitigation strategies (to resist external hazards) expected from implementing the recent orders will NOT be included in this first HRA study, but we will hold the option of another HRA study when those enhancements are in place.

A5-14 F1..

4. The NTTF Tier 3 information Commission paper is up to OEDO. For recommendation 5 (consideration of spent fuel transfer from pools to dry storage) it notes that in addition to the Spent Fuel Scoping Study and this HRA study, the staff will also study the effects on fuel safety associated with increased fuel handling incident to an accelerated dry cask storage campaign. NMSS is in the lead for this latter item but is planning to engage RES for support. Additionally, similar *direction on studying fuel handling may be forthcoming from the Commission* in the currently DRAFT SRM – M120607C (from the recent ACRS meeting with the Commission).

5. I have attached an updated Collaboration Group table based upon inputs received to date. *This is the list we will use to communicate these updates.* Please let James Chang, Sean Peters, Rich Correia, and I know if you would like to make any changes.

Thanks again and please let us know if you have any questions, comments, or concerns.
Doug

Doug Coe
Deputy Director
Division of Risk Analysis (DRA)
Office of Nuclear Regulatory Research (RES)
U.S. Nuclear Regulatory Commission
Rockville, MD
301-251-7914
doug.coe@nrc.gov

Powell, Eric

BKDA FSM

From: Wagner, Katie
Sent: Tuesday, May 15, 2012 7:56 PM
To: Barto, Andrew; Sullivan, Randy; Schrader, Eric; Jones, Steve Mitman, Jeffrey; Bowman, Eric; Witt, Kevin; Tegeler, Bret; Powell, Eric
Cc: Gibson, Kathy; Scott, Michael; Poole, Brooke; Lewis, Robert; Ruland, William; Giitter, Joseph; McGinty, Tim; Ader, Charles; Bergman, Thomas; Skeen, David; Evans, Michele; Clifford, James; Lee, Richard; Coyne, Kevin; Hogan, Rosemary; Santiago, Patricia; Wood, Kent; Hansell, Samuel; Ennis, Rick; Esmaili, Hossein; Helton, Donald; Murphy, Andrew; Nosek, Andrew; Pires, Jose
Subject: RE: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

All – Please note that the results of the consequence analysis are preliminary because they are calculated with a 10-mile evacuation model. While a 10-mile evacuation model is applicable to some smaller releases, it is not realistic for some of the larger releases. Updated results with the new evacuation models are expected in the coming weeks. – Thanks, Katie

RES FSM

From: Wagner, Katie
Sent: Tuesday, May 15, 2012 7:42 PM
To: Barto, Andrew; Sullivan, Randy; Schrader, Eric; Jones, Steve; Mitman, Jeffrey; Bowman, Eric; Witt, Kevin; Tegeler, Bret; Powell, Eric
Cc: Gibson, Kathy; Scott, Michael; Poole, Brooke; Lewis, Robert; Ruland, William; Giitter, Joseph; McGinty, Tim; Ader, Charles; Bergman, Thomas; Skeen, David; Evans, Michele; Clifford, James; Lee, Richard; Coyne, Kevin; Hogan, Rosemary; Santiago, Patricia; Wood, Kent; Hansell, Samuel; Ennis, Rick; Esmaili, Hossein; Helton, Donald; Murphy, Andrew; Nosek, Andrew; Pires, Jose
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Katie Wagner
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(301) 251.7917
Katie.Wagner@nrc.gov

A5-15

Powell, Eric

BAtta FBM

From:	Wagner, Katie
Sent:	Tuesday, May 15, 2012 7:56 PM
To:	Barto, Andrew; Sullivan, Randy; Schrader, Eric; Jones, Steve; Mitman, Jeffrey; Bowman, Eric; Witt, Kevin; Tegeler, Bret; Powell, Eric
Cc:	Gibson, Kathy; Scott, Michael; Poole, Brooke; Lewis, Robert; Ruland, William; Glitter, Joseph; McGinty, Tim; Ader, Charles; Bergman, Thomas; Skeen, David; Evans, Michele; Clifford, James; Lee, Richard; Coyne, Kevin; Hogan, Rosemary; Santiago, Patricia; Wood, Kent; Hansell, Samuel; Ennis, Rick; Esmaili, Hossein; Helton, Donald; Murphy, Andrew; Nosek, Andrew; Pires, Jose
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Katie Wagner
General Engineer
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(301) 251.7917
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Powell, Eric

From: Weerakkody, Sunil
Sent: Wednesday, May 23, 2012 11:45 AM
To: Powell, Eric
Subject: RE: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

NRR

Nice work! Thanks for copying. I browsed through some of the comments.

Sunil

NRO ok to release

From: Powell, Eric
Sent: Tuesday, May 22, 2012 4:43 PM
To: Wagner, Katie
Cc: Ader, Charles; Mrowca, Lynn; Weerakkody, Sunil
Subject: RE: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

Katie,

In the attached document you will find my comments.

Thanks,
Eric

RES FSME

From: Wagner, Katie
Sent: Tuesday, May 15, 2012 7:42 PM
To: Barto, Andrew; Sullivan, Randy; Schrader, Eric; Jones, Steve; Mitman, Jeffrey; Bowman, Eric; Witt, Kevin; Tegeler, Bret; Powell, Eric
Cc: Gibson, Kathy; Scott, Michael; Poole, Brooke; Lewis, Robert; Ruland, William; Glitter, Joseph; McGinty, Tim; Ader, Charles; Bergman, Thomas; Skeen, David; Evans, Michele; Clifford, James; Lee, Richard; Coyne, Kevin; Hogan, Rosemary; Santiago, Patricia; Wood, Kent; Hansell, Samuel; Ennis, Rick; Esmaili, Hossein; Helton, Donald; Murphy, Andrew; Nosek, Andrew; Pires, Jose
Subject: ACTION: Spent Fuel Pool Scoping Study (SFPSS) Working Draft for Working Group Review by EOB on Tue., May 22nd

Attachment contains OUO – Sensitive Internal Information

All,

Attached is a working draft for review and comment by Other-Office Working Group members by end-of-business on Tuesday, May 22nd. We understand that this is a large document and the review time is relatively short, we appreciate your input to help meet our deadlines. A few notes about this document:

- This is a working draft which has not been formally reviewed by the SFPSS Team or technical writing editors yet due to the tight schedule involved. The formatting of the document has not been finalized at this time. Equations which did not show up in the main .pdf document are attached in a MS Word file.
- The current plan is that the SFPSS report will be sent by memo from Brian Sheron, RES to Eric Leeds, NRR at the end of June for their consideration as part of the NTTF Tier 3 issue regarding transfer of spent fuel from pools to casks.
- RES and NRR are still determining the path and schedule to obtain public comment on the report and providing it to the Commission.
- Division Directors have been cc'ed on this email at the request of RES Division Management.

Please let me know if you have any questions or comments.

Summary of Comments on SFPSS Report - ELP Comments.pdf

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Author: ELP1 Subject: Sticky Note Date: 5/22/2012 2:05:18 PM
Mitigative strategies are what we have required

Author: ELP1 Subject: Highlight Date: 5/22/2012 2:05:07 PM

Author: ELP1 Subject: Highlight Date: 5/22/2012 2:05:05 PM

Author: ELP1 Subject: Sticky Note Date: 5/22/2012 2:08:38 PM
drain down event has occurred

Author: ELP1 Subject: Highlight Date: 5/22/2012 2:05:00 PM

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Author: ELP1 Subject: Highlight Date: 5/22/2012 2:23:39 PM

Author: ELP1 Subject: Sticky Note Date: 5/22/2012 2:25:11 PM
See highlighted section below (#8)

Author: ELP1 Subject: Sticky Note Date: 5/22/2012 2:26:04 PM
Has there been discussion about some kind of requirement in this area? Just a thought.

Author: ELP1 Subject: Highlight Date: 5/22/2012 2:23:48 PM

Author: ELP1 Subject: Highlight Date: 5/22/2012 2:23:45 PM

Author: ELP1 Subject: Sticky Note Date: 5/22/2012 2:29:46 PM
I think you mean "safety"

Author: ELP1 Subject: Sticky Note Date: 5/22/2012 2:32:31 PM
This doesn't seem to be consistent with numbers 5 & 8 above. On second thought, I guess saying that something could be safer in a low-density configuration, isn't the same as saying high density configurations are not safe. I will continue reading to understand this point better, but I wanted to put a note here anyways.

Author: ELP1 Subject: Highlight Date: 5/22/2012 2:29:26 PM

Author: ELP1 Subject: Highlight Date: 5/22/2012 2:29:24 PM

Author: ELP1 Subject: Highlight Date: 5/22/2012 2:27:57 PM

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Author: ELP1 Subject: Sticky Note Date: 5/22/2012 3:11:41 PM

Why? With the NTT Task Force recommendation of making the 50.54(hh)(2) equipment more robust and required (or whatever language they use), why not look at sabotage events in a general manner. Instead of saying aircraft crashes, we could say losses of large areas of the plant due to any number of threats. I think it would be more flexible and comprehensive this way.

Author: ELP1 Subject: Highlight Date: 5/22/2012 3:07:22 PM

Author: ELP1 Subject: Highlight Date: 5/22/2012 3:12:31 PM

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Author: ELP1 Subject: Sticky Note Date: 5/22/2012 4:11:24 PM

Based on the guidance in NEI 06-12, the site will have sufficient fuel for the pumping source to operate for 12 hours without off-site support. This means that the 50.54(hh)(2) equipment relied upon to mitigate an event is only required to be able to operate for that amount of time. This is a potential issue, because if the study is assuming off-site support doesn't arrive until 24 hours after the event (and implemented sometime after) then for at least 12 hours the plant will not be able to provide makeup or spray to the SFP.

Author: ELP1 Subject: Highlight Date: 5/22/2012 4:02:40 PM

RES/FSME

Scoping Study on the Consequences of a Well-Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool at a Selected US Mark I Boiling Water Reactor

Interim Report for External Stakeholder Review and Comment
June 2012

Project Manager
Katie Wagner – now in FSME

Co-Authors
Hossein Esmaili
Donald Helton
Andrew Murphy
Andrew Nosek
Jose Pires

Office of Nuclear Regulatory Research
US Nuclear Regulatory Commission

Note – This report will eventually be made publicly available. In cases where information currently included is specifically believed to be non-public (for other than pre-decisional reasons), these passages are [redacted].

A5-16

ABSTRACT

This report provides the methodology and results of a limited-scope consequence study to update the best-estimate consequences expected from the application of a postulated well-beyond-design-basis earthquake (with an estimated frequency of occurrence of 1 event in 61,000 years) to a selected US Mark I boiling water reactor spent fuel pool. The primary objective of the study is to provide updated and publicly available consequence estimates of a representative, postulated spent fuel pool severe accident under high-density and low-density loading conditions to frame ongoing discussions regarding the impact (from a defense-in-depth perspective) of requiring operators of US nuclear power plants to expedite movement of fuel from the spent fuel pool to onsite, dry cask storage.

Regarding the event considered in this report, it is important to remember that the study is intentionally focusing on a very challenging and very unlikely event. All spent fuel pools are designed to seismic standards consistent with other important safety-related structures on the site. The pool and its supporting systems are located within structures that protect against natural phenomena and flying debris. The pools' thick walls and floors provide structural integrity and further protection of the fuel from natural phenomena and debris. In addition, the deep water above the stored fuel (typically more than 20 feet above the top of the spent fuel rods) would absorb the energy of debris that could fall into the pool. Finally, the racks that support the fuel are designed to keep the fuel in its designed configuration after a seismic event. It is only because such a challenging event is studied that any offsite consequences are predicted.

ACKNOWLEDGEMENTS

The authors would like to acknowledge the following individuals who directly or indirectly contributed to this work. The following individuals directly contributed in the form of analyses and support:

Don Algama, US Nuclear Regulatory Commission
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Larry Humphries, Sandia National Laboratories
Joe Jones, Sandia National Laboratory
Thomas Miler, Oak Ridge National Laboratory
Bruce Patton, Oak Ridge National Laboratory
Joel Piper, Rotatee from the Department of Homeland Security
Kenneth C. Wagner, dycoda, LLC

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Eric Bowman	Jeff Mitman	Randy Sullivan	
Sam Hansell	Eric Powell	Bret Tegeler	
John Hughey	Wayne Schmidt	Kent Wood	

Finally, the authors wish to acknowledge the guidance and support of NRC management, and in particular Brian Sheron, who provided the initial direction for the study.

EXECUTIVE SUMMARY

Various risk studies (most recently NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001) have shown that storage of spent fuel in a high-density configuration in SFPs is safe and that the risk is appropriately low. These studies used simplified and sometimes bounding assumptions and models for characterizing the likelihood and consequences of beyond-design-basis SFP accidents. As part of NRC's post-9/11 security assessments, SFP modeling using detailed thermal-hydraulic and severe accident progression models integrated into the MELCOR code were developed and applied to assess the realistic heatup of spent fuel under various pool draining conditions. Moreover, in conjunction with these post-9/11 security assessments, NRC has required enhancements via 10 CFR 50.54(hh)(2) for operating reactor SFP storage that are directed at further improving the coolability of spent fuel under event conditions in which a substantial amount of water has drained from the storage pool. These mitigative strategies are being re-visited as part of the agency's Japan Lessons Learned imitative, and Orders have been issued to further upgrade nuclear power plants' capabilities in these areas.

Recently, the agency restated its views on the safety of spent fuel stored in high-density configurations in a number of forums, including its response to Petition for Rulemaking (PRM)-51-10 and PRM-51-12, the revision to NUREG-1437 (the Generic Environmental Impact Statement for License Renewal) and various responses to external stakeholders related to the March 2011 events at the Japanese Fukushima Daiichi reactors. However, this position relies in part on the findings of the aforementioned security assessments, which are not publicly available. The renewed interest in spent fuel storage engendered from the changes in path forward of the planned geologic repository and from the aforementioned events in Japan has rekindled interest in capturing the consequences from postulated accidents associated with high-density SFP storage in an updated safety study. An SFP risk study is being planned as part of a larger initiative involving the conduct of a site Level 3 probabilistic risk assessment (PRA), as documented in SECY-11-0089 and the associated Staff Requirements Memorandum.

In the interim, a desire existed to produce updated consequence estimates for a particular scenario of interest that can in part act as a bridge between the current state-of-knowledge (much of which is greater than 10 years old or security-related and, thus, nonpublicly available) and future studies. The current study fulfills that gap, investigating an important piece of the overall puzzle. Other pieces of this puzzle haven been informed by past studies, will be addressed by future studies, or will be addressed through other inputs in to the decisionmaking process. This bigger picture is addressed via a Japan Lessons Learned activity related to transfer of spent fuel from pool to cask storage.

Two broad situations are considered in this study, which represent the following:

(1) A condition representative of the current situation for the selected site (i.e., high-density loading in the SFP, a relatively full SFP, and current regulatory requirements with respect to fuel configuration and preventive/mitigative capabilities); and

(2) A condition where expedited movement of older fuel to dry cask storage has already been achieved (i.e., low-density loading in the SFP and current applicable regulatory requirements with respect to fuel configuration and preventive/mitigative capabilities).

For purposes of obtaining a near-term perspective on the issue, a single site and single assumed operating cycle are used. The site characterization (seismic response, decay heat, radionuclide inventory, etc.) have been based on readily available information that primarily stemmed from sources such as the NUREG-1150 study, seismic information developed by the United States Geological Survey (USGS), and the post 9/11 security assessments[1]. Later in the project, additional information was provided by the licensee which generally corroborates the assumptions made in this study.

A boiling water reactor plant was chosen for this analysis, in part because these types of reactors often engender more interest from external stakeholders owing to the fact that the Mark I and Mark II designs have spent fuel pools are elevated relative to ground level. In the context of safety-related events (as opposed to security-related events), the elevation of the pool will affect the transmission of seismic loads through the structure, can potentially inhibit accessibility for taking mitigative action, and can potentially lead to flooding of safety-related equipment, if the pool and surrounding structures are significantly damaged. The selection of a boiling water reactor design is not intended to suggest that these designs are more vulnerable to spent fuel pool accidents. In reality, there are differences between the major design types (PWRs versus BWRs) which make each more or less susceptible to spent fuel pool accidents on a scenario-specific basis. Similarly, the selection of a site that has a separate SFP for each reactor (as opposed to a shared pool) is also not intended to suggest that these situations are inherently more vulnerable.

The study, which is described in much greater detail in the associated report, has led to the following preliminary findings. *All of the findings below are subject to the limitations spelled out in the associated report.*

1. Past studies have indicated that large seismic events could lead to the loss of structural integrity of the spent fuel pool liner. The current work further confirms that such a condition is unlikely but possible. The specific conditions under which a failure might occur are very site-specific.

2. Past studies have indicated that events that cause a loss of pool cooling, but not structural failure of the pool liner, lead to very slowly progressing accidents. For the conditions studied here, no set of conditions short of liner failure were sufficient for leading to a radioactive release in less than 3 days. In most cases, the available time to prevent the accident was much greater than 3 days.

3. In the cases studied, which in general did not account for multiple and concurrent reactor / SFP accidents, the precise time to diagnose the need for SFP mitigation did not have a strong effect on the course of events.

    - Nevertheless, the improved reliable and available SFP indication required by the Order of March 12, 2012, is still appropriate given its essential nature in ensuring that plant personnel can effectively prioritize emergency actions. The availability of such instrumentation may have changed the mode (makeup versus sprays) deployed for some situations studied here.

[1] The post 9/11-security assessments included consideration of SFPs and resulted in the collection of information and the development of computer models that provided a convenient starting point for the current study.

4. This study considered variations in both pool loading configuration and the effective deployment (or lack thereof) of mitigation. Of these, effective deployment of mitigation had the largest impact on <u>preventing</u> a release of radioactive material.

5. Results show that the use of favorable fuel configurations (i.e., 1x4) has a very positive effect in promoting coolability and reducing the likelihood of a release should the SFP become drained. In this study, without mitigative action, fuel was air coolable at 60 days[2], as opposed to estimates of roughly 1 year from past studies when hotter fuel was contiguously stored. For the pattern currently employed at Peach Bottom (1x8), an even shorter coolability time is likely.

The following additional recommendations are preliminary, awaiting the outcome of the ongoing consequence analyses. For the time being, they are based on preliminary results and/or insights form past studies.

6. The difference between high-density and low-density loading situations were as follows:

   - In terms of whether a release occurred within 3 days (i.e., likelihood), no difference was seen.
   - In terms of the size of releases (i.e., an indicator of consequences), in the case that a release did occur a large difference was seen.
   - Overall (i.e., likelihood and consequences) the scenario-specific offsite consequences were higher for high-density loading cases, but below past estimates for both.

7. For the situations where a radioactive release was predicted, emergency preparedness was successful in reducing the likelihood of early health effects to essentially zero.

8. For the situations where a radioactive release was predicted, the predicted frequency of occurrence of latent health effects was largely affected by the pool loading configuration.

9. The general reduction in estimated consequences for the range of seismic events studied here (0.5 to 1g), along with the relative increase in the frequency of larger seismic events (> 1g) in the newer hazard updates, suggests that the leading contribution to risk may have shifted to these extremely large events. If the conditional probabilities of having a release for the extremely large events is commensurate with that from the events studied here, the overall estimated scenario frequencies would still be well below those of past studies. This assumption presumes that the damage to the fuel, racks, and overlying building is not substantively different from that assumed here.

10. This study has not uncovered any information that would challenge the existing agency position that spent fuel is stored safety in high-density configurations. Further, it does not suggest that past SFP risk estimates from large seismic events are non-conservative. Thus, under the existing regulatory framework, it does not challenge the established regulatory position that expedited fuel movement to casks is justified from a value/impact perspective, even without consideration of the negative aspects of expedited fuel movement (e.g., worker dose).

[2] The actual time evaluated was 37 days for that operating cycle phase. There is reason to believe that the fuel may be air coolable in less than 60 days for a wide range of conditions, based on other separate effects analyses.

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1. INTRODUCTION AND BACKGROUND

All operating reactors in the US are of the light-water reactor design. They utilize upright fuel assemblies (usually roughly 12 feet in length) with low-enriched Uranium oxide fuel (less than 5% Uranium-235). The fuel assemblies (which are comprised of numerous fuel rods – typically 80-100 rods for contemporary boiling water reactor fuel and 200 – 300 rods for contemporary pressurized water reactor fuel) are placed in the reactor for 2 to 3 operating cycles. Each operating cycle typically lasts 18 to 24 months. At the end of their reactor "life" the assemblies are placed in large pools of water adjacent to the reactor (how adjacent depends on the plant design) that are roughly 12 meters (40 feet) deep. The fuel assemblies are later loaded in to casks, which are then moved to an Independent Spent Fuel Storage Installation (ISFSI), usually onsite. The casks are drained of water and inerted with helium during the loading process. This situation leads to the vernacular terms of "wet storage" (to describe storage in the spent fuel pool) and "dry storage" (to describe storage in casks).

Spent fuel pools in the United States were originally designed to store 1-2 reactor cores worth of spent fuel, so that the fuel could "cool down" (become less thermally and radioactively "hot") prior to its movement to a long-term storage repository. Owing to delays in the identification, licensing and construction of such a repository, US nuclear power plants "re-racked" their spent fuel pools in the 1980s and 1990s to allow for the storage of larger numbers of spent nuclear fuel assemblies (e.g., roughly 4 reactor cores worth for the plant studied in this report). Throughout this time (including present day), the NRC has maintained that spent fuel pools provide adequate protection of the public health and safety in either low-density or high-density storage configurations. The basis for this position is discussed later in this section.

As also described later in this section, external stakeholders have periodically challenged this position. This tension is natural because there are a number of complex considerations when weighing low-density versus high-density spent fuel storage. In understanding this tension, let's first start with a few basic tenets:

- Thermal and radioactivity loads associated with freshly-discharged fuel necessitate the need for wet storage
- All spent nuclear fuel, regardless of age (a.k.a, time since discharge from the reactor) produces both heat and radiation

Now, let's consider some less-obvious considerations. The list below presents considerations from the perspective of the pros and cons associated with postulated transitioning from the existing usage of high-density racking in the US back to the usage of low-density storage. The list is subdivided in to two parts, those considerations that are covered within this study, and those that are not.

Considerations covered within this study

- Expedited movement of fuel from the spent fuel pool to dry storage will decrease the inventory of longer-lived radionuclides such as Cesium-137 present in the spent fuel pool
- As a result of the above, less radioactive material would be present in the pool if a radioactive release occurred, which would be expected to reduce potential land contamination and economic impacts

- Removal of older fuel reduces the overall heat load in the pool, which can have the effect of delaying the start of a radioactive release (and thus increasing the time available to take mitigative action) for many types of accidents
- Removal of older fuel will increase the area available for air circulation (natural circulation) should the pool become completely drained (the effect of this is somewhat limited by the nature of spent fuel racks as discussed later in this report)
- Removal of older fuel will increase the volume available for cooling water (note that this is mathematically a small effect with the older fuel comprising on the order of 5% of the total pool volume – recall that most of the pool is occupied by water, not fuel)

Considerations not covered within this study

- Current regulatory requirements under 10 CFR Part 72 limit (from a practicality perspective) the ability to transfer fuel with less than roughly 3 to 5 years of cooling
- Discharging large amounts of fuel (and thus greatly increasing the amount of fuel contained in the ISFSI) might require a Part 72 rulemaking effort (e.g., to accommodate increases in the design-basis accident site area boundary allowable dose limits) and potentially increase the risk associated with the ISFSI
- Expedited discharging of fuel from the spent fuel pool to dry storage increases the frequency of postulated cask drops, which in turn increases the risk of causing damage to the pool that could lead to a radioactive release
- Expedited discharging of fuel increases occupational doses for workers involved with the management and transfer of the spent fuel
- Earlier movement of fuel increases the probability that fuel will have to be re-packaged later for shipment to the eventual long-term repository or interim storage site

For each of the items covered by this report, these items are re-visited in Section 10 by reflecting on the quantitative results provided by this study. The reader may quickly note that the first set of considerations are generally "pros" associated with expedited fuel movement to casks, while the latter list are generally "cons". Why focus on the pros for this study? The agency's position, that spent fuel storage in either pools or casks is safe is based on a number of past studies and regulatory activities that are discussed later in this chapter. This regulatory position is solid. In re-assessing this position, we have started by investigating whether any of the "pros" are more compelling than past studies suggest. If they are, then the issue can be addressed more holistically to see if new information challenges the existing regulatory position. Otherwise, there is insufficient motivation to spend the additional agency resources associated with a more holistic study, and these resources are better devoted to other aspects of the agency's mission of protecting people and the environment.

1.1. Project Impetus

Various risk studies (most recently NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001) have shown that storage of spent fuel in a high-density configuration in SFPs is safe and that the risk is appropriately low. These studies used simplified and sometimes bounding assumptions and models for characterizing the likelihood and consequences of beyond-design-basis SFP accidents. As part of NRC's post-9/11 security assessments, SFP modeling using detailed thermal-hydraulic and severe accident progression models integrated into the MELCOR code were developed and applied to assess the realistic heatup of spent fuel under various pool draining conditions.

Moreover, in conjunction with these post-9/11 security assessments, NRC has required enhancements via 10 CFR 50.54(hh)(2) for operating reactor SFP storage that are directed at further improving the coolability of spent fuel under event conditions in which a substantial amount of water has drained from the storage pool.

Recently, the agency has restated its views on the safety of spent fuel stored in high-density configurations in a response to Petition for Rulemaking (PRM)-51-10 and PRM-51-12 as well as the revision to NUREG-1437 (the Generic Environmental Impact Statement for License Renewal, Draft Report for Comment). However, this position relies in part on the findings of the aforementioned security assessments, which are not publicly available. The renewed interest in spent fuel storage engendered from the changes in path forward of the planned geologic repository and from the events in Japan following the March 2011 earthquake has rekindled interest in capturing the consequences from postulated accidents associated with high-density SFP storage in an updated safety study. An SFP risk study is being planned as part of a larger initiative involving the conduct of a site Level 3 probabilistic risk assessment (PRA), as documented in SECY-11-0089 and the associated Staff Requirements Memorandum. In the interim, a desire existed to produce updated consequence estimates for a particular scenario of interest that can in part act as a bridge between the current state-of-knowledge (much of which is greater than 10 years old or security-related and, thus, nonpublicly available) and future studies. The current study fulfills that gap, investigating an important piece of the overall puzzle. Other pieces of this puzzle haven been informed by past studies, will be addressed by future studies, or will be addressed through other inputs in to the decisionmaking process.

1.2. Technical Approach

Two broad situations are considered in this study, which represent the following:

(1) A condition representative of the current situation for the selected site (i.e., high-density loading in the SFP, a relatively full SFP, and current regulatory requirements with respect to fuel configuration and preventive/mitigative capabilities); and

(2) A condition where expedited movement of older fuel to dry cask storage has already been achieved (i.e., low-density loading in the SFP and current applicable regulatory requirements with respect to fuel configuration and preventive/mitigative capabilities).

For purposes of obtaining a near-term perspective on the issue, a single site and single assumed operating cycle are used. The site characterization (seismic response, decay heat, radionuclide inventory, etc.) have been based on readily available information that primarily stemmed from sources such as the NUREG-1150 study, seismic information developed by the United States Geological Survey (USGS), and the post 9/11 security assessments[3]. Later in the project, additional information was provided by the licensee which generally corroborates the assumptions made in this study.

A boiling water reactor plant was chosen for this analysis, in part because these types of reactors often engender more interest from external stakeholders owing to the fact that the Mark I and Mark II designs have spent fuel pools are elevated relative to ground level. In the context of safety-related events (as opposed to security-related events), the elevation of the pool will

[3] The post 9/11-security assessments included consideration of SFPs and resulted in the collection of information and the development of computer models that provided a convenient starting point for the current study.

affect the transmission of seismic loads through the structure, can potentially inhibit accessibility for taking mitigative action, and can potentially lead to flooding of safety-related equipment, if the pool and surrounding structures are significantly damaged. The selection of a boiling water reactor design is not intended to suggest that these designs are more vulnerable to spent fuel pool accidents. In reality, there are differences between the major design types (PWRs versus BWRs) which make each more or less susceptible to spent fuel pool accidents on a scenario-specific basis. Similarly, the selection of a site that has a separate SFP for each reactor (as opposed to a shared pool) is also not intended to suggest that these situations are inherently more vulnerable.

1.3. Site Familiarization

The site selected (Peach Bottom) has two General Electric Type 4 boiling water reactors with Mark I containments. Unit #3 is the focus of this study (Unit #1 is no longer in operation). Each reactor has a dedicated spent fuel pool, and the pools do not share a common refueling floor, as is the case with some plants of this design. Most other aspects of the reactor, spent fuel pool, and reactor building are similar to BWR designs of this vintage. Two small power uprates have been approved for this site (1995 and 2002), with an extended power uprate submittal planned by the licensee in the near future.

Figure 1: The Peach Bottom Site

Regarding the SFPs, the existing high-density racks were placed in-service in 1986, and were designed and manufactured by Westinghouse Electric Corporation. As of 2010, the Unit #3 SFP contained 2,945 assemblies, while the Unit #2 SFP contained 2,844. Both SFPs maintain enough open locations to allow for an emergency full core offload, if required. The site also has an Independent Spent Fuel Storage Installation (ISFSI) for dry cask storage, utilizing the TN-68 cask design.

Finally, with respect to Emergency Preparedness, the site is located in a state (Pennsylvania) which has state-specific Protective Action Guidelines.

1.4. Basic Scenario Development

There are a few key aspects of the way this study is conducted that bare mentioning at this point. These are:

- A large seismic event is the only initiator considered.
- As mentioned previously, both the current situation (a high-density loading configuration in the pool) and an alternate situation (low-density loading configuration in the pool) are analyzed; a situation where the pool has been re-racked to a low-density rack configuration is not considered, because such a situation would be inefficient in terms of regulatory benefit given that much of the benefit of this situation could be achieved by storing less fuel in the existing racks (recall that BWR fuel is channeled, which reduces the benefit of cross flow if the pool were to become drained).
- The study focuses on the spent fuel pool, not the reactor, though for instances where the two are hydraulically connected, both are considered to a certain extent.
- The study does not attempt to quantify the reliability of mitigation, but rather treats every scenario considering both the case with successful mitigation deployment and the case with unsuccessful mitigation deployment.
- All portions of the operating cycle are considered.
- Detailed computer modeling is used to predict the plant's response to the event, in terms of structural response, accident progression, mitigation effectiveness (when credited), and offsite consequences.

In cases where the above represent limitations on the study's scope or results, these are justified in this report. In particular, Section 2 of this report provides the study's key limitations and assumptions.

1.5. Rationale for Focusing on Seismic Hazard

This section seeks to provide context regarding the suite of potential initiating events that can lead to a SFP accident, and why a seismic event is the focus of this study.

A typical risk assessment would start with a comprehensive look at relevant initiating events, and would conduct a *failure modes and effect analysis*. For the present study, due to (i) relative simplicity of the SFP and its supporting infrastructure as compared to a reactor and its supporting infrastructure and (ii) the much lower assembly decay heats, the majority of potential SFP accident risk is believed to emanate from either:

- Events that have the potential to cause a sizable leak in the SFP, or
- Events that might preclude operator action to inject water in to the pool for an extended period of time (e.g., days)

When one considers the various possible initiators, the former criteria points to:

1. Very large (i.e., well beyond the design-basis) seismic events
2. Heavy load (e.g., cask) drops, and
3. Inadvertent aircraft crashes

In addition to these, the second criteria also points to:

4. Loss-of-offsite power events caused by severe weather (namely hurricanes), with concurrent loss of emergency onsite AC power (either due to the same event or due to coincidental hardware failures), and
5. Lack of accessibility caused by a reactor accident that has released radioactive material outside of primary containment (or an accident involving the other spent fuel pool)

Note that sabotage events have been excluded from the scope of this study; however, many of the insights obtained will also be applicable to sabotage events if they resulted in an analogous damage state.

Items #1 (seismically-induced station blackout), #2 (cask drops), and #4 (severe weather LOOPs) have been considered in most other SFP studies, and are discussed further below. For item #3, past studies (namely NUREG-1738 [NRC, 2001]) have concluded that the risk of this initiator is bounded by other initiators for both PWRs and BWRs, based on quantitative estimates of likelihood and expected damage (see Section 3.5.2 of that study). Item #5 (effects of a concurrent reactor accident) generally have not been studied in prior efforts. They are partially considered here, but not rigorously (see Section 9.2 of this study for more information).

Past studies have had different conclusions about the relative contribution to risk/consequences from the various initiating events considered. Table 1 below summarizes fuel uncovery frequencies from NUREG-1353 and NUREG-1738. INEL-96/0334 also contained accident frequencies, but they are not directly comparable in that (i) the initiators are grouped in a different manner and (ii) the results focus on the time to pool boiling rather than a fuel damage precursor or surrogate. For both NUREG-1738 and NUREG-1353 seismic events were the largest contributor to the frequency of fuel uncovery.

Table 1: Frequency of SFP Fuel Uncovery (/yr)

Initiating Event Class	**NUREG-1353** (1989) [BWR, best-estimate[1]]	**NUREG-1738** (2001)
Seismic events	$7 \cdot 10^{-6}$	$2 \cdot 10^{-6}$ (LLNL) $2 \cdot 10^{-7}$ (EPRI)
Cask / heavy load drop	$3 \cdot 10^{-8}$	$2 \cdot 10^{-7}$
LOOP – severe weather	-	$1 \cdot 10^{-7}$
LOOP – other	-	$3 \cdot 10^{-8}$
Internal fire	-	$2 \cdot 10^{-8}$
Loss of pool cooling	$8 \cdot 10^{-8}$	$1 \cdot 10^{-8}$
Loss of coolant inventory	$1 \cdot 10^{-8}$	$3 \cdot 10^{-9}$
Inadvertent aircraft impacts	$6 \cdot 10^{-8}$	$3 \cdot 10^{-9}$
Missiles – general	$1 \cdot 10^{-8}$	-
Missiles - tornado		$< 1 \cdot 10^{-9}$
Pneumatic seal failures	$3 \cdot 10^{-8}$	-

[1] These numbers have not been multiplied by the stated conditional probability of having a Zirconium fire of 0.25.

For this reason, a seismic event was judged to be the logical focus of this limited-scope consequences assessment. Based on a review of the seismic hazard for the particular site studied, and consideration of seismic hazard binning from contemporary seismic PRA methodologies, a specific range of ground motions was chosen for this study (see Section 3). This range of ground motions represents a good compromise between more likely events that

would not be expected to lead to any consequences versus less likely events that would lead to greater consequences (recall that risk is the product of the likelihood times the consequences).

1.6. Operating Cycle Phase Approach

During a given operating cycle, the spent fuel pool:

- will change configurations from being an isolated pool to being hydraulically connected to the reactor vessel (and back again)—these will be referred to as pool-reactor configurations to distinguish from the different spent fuel loading configurations;
- may have spent fuel offloaded temporarily from the reactor;
- will have spent fuel offloaded permanently from the reactor;
- will likely have spent fuel moved around within the SFP (as part of complying with regulatory requirements related to heat distribution, criticality, and boraflex coupon sampling);
- may have older spent fuel offloaded in to casks;
- will experience changes in the peak assembly decay power (of interest for draindown events and spray mitigation) due to the above as well as radioactive decay; and
- will experience changes in the total decay power of all assemblies (of interest for pool heatup/boiling and makeup mitigation) because of the above as well as radioactive decay.

To faithfully represent these temporally changing conditions, one would need to break up the operating cycle into numerous small periods of time or Operating Cycle Phases (OCPs). However, the number of OCPs considered is nearly a linear multiplier on the amount of resources needed because each period of time requires its own set of accident progression and consequence analyses. Past studies have taken the approach of selecting specific points in time of interest, and comparing results for those specific times. This study takes a similar approach, but places more emphasis on the definition of these times as quasi-steady representations of the portion of the operating cycle that they represent. This allows for more faithful representation of the annualized frequencies of offsite consequences. The specific selection of these phases is described further in Section 5.2.

1.7. Overview of Past Studies

A number of past studies have been performed to look at various aspects of spent fuel and spent fuel pool safety, security, and/or risk. The major regulatory activities are shown pictorially in Figure 2. A more comprehensive chronicling of these past studies, as well as other aspects of general interest pertinent to the current effort, are briefly described in the ensuing text.

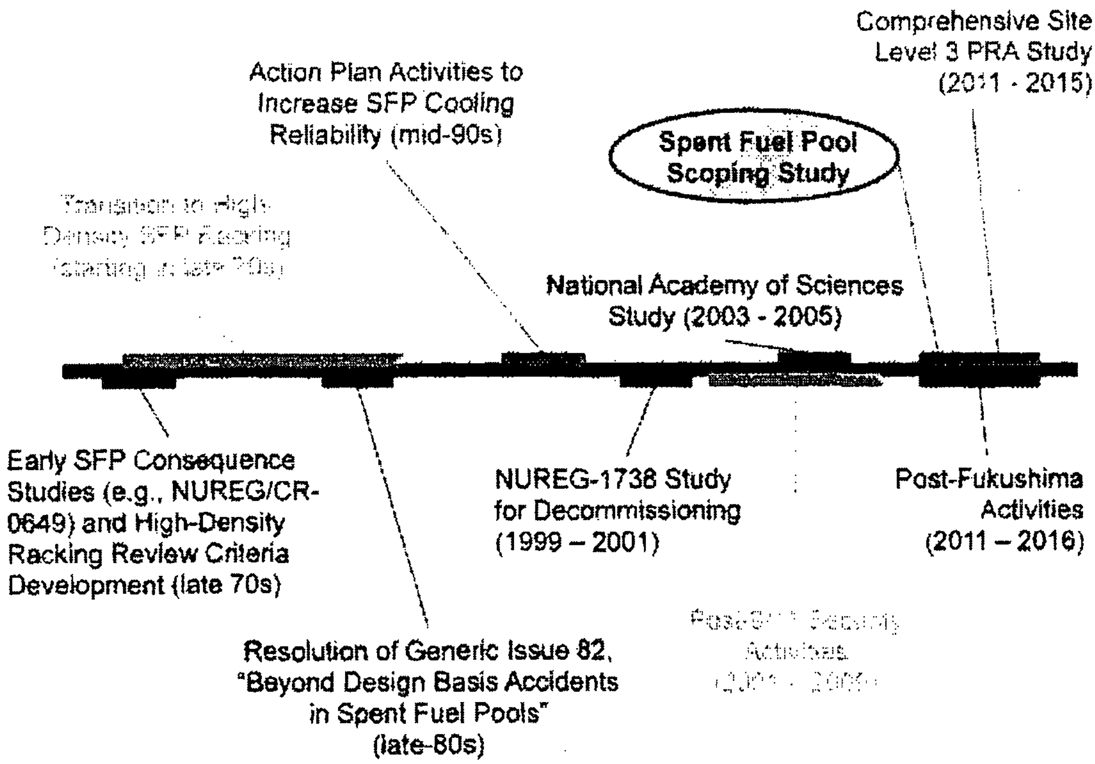


Figure 2: Graphical Overview of Significant SFP-Related Activities

In 1979, the NRC issued NUREG/CR-0649, "Spent Fuel Heatup Following Loss of Water During Storage," which provided an analysis of spent fuel heatup following a hypothetical accident involving drainage of the storage pool [NRC, 1979]. The report included analysis to assess the effect of decay time, fuel element design, storage rack design, packing density, room ventilation, drainage level, and other variables on the heatup characteristics of spent fuel stored in a spent fuel pool and to predict the conditions under which clad failure would occur. The report concluded that the likelihood of clad failure due to rupture or melting following a complete drainage is extremely dependent on the storage configuration and the spent fuel decay period, and that the minimum prerequisite decay time to preclude clad failures may vary from less than 10 days for some storage configurations to several years for others. The potential for reducing this critical decay time either by making reasonable design modifications or by providing effective emergency countermeasures was found to be significant. Note that this study considered both low-density racking and mitigative accessibility.

In the late 1980s, work related to Generic Issue 82 culminated in the publishing of two related reports: NUREG/CR-5281, "Value/Impact Analysis of Accident Preventive and Mitigative Options for Spent Fuel Pools," and NUREG-1353, "Regulatory Analysis for the Resolution of

Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools." [NRC, 1989a] and [NRC, 1989b] The former study investigated options including limited low-density re-racking of spent fuel, installation of water sprays above the spent fuel pool, and the installation of redundant cooling and/or makeup systems. The results of these studies indicated that the measures were, in general, not likely to be cost effective. The reason for this is due to both the low likelihood of a spent fuel pool accident that could result in a significant radiological release and the high cost of proposed modifications. The report goes on to conclude that these insights are largely contingent upon compliance with guidelines developed for licensees to assure the safe handling of heavy loads in the vicinity of spent fuel pools, thus reducing the likelihood of the structural failure of the pool and rapid loss of water inventory due to a cask drop event.

The latter report (NUREG-1353) also includes spent fuel pool analysis, and concludes that if the decay heat level is high enough to heat the fuel rod cladding to about 900 degrees C the oxidation becomes self-sustaining, resulting in a Zircaloy cladding fire. The conditional probability of a Zircaloy cladding fire given a complete loss of water was found to be 1.0 for PWRs and 0.25 for BWRs in high-density configurations. The conditional probability of a Zircaloy cladding fire given a complete loss of water in low density storage racks is estimated to be at least a factor of five less than for the high density configurations. The report goes on to state that although most of the spent fuel pool risk is derived from beyond design basis earthquakes, this risk is no greater than the risk from core damage accidents due to seismic events beyond the safe-shutdown earthquake. Therefore, reducing the risk from spent fuel pools due to events beyond the safe-shutdown earthquake would still leave at least a comparable risk due to core damage accidents. As a result of this conclusion, the results justified the decision that no regulatory action was needed.

In 1996, an NRC-sponsored Idaho National Laboratories study entitled, "Loss of Spent Fuel Pool Cooling PRA: Model and Results," was issued. [INL, 1996] This study considered a dual-unit plant and the following initiators:

- Loss of spent fuel pool cooling
- Loss of offsite power
- Loss of spent fuel pool inventory (did not include heavy load drops)
- Loss of primary (reactor) coolant
- Seismic events

The results of this study indicated that for plant studied, the annual probability of spent fuel pool boiling is $5x10^{-5}$ and the annual probability of flooding associated with spent fuel pool accidents is $1x10^{-3}$. Qualitative arguments are provided to show that the likelihood of core damage due to spent fuel pool boiling accidents is low for most U.S. commercial nuclear power plants. It is also shown that, depending on the design characteristics of a given plant, the likelihood of either: a) core damage due to spent fuel pool-associated flooding, or b) spent fuel damage due to pool dryout, may not be negligible.

The next year, two additional reports were issued: NUREG-1275, Volume 12, "Operating Experience Feedback Report: Assessment of Spent Fuel Cooling," and "Follow-up Activities on the Spent Fuel Pool Action Plan." [NRC, 1997a] and [NRC, 1997b] The former report concluded that the typical plant may need improvements in SFP instrumentation, operator procedures and training, and configuration control. [Note that this is the conclusion stated in the report, and has not been placed in the regulatory context of balance-of-plant activities since the issuance of that report.] The staff determined that loss of SFP coolant inventory greater than 1 foot occurred at

a rate of about 1 event per 100 reactor years. Loss of SFP cooling with a temperature increase greater than 20 degrees F occurred at a rate of approximately 3 events per 1,000 reactor years. The primary cause of these events was found to be human error. The report also concluded that utilities' efforts to reduce outage duration resulted in full core off-loads occurring earlier in outages. This increased fuel pool heat load was felt to be important because it reduces the time available to recover from a loss of spent fuel pool cooling event early in the outage.

In the latter report (the Spent Fuel Pool Action Plan), the staff performed probabilistic screening analyses and found that, in most cases, event frequencies for sequences associated with identified spent fuel pool design issues were sufficiently low that further analyses were not warranted. In one instance where the probabilistic screening criteria were met, the staff performed a deterministic evaluation of the issue using plant-specific information and found that safety enhancements were not warranted.

A couple of years later, the agency conducted a spent fuel pool risk study for decommissioning plants to look at the relaxation of emergency preparedness requirements, and in 2001 the final version was issued as NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants." [NRC, 2001] The results of the study indicated that the risk at spent fuel pools is low and well within the Commission's Quantitative Health Objectives (QHOs). The risk was found to be low because of the very low likelihood of a zirconium fire, even though the consequences from a zirconium fire could be serious. The report found that the event sequences important to risk at decommissioning plants were limited to large earthquakes and cask drop events. This report represented a significant undertaking, and remains one of the prominent studies cited in NRC decisionmaking on spent fuel pools. However, there are some important conservatisms associated with this study that need to be considered if it is applied outside of its intended context (emergency preparedness for decommissioning reactors). These conservatisms include: (i) the use of assumed and often bounding configurations, (ii) simplified treatment of the thermal-hydraulic response, (iii) conservative assumptions regarding structural response, and (iv) emergency preparedness response representative of a decommissioned site.

On the heels of the aforementioned study, the agency also released NUREG/CR-6441 in 2002, entitled, "Analysis of Spent Fuel Heatup Following Loss of Water in a Spent Fuel Pool: A Users' Manual for the Computer Code SHARP." [NRC, 2002] This document included an analysis of spent fuel heatup, using "representative" design parameters and fuel loading assumptions. Sensitivity calculations were also performed to study the effect of fuel burnup, building ventilation rate, baseplate hole size, partial filling of the racks, and the amount of available space to the edge of the pool. The spent fuel heatup was found to be strongly affected by the total decay heat production in the pool, the availability of open spaces for air flows, and the building ventilation rate. Note that the spent fuel pool analyses performed by the NRC after this time did not rely on this computer code. Rather, they relied on the use of the MELCOR computer code, with supporting analysis using the COBRA-SFS, FLOW3D and Fluent codes, along with confirmatory experiments at Sandia National Laboratories.

In response to the events of September 11, 2001, the NRC undertook studies (referred to hereafter as security assessments) of spent fuel storage in pools and casks. While this work was underway, Robert Alvarez et al. published the paper, "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States," 2003 (hereafter referred to as the 2003 Alvarez paper). [Alvarez, 2003] In response, the NRC issued a review of the paper (also in 2003) which concluded that the assessment performed of possible spent fuel pool accidents stemming from potential terrorist attacks in the 2003 Alvarez paper did not address such events in a realistic

manner. [NRC, 2003a] The NRC response went on to state that, in many cases, the authors of the 2003 Alvarez paper rely on studies that made overly conservative assumptions or were based on simplified and very conservative models. The NRC concluded that the fundamental recommendation of the 2003 Alvarez paper, namely that all spent fuel more than five years old be placed in dry casks through an expedited 10-year program costing many billions of dollars, is not justified.

Continued discussions on the issue of spent fuel pool safety and security led to a 2004 - 2005 National Academy of Sciences study, documented in "Safety and Security of Commercial Spent Nuclear Fuel Storage," the National Academies Press, 2006. [NAP, 2006] This study was Congressionally mandated (e.g., see [Congress, 2005]). The National Academies Committee was briefed on numerous occasions by NRC staff regarding past and ongoing studies related to the subject topic. The study resulted in a classified report and the aforementioned publicly available report. The publicly available report documented numerous findings and recommendations, many of which were addressed as part of the NRC's continued activities in this area (e.g., the "Phase 2" site-specific assessments). The NRC's initial response to the study was documented in a letter from the NRC Chairman (Nils Diaz) to Senator Peter Domenici in March 2005. [NRC, 2005]

In parallel to the National Academies study, the NRC continued performing the aforementioned security assessments, which were completed in the 2006 – 2008 timeframe. While the results of these studies are non-publicly available due to their nature, the conclusions of the studies were integrated in to the NRC's regulatory licensing and oversight processes (e.g., the "Power Reactor Security Rulemaking" codified in 10 *Code of Federal Regulations* Part 50, Subpart 54(hh)). Activities related to the development of new security-related requirements were later documented in a memorandum to the NRC Commission entitled, "Documentation of Evolution of Security Requirements at Commercial Nuclear Power Plants With Respect to Mitigation Measures for Large Fires and Explosions," dated February 4, 2010. [NRC, 2010]

Also in parallel to the above activities, the agency conducted a pilot PRA for dry cask storage documented in NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System At a Nuclear Power Plant," issued in 2007. [NRC, 2007a] The results of the report's analysis indicate that dry cask storage risk is solely from latent cancer fatalities, and no prompt fatalities are expected. Dry cask storage risk was found to be dominated by accident sequences occurring in three stages of the handling phase. These involved the drop of the transfer cask through the equipment hatch (termed Stage 18) and drops of the multi-purpose canister (MPC) into the storage overpack (Stages 20 and 21). The aggregated risk values were quite low. The estimated aggregate risk was an individual probability of a latent cancer fatality of $1.8 \cdot 10^{-12}$ during the first year of service, and $3.2 \cdot 10^{-14}$ per year during subsequent years of storage. Note that when insufficient information was available, "conservative bounding assumptions or estimates" were used. Other limitations of the study included no consideration of uncertainty and conservative assumptions about the translation of failure modes to hole sizes.

Two other documents of regulatory interest were issued in 2008 and 2009, respectively. The first was the denial of two petitions for rulemaking (PRMs) as documented in SECY-08-0036 and the associated Staff Requirements Memorandum. [NRC, 2008a] These documents describe the NRC's denial of PRMs filed by the Attorney General of the Commonwealth of Massachusetts and the Attorney General for the State of California, which presented nearly identical issues and requests for rulemaking concerning the environmental impacts of high-density storage of spent nuclear fuel in spent fuel pools.

The second document is the issuance in 2009 of the Draft Report for Comment of Revision 1 to the NRC's *Generic Environmental Impact Statement* (GEIS) *on License Renewal* (NUREG-1437, Revision 1, Draft Report for Comment). [NRC, 2009a] This document reevaluated SFP environmental considerations related to spent fuel pools by considering information developed since the original License Renewal GEIS was issued in 1996. [NRC, 1996] The update concluded that the environmental impacts from accidents at spent fuel pools (as quantified in NUREG-1738) can be comparable to those from reactor accidents at full power (as estimated in NUREG-1150, "Severe Accident Risk: An Assessment for Five U.S. Nuclear Power Plants," 1990) [NRC, 1990]. The updated GEIS goes on to state that subsequent analyses performed, and mitigative measures employed, since 2001 have further lowered the risk of spent fuel pool accidents; and then, even the conservative estimates from NUREG-1738 are much less than the impacts from full power reactor accidents as estimated in the original 1996 GEIS. As a result of these considerations, the update concludes that the environmental impacts stated in the 1996 GEIS bound the impact from SFP accidents.

Finally, in July 2011, the NRC issued, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near Term Task Force Review of Insights from the Fukushima Dai-ichi Accident." [NRC, 2011a] In the section of the report on spent fuel pools, the task force concluded that "clear and coherent requirements to ensure that the plant staff can understand the condition of the spent fuel pool and its water inventory and coolability and to provide reliable, diverse, and simple means to cool the spent fuel pool under various circumstances are essential to maintaining defense-in-depth." The task force goes on to recommend Orders addressing: (i) spent fuel pool instrumentation, (ii) safety-related AC power for spent fuel pool makeup, (iii) technical specification revision regarding onsite AC power for spent fuel pool makeup and instrumentation, and (iv) a seismically-qualified spray capability. The task force also recommended rulemaking or licensing actions (or both) to require the above actions.

The US nuclear industry has also undertaken various studies related to spent fuel storage and transportation, some of which are publicly available and some of which are referenced in related publicly available reports. Examples include:

- EPRI TR-1003011, "Dry Cask Storage Probabilistic Risk Assessment Scoping Study," 2002 (non-public)
- EPRI TR-1009691, "Probabilistic Risk Assessment (PRA) of Bolted Storage Casks: Updated Quantification and Analysis Report," 2004 (non-public)
- EPRI TR-1021049, "Impacts Associated With Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling," 2010. (public)

The latter report is of particular interest for the present effort. The report assesses the cost and risk impacts (from a worker dose perspective) associated with transfer of spent nuclear fuel from spent fuel pools to dry storage after five years of cooling. The report concludes that expedited fuel movement would result in an increase cost to the U.S. nuclear industry of $3.6 billion, with the increase primarily related to the additional capital costs for new casks and construction costs for the dry storage facilities. The report goes on to conclude that early movement of spent fuel into dry storage would have "significant radiological impacts." These impacts are stated in terms of worker radiation exposure, and are estimated to be 507 person-rem over 60 years due to the additional handling of spent fuel. With respect to spent fuel pool accidents, the report estimates that an additional 711 dry storage packages would have to be handled compared to the case without expedited fuel movement, thus increasing the risks associated with cask movement. [NAC, 2011] makes similar arguments with respect to the impacts of expediting fuel

movement. Neither study attempts to calculate offsite consequences associated with postulated spent fuel pool accidents.

Regarding this last point, the EPRI and NAC studies make points related to spent fuel pool safety that warrant additional context, particularly if compared to the present study. To this end, it is helpful to start by looking at the spent fuel pool inventories used in this study versus industry averages (as presented in the NAC white paper).

Table 2: Comparison of Fuel Age and Heat Load Against Industry Averages

	Mass as a % of all fuel		Heat generation as a % of all fuel	
Time since discharge (yrs)	Industry average	This study	Industry average	This study
< 5	22%	18%	58%	84%
5 – 9	22%	27%	22%	9%
10 – 14	16%	18%	9%	3%
15 – 19	15%	19%	6%	2%
20 – 24	10%	17%	3%	2%
25 – 29	6%	1%	1%	<1%
30 – 34	4%	-	<1%	-
Remainder	4%	-	<1%	-

The NAC white paper, and to a lesser extent the aforementioned 2010 EPRI study, make the case that heat load distributions like the ones in Table 2 support the notion that moving fuel older than 5 years has only modest effects on the overall spent fuel pool heat load (and thereby the cooling requirements and mitigative time available for beyond design-basis spent fuel pool accidents). The values in the table for the site studied here highlight the caution that accompanies treating the heat load as a point estimate.

The NAC white paper goes on to make the case that if increased fuel movement from pools to dry cask storage is required, it should be for fuel older than 8 to 10 years, because current regulatory requirements, infrastructure (e.g., cranes, cask designs), and industry practices can accommodate this imposition with less impact than a 5 year cutoff. Of note is the fact that both the NAC and EPRI studies ignore the effect of the older fuel on the overall spent fuel pool radionuclide inventory, which for radionuclides like Cesium-137 (30 year half-life) is much more constant across fuel age than is the heat generation rate. However, consideration of this very important aspect would arguably further support the benefit of moving fuel older than 8 to 10 years because the older fuel would have comparable contributions to the overall radionuclide inventory for long-lived isotopes. The only caution here is that the 5 to 10 year old fuel is likely to be higher burnup than 10+ year old fuel, so the small increment in the half-life is not the only factor affecting their relative contributions.

1.8. Potential Follow-On Work and Related Activities

As noted earlier, there are a number of follow-on activities that could be undertaken to further flesh out the issue at hand. For example, one could focus on other aspects of the comprehensive risk (e.g., cask risk), general limitations of this study (e.g., repeating the study for a pressurized water reactor SFP), or specific limitations of this study (e.g., detailed analysis of the effects of the seismic event on the integrated neutron poison material in the SFP racks). Which work will and will not be pursued is a decision that will be made after the current study

has been reviewed, and in the context of the ongoing regulatory actions related to the 2011 Fukushima accident (and in particular a "Tier 3" item from these activities specific to expedited fuel movement). It is important to recognize that there are several ongoing activities that have a peripheral relationship to this study. These include, but are not limited to:

- Order EA-12-049 related to mitigation strategies for beyond-design-basis external events, dated March 12, 2012, and related correction dated March 13, 2012
- Order EA-12-051 related to reliable spent fuel pool instrumentation, dated March 12, 2012, and related correction dated March 13, 2012
- An ongoing rulemaking related to security requirements for ISFSIs.
- Re-evaluation of the role of defense-in-depth in regulatory decision-making.
- Re-consideration of the use of land contamination and economic consequences in the context of regulatory decision-making.

Having said this, there is already a related activity underway to perform a site Level 3 probabilistic risk assessment for an operating pressurized water reactor site, including consideration of both wet and dry storage. The details of this study are provided in SECY-11-0089 and the associated Staff Requirements Memorandum. While the study does not specifically focus on expedited fuel movement, it will inherently inform the issue of comparative risk between wet and dry storage.

1.9. Layout of Remainder of This Report

The remaining sections of this report provide:

- Major assumptions and limitations
- Seismic hazard characterization
- Structural analysis methods and results
- Scenario delineation and probabilistic considerations
- Accident progression analysis methods and results
- Offsite consequence analysis methods and results
- Planned future consideration of uncertainty
- Other issues warranting additional discussion

Finally, Appendix A provides responses to Frequently Asked Questions (FAQs).

2. MAJOR ASSUMPTIONS

Assumptions made during the conduct of this study are documented throughout this report. For reader convenience, major assumptions are catalogued in Table 3.

Table 3: Major Assumptions

Topical Area	Major Assumption	Comment
Overall Approach	A BWR Mark I with a non-shared SFP is studied.	This plant was chosen for a mix of reasons, including availability of computer models and a perception of greater external stakeholder interest in elevated (relative to grade) SFPs. Its selection does not denote a belief that this type of design is more vulnerable.
	The well-beyond design basis earthquake is assumed to occur. This is a very unlikely event.	The earthquake studied has an estimated frequency of occurrence of one time in 61,000 years.
	Multi-unit / concurrent reactor accidents are not, in general, considered.	Specifically, the reactor (and its decay heat) are treated during the outage until the level in the reactor well / SFP drops to below the bottom of the fuel transfer canal. Beyond that point, and in all portions of the post-outage scenarios, the reactor is not considered as a source of steam, fission products or hydrogen. The intent is to scope some of the associated limitations via sensitivity studies. Recall that unmitigated scenarios are being treated, which in part address the case where a reactor of other SFP event prevents operator action. Also consider that the current analysis focuses on the delta between high-density and low-density loading, either of which would be affected by a concurrent reactor or SFP accident.
Seismic Hazard Characterization		
Structural and Related Initial Damage State Characterization	No debris generated by the seismic event enters the SFP.	Based on the expected structural response of the building, overhead crane, etc. there is no expectation that heavy debris that would damage the pool and fuel will be generated.
	The seals of the refueling gate do not fail.	Finite element analysis does not predict large deformations in this area that would suggest such an event is likely. Details of the gates provided by the licensee show that there are two gates with a gap in between and that each gate has mechanical seals to prevent leakage. These seals are kept under pressure by passive mechanical means (do not depend on either AC or DC power availability) that are unlikely to fail under the earthquake.

Topical Area	Major Assumption	Comment
Scenario Delineation and Probabilistic Treatment	**Offloading of older fuel in to casks** (as part of the normal fuel management practices as opposed to an expedited fuel movement program) is *not explicitly treated.*	See Section 5.2.
	A **full core offload** is not treated (except as discussed to the right) as either part of the routine refueling or in the context of an emergent need to defuel the reactor later in the operating cycle (e.g., due to a forced shutdown that requires accessing the lower internals of the reactor vessel).	In reality, the full core's decay heat is considered during the outage, in that the reactor and SFP are hydraulically connected, and all fuel contributing to pool heatup is considered until the point that water level drops below the fuel transfer gate (and the reactor well and SFP become hydraulically disconnected). That being said, the radioactive material from the fuel remaining in the reactor is not considered, since the simulation focuses only on the SFP once the reactor well and SFP have become hydraulically disconnected. A sensitivity is proposed to address this effect. The rationale for choosing a "core shuffle" rather than a full core offload is because the former is the typical case for BWRs. Emergent core offloads later in the operating cycle are not typical, and thus are not treated.
	New fuel temporarily stored in the spent fuel pool is not treated.	See Section 5.2.
	For the low-density loading situation, the **high-density racking** will be used as opposed to low-density racking.	Re-racking the pool would represent a significant expense, along with additional worker dose, and was not felt to be the likely regulatory approach taken based on consultation with the Office of Nuclear Reactor Regulation. Much of the benefit of low-density racking is achieved by the implementation of a favorable fuel configurations (1x4). Additionally, to get the full benefit of low-density racking, BWR fuel would likely need to have the channel boxes removed.

Topical Area	Major Assumption	Comment
	An assembly in the **lifted position** (i.e., in the process of being moved) at the time of the seismic event is not treated. The current tools do not allow for explicit treatment of this situation. Such a situation could lead to accessibility issues (which are already treated via the scenarios without 50.54(hh)(2) equipment), but could also lead to a small earlier release for some situations. This possibility may be investigated further via a sensitivity study.	
	Mode of mitigation deployment.	For OCP #1/#2 with the "moderate" leakage condition, makeup is deployed. Other assumptions about mitigation deployment could result in the deployment of sprays instead. This difference, which shows the value of addition instrumentation, may be pursued via a sensitivity study.
Accident Progression Analysis	Best-estimate **Ruthenium release rates** calculated by the MELCOR code are used. These release rates are most similar to the low Ruthenium release case from NUREG-1738.	This is the best-estimate for actual releases, based on the current state-of-knowledge in this area. Past studies for which this was a concern (namely NUREG-1738) used assumed source terms from previous studies, rather than mechanistic and integrated modeling.
	Radionuclide releases occur only if they commence prior to 72 hours. Otherwise, the project assumes the scenario results in no offsite consequences.	

Topical Area	Major Assumption	Comment
	Debris entering the pool as a result of any modeled hydrogen deflagration / detonation is not considered.	Such debris could be generated, and could fall in to the pool. However, the occurrence of a hydrogen deflagration/detonation in this study denotes that the fuel has already become uncovered and is undergoing a fission product release. Thus, debris would primarily serve to inhibit longer-term recovery actions not considered in this study. The occurrence of a hydrogen deflagration/detonation from a concurrent reactor accident has the potential to generate debris prior to SFP uncovery, but this situation is inherently tied to the lack of comprehensive treatment of multi-unit aspects in this study.
	Aerosol resuspension inside the reactor building, such as from hydrogen, deflagration will not be significant.	Hydrogen burns in the spent fuel pool bay are predicted to occur about the time of fuel gap release, and well before significant amounts of radioactive aerosols may settle on the floor.
	The effects of **molten core concrete interaction** are not considered	Heat transfer from the debris to the pool floor is modeled in MELCOR, and the code models fission product release from hot debris. In some cases, the debris temperature remains above typical concrete ablation temperature (~1500 K). These cases involve large scale debris relocation and large releases of volatile fission products.
	The effective **decontamination factor** (DF) of the reactor building, as measured at the point in time at the end of the MELCOR calculation, can be used to reasonably estimate a cumulative release.	The use of an effective DF is based on a new methodology for spent fuel pools in an effort to account for a spatial distribution of the inventory, and to more accurately account for the magnitude of the release based on the radionuclide, not just the chemical group. Because the DF is not constant, some of the effects of using a constant DF can be identified and accounted for. Therefore special attention is needed to reduce this error by hand, and in order to allow the offsite consequence code to process the source term.
Offsite Consequence Analysis	Calculated results are from **atmospheric-type releases** only	Atmospheric releases are the primary concern of the project. See section 7.1 for more information on this assumption.
	Distance truncation (from point of release)	Health effects are reported as latent cancer fatality risk (and early fatality risk), and as a function of distance. The reported LCF risk will include all distances that have doses above the PAGs. See section 7.1 for more information on this assumption.

Topical Area	Major Assumption	Comment
	The effect of **low dose radiation** on latent cancer fatalities is uncertain, and therefore a range of dose response model truncations will be reported.	See section 7.1 for more information on this assumption.
	The public will behave in an orderly fashion during a severe accident, and can be represented by cohorts.	
	Decontamination will occur only if it will eventually allow for the return of land to habitability and if it is economic to do so.	A long-term cleanup policy for severe accidents does not currently exist, although rulemaking for such guidance is currently in process. In addition, such guidance could likely allow for the development of localized cleanup goals after an accident, as to allow for a number of factors that include socio-political, technical, and economic considerations. Given that such a policy for long-term cleanup does not currently exist (and because a developed policy may not contain explicit cleanup goals), the project instead uses dose levels associated with habitability as the point in deciding when land is to be decontaminated. This is consistent with previous studies.
Other Issues	This study represents a limited-scope **consequence study** as opposed to a probabilistic risk assessment.	This approach focuses resources on a particular scenario of interest and places greater emphasis on modeling fidelity for that scenario, but also limits the potential end-uses of the study. See Section 9.1 for more information on this assumption.
	Multi-unit aspects are only considered for certain aspects of the study.	See Section 9.2 for more information on this assumption.
	Other considerations associated with expedited fuel movement.	See Section **Error! Reference source not found.** for more information on this assumption.
	Inadvertent criticality events are not considered.	See Section 9.5 for more information on this assumption.

3. SEISMIC HAZARD CHARACTERIZATION

3.1. Basis for Probabilistic Estimates

The primary sources of information for seismic hazard estimates at nuclear power plant sites have been information developed by: (i) the NRC/Lawrence Livermore National Laboratories (LLNL) (Bernreuter et al., 1989, Sobel, 1994) herein called the LLNL model; (ii) the Electric Power Research Institute (Toro et al, 1989), herein called the EPRI model; and (iii) the United States Geological Survey (USGS) in the mid-2000s (Peterson et al, 2008), herein called the USGS 2008 model. Both the LLNL and EPRI models were utilized in the implementation of the Individual Plant Evaluation for External Events (IPEEE) program (NUREG 1742). The USGS 2008 has been utilized in the National Seismic Hazard Mapping Project and by the NRC for the seismic hazards estimates used in screening level assessments for Generic Issue 199 (NRC, 2012a).

The USGS 2008 information is being further developed and updated by a group of stakeholders, including the NRC, in a collaborative study which includes (a) the seismic source zone characterization, and (b) the ground motion attenuation models. In addition, the NRC is developing independent methods and computer codes, which will be publicly available when completed, to combine (a) and (b). Although part (a) of this updating effort has been completed (NRC 2012b), parts (b) and (c) are still ongoing. Therefore, the SFP Scoping Study used the earlier USGS information instead of the ongoing update program

Comparisons of hazard estimates for the reference site, Peach Bottom, a rock site, obtained with those three information sources are graphically shown in Figure 3 (PGA) and in Figure 4 (1, 5 and 10 Hz spectral acceleration), which supports the following observations:

- For the PGA, USGS 2008 predicts higher annual probability of occurrence for high-level, low probability events, specifically for events with PGAs greater than about 0.35 g.
- For moderate PGAs, from about 0.1 g to 0.35g, LLNL is higher than USGS 2008. Above about 0.35 g, the lower probability events, USGS 2008 is higher than LLNL until both hazard estimates differ by factors of about 2.5 at 0.75 g and about 3 at 1.0 g.
- The EPRI hazard estimates are lower than those from the USGS 2008 for all PGAs. Specifically, hazard estimates based on the recent USGS update are about 2 times greater at 0.2 g with the difference increasing to about 10 times at 1.0 g.
- Thus, in terms of PGA, the seismic hazard estimates used for this study are about 2.5 times greater than LLNL IPEEE estimates and about 6 times greater than EPRI IPEEE estimates.
- The USGS and LLNL curves are comparable for each representation, with USGS 2008 sometimes being higher (higher annual probability of occurrence) and LLNL sometimes being higher.
- Generally, the 10 Hz curve is the highest, followed by the 5 Hz curve, followed by the PGA curve, followed by the 1 Hz curve. The notable exception to the above is the fact that the 5 Hz LLNL curve is higher than the 10 Hz curve.

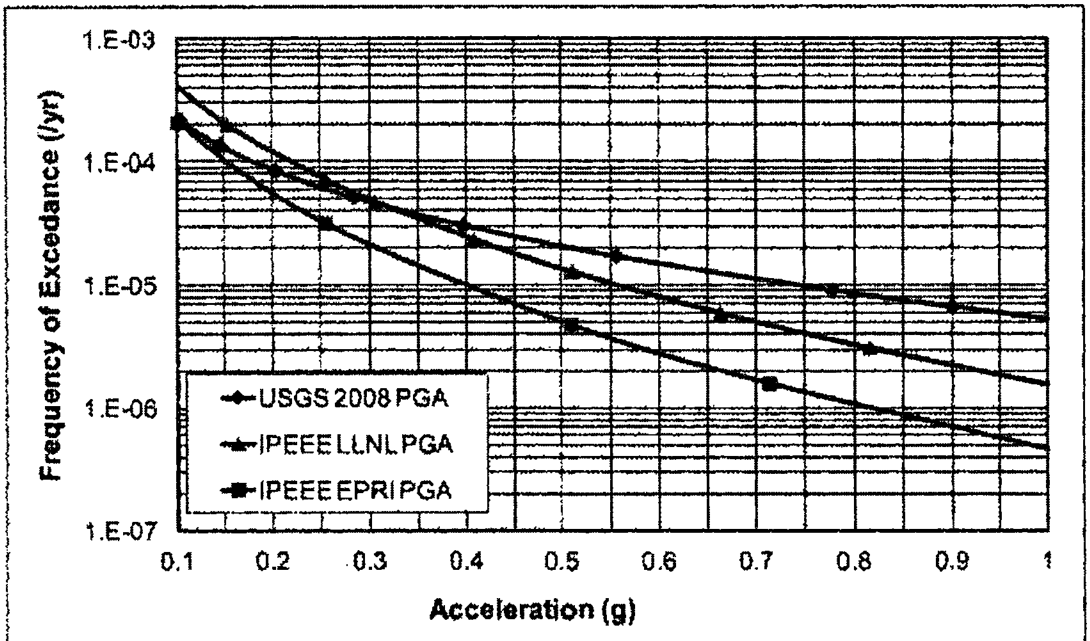


Figure 3: Comparison of PGA exceedance frequencies at Peach Bottom

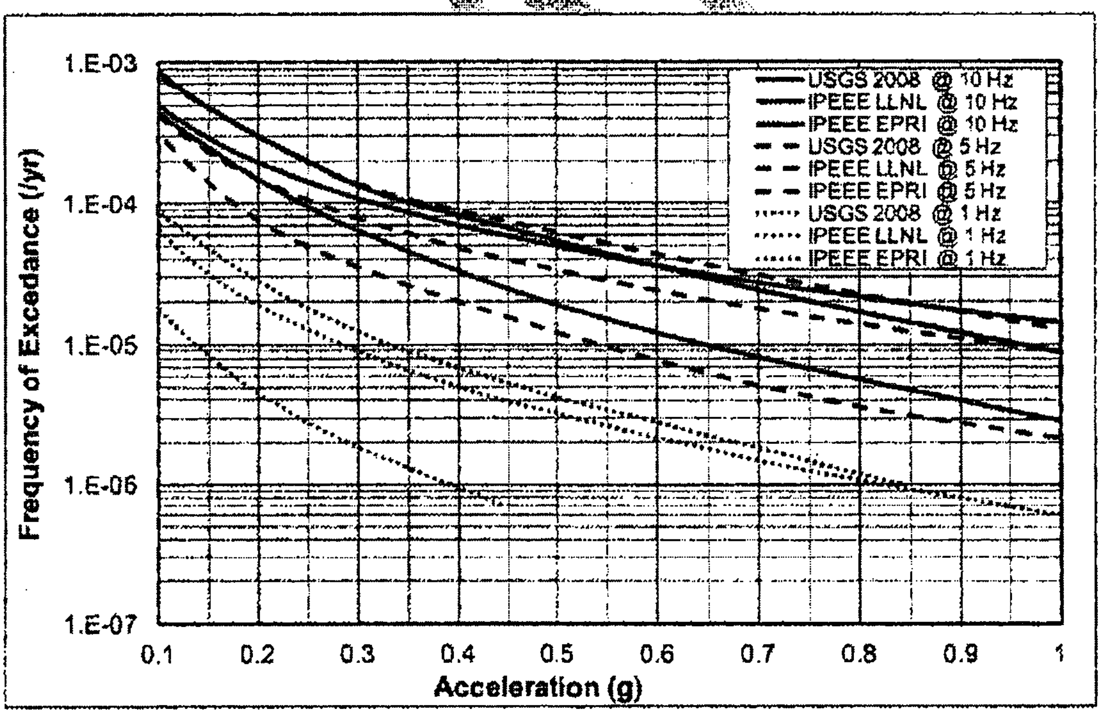


Figure 4: Comparison of spectral exceedance frequencies at Peach Bottom

A comparison of the annual frequency of exceeding a given PGA for all Mark I sites (
Figure 5) shows that the reference site (Peach Bottom) falls close to the upper end of the group in terms of hazard estimates. When comparing the annual frequency of exceeding a given 1 Hz spectral acceleration (Figure 6), the reference site is in the upper half of the group. (

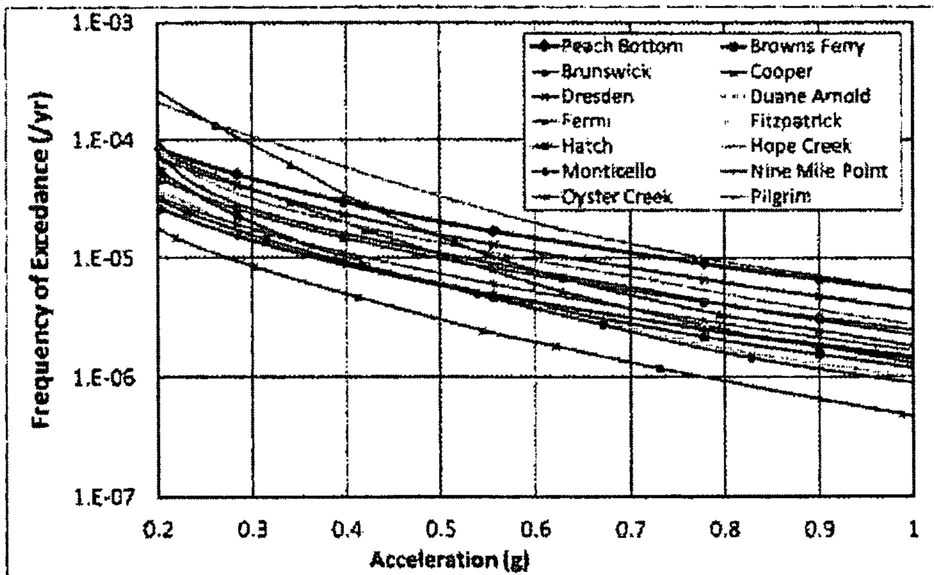


Figure 5: Comparison of annual PGA exceedance frequencies for U.S. Mark I reactors s (USGS 2008)

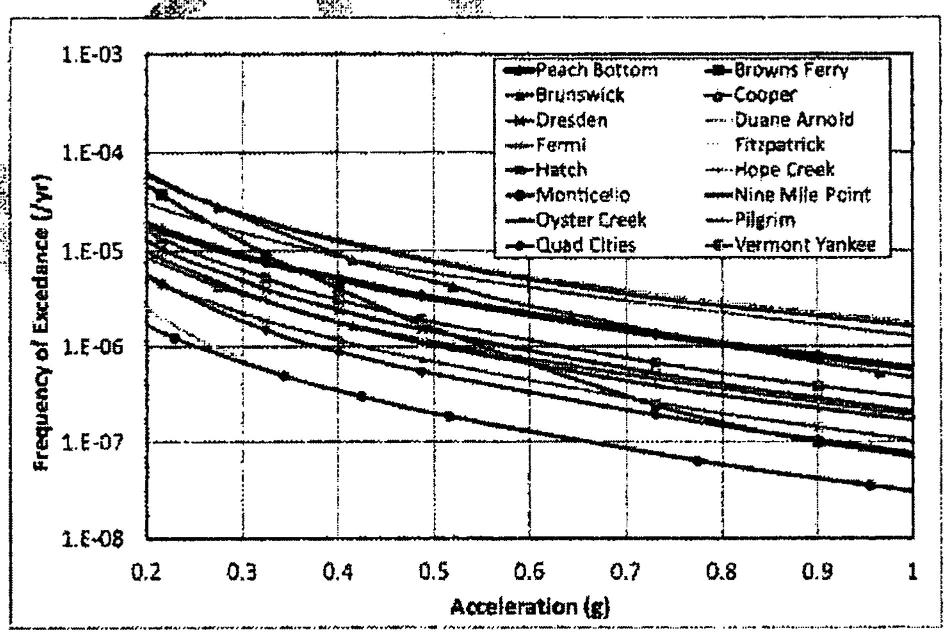


Figure 6: Comparison of annual 1 exceedance frequencies for 1 Hz spectral accelerations for U.S. Mark I reactors (USGS 2008)

3.2. Characterization of the Event Likelihood

Hazard exceedance frequencies can be translated into initiating event frequencies by partitioning the PGA range into a number of discrete categories (bins) defined in terms of PGA intervals. These bins define a discrete number of seismic event scenarios with increasing intensity (PGA). The NRC handbook entitled "Risk Assessment of Operational Events, Volume 2 - External Events," (NRC, 2008) describes a way of defining those bins that recommends the use of, at least, three bins unless plant-specific considerations require more bins. The SFP scoping study used four bins.

The resulting bins are as shown in Table 4, along with the tabulated frequencies for various spectral and peak accelerations. Note that for bin 4, the representative bin PGA has been set to 1.2 g by convention, whereas for the other bins it is the geometric mean of the interval endpoints. These results are shown graphically in Figure 7.

Table 4: Seismic bins and initiating event frequencies

Bin #	Bin Range (g)	Bin PGA (g)	Approximate Initiating Event Frequency for USGS 2008 (/yr)
1	0.05 - 0.3	0.12	5.2E-04
2	0.3 - 0.5	0.9	2.7E-05
3	0.5 - 1.0	0.7	1.7E-05
4	> 1.0	1.2[1]	4.9E-06

[1] Assumed based on PRA modeling convention

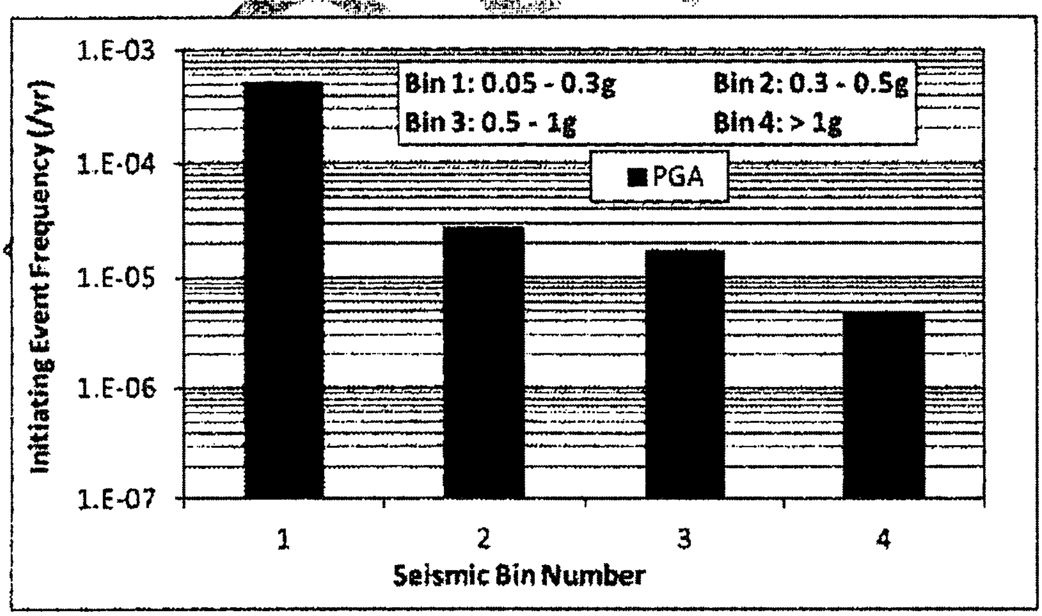


Figure 7: Comparison of seismic initiating event frequencies

Based on this information and on a review of the results of past studies, which indicate that damage to the SFP and other relevant SSCs is considered unlikely for events in bins 1 and 2, the SFP Scoping Study focused on bin 3. It was judged that seismic bin 3 provides the best compromise between events with higher occurrence frequencies that would lead to little or no damage versus higher consequence events with very low frequencies. Review of past studies, e.g., NUREG-1738 (Collins and Hubbard, 2001) indicates that events in bin 3, with initiating annual frequencies of the order of 1 to 2 x 10^{-5}, have the potential of challenging the structural integrity of the spent fuel pool, i.e., of causing a leak, at Peach Bottom. Thus, this is the initiating earthquake chosen for the SFP Scoping Study. It is an event that is no more severe than events considered in past reactor and SFP probabilistic risk assessment and consequence studies.

The SFP Scoping Study therefore considers a challenging, but very low probability earthquake as the initiating event, which was selected based on the considerations indicated above. This translates into a seismic event with a PGA several times greater than that associated with the Design Basis Earthquake (DBE) currently called the Safe Shutdown Earthquake (SSE). The PGA for the Peach Bottom SSE is 0.12 g. (This is about a magnitude 5.3 earthquake at about 25 km.) While the probability of occurrence of this earthquake was not used in its determination, the annual probability of occurrence for this PGA is about 1.8×10^{-4} (approximately one event in 5,500 years) when calculated using the EPRI and USGS 2008 models, and about 3.2×10^{-4} (approximately one event in 3,200 years) when calculated using the LLNL model. An initial determination, largely based on the results of past studies (Collins and Hubbard, 2001, Prassinos, 1989) and engineering judgment, was that the ground motions associated with the SSE (bin 1 event, i.e., the least severe bin) would not be large enough to damage to the SFP at Peach Bottom.

The frequency content of the ground motions considered for the SSE defined in terms of a ground motion response spectrum differs from the frequency content of the estimated uniform hazard response spectrum for this rock site, called the Ground Motion Response Spectrum (GMRS) (NRC, 2007) based on the USGS 2008 information. Other ground motion response spectra of interest for the SFP Scoping Study are the free-field response spectra used in the seismic PRA for the NUREG-1150 study. Volume 1, Part 3 of NUREG/CR-4550 (Lambright et al., 1990) provides the horizontal and vertical free-field response spectrum used in the NUREG-1150 seismic PRA for Peach Bottom in terms of the median spectral ordinates for various values of the PGA. The frequency content for the ground motions considered in the NUREG-1150 also differs from the frequency for the site GMRS.

The information above coupled with the review of previous studies (e.g., Collins and Hubbard, 2001) suggests that the frequency of a seismic event that has the potential for challenging the integrity of the spent fuel pool at Peach Bottom is on the order of $1.7 \cdot 10^{-5}$/yr (i.e., approximately one event in 60,000 years). This frequency is contrasted against other sources of information in Table 5. For reference, for the Mineral, Virginia earthquake of August 23, 2011 near the North Anna nuclear power plant, the NRC staff concluded using data from USGS instruments that the PGA at the North Anna site was about 0.26g (NRC, 2011). Using the USGS 2008 information for North Anna, the hazard frequency for an event with this PGA is about $1.2 \cdot 10^{-4}$/yr (1 event every 8,300 years). This frequency places the Mineral, VA event in seismic bin 1.

Table 5: Comparison of seismic frequencies from various sources

Source	Estimated initiating event frequency of a large seismic event	Notes
USGS 2008	$1.7 \cdot 10^{-5}$/yr (one event in 60,000 years)	Frequency of seismic bin 3
Peach Bottom 3 SPAR-EE Model (v3.21, Rev. 1)	$1.3 \cdot 10^{-5}$/yr (one event in 77,000 years)	Frequency of seismic bin 3 (of 3)
NUREG-1738[1]	$1.1 \cdot 10^{-5}$/yr (one event in 90,000 years)	Frequency of seismic hazard between 0.51g to 1.02g
INEL-96/0334	$3.2 \cdot 10^{-6}$/yr (one event in 310,000 years)	Frequency of seismic events > 0.6g

[1] Initiating event frequencies reported are those based on the LLNL models (Sobel, 1994).

3.3. Characterization of the Ground Motion Response Spectra

The free-field ground motion response spectrum (GMRS) for horizontal earthquake shaking for this site is based response spectra and PGA used in conjunction with research assessments for the GI-199, which used the USGS 2008 model. Specifically, based on the USGS 2008 model, an average free-field response spectrum called the site GMRS was determined in the GI-199 study for the Peach Bottom site (a rock site). This site GMRS has a zero period spectral acceleration (PGA) of about 0.34 g. Figure 8 shows a comparison of a modified version of the GMRS (modified to include frequencies as low as 0.2 Hz) considered for this study to the horizontal response spectrum for the site SSE (PGA of 0.12 g) for 5-percent damping.

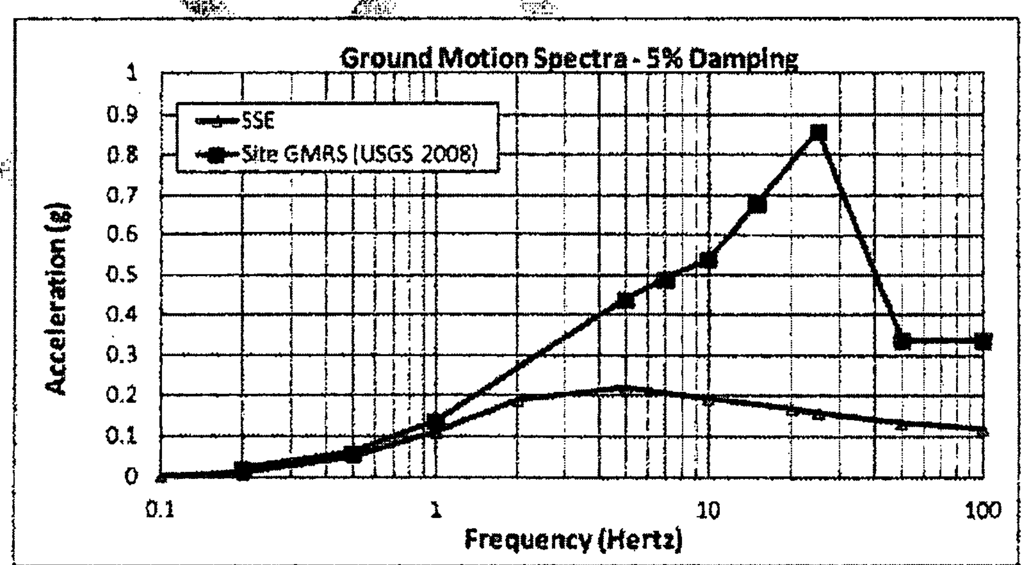


Figure 8: Site GMRS (GI-199, USGS 2008) and SSE (Horizontal Ground Motion)

The ground motion response spectra for the event considered in the SFP Scoping Study (about 0.7 g PGA) were derived from the GMRS for horizontal ground motions at the site as follows:

- Horizontal shaking - The site GMRS derived on the basis of the USGS 2008 model scaled to a PGA (zero period acceleration) of about 0.70 g (specifically 0.71 g).
- Vertical shaking - The response spectrum for the vertical ground motion is taken to the same as the response spectrum for the horizontal ground motion on the bases that nearby earthquakes would control the ground shaking spectra for this event and that the frequencies of interest for this study are frequencies above 5 Hz (ASCE, 1999, McGuire, Silva and Costantino, 2001).

Other ground motion response spectra of interest for the SFP Scoping Study are the free-field response spectra used in the seismic PRA for the NUREG-1150 study (Lambright et al., 1990). Figure 9 provides a comparison of the frequency content of the horizontal response spectra (5-percent damping) for the SSE, the median response spectrum used in NUREG-1150 study, and the site GMRS based on the USGS 2008 model. For this comparison, all spectra are scaled to a PGA of 1.0 g. When the three response spectra under consideration are scaled to the same PGA, the information in Figure 9 supports the following observations:

- For frequencies between about 10 Hz and 45 Hz, the site GMRS (based on the recent USGS 2008 model) has higher spectral accelerations than the ground shaking considered for the SSE and for the NUREG-1150 study.
- For frequencies between about 0.5 Hz and 10 Hz, the site GMRS (based on the recent USGS 2008 model) has lower spectral accelerations than the ground shaking considered for the SSE and the NUREG 1150 study.

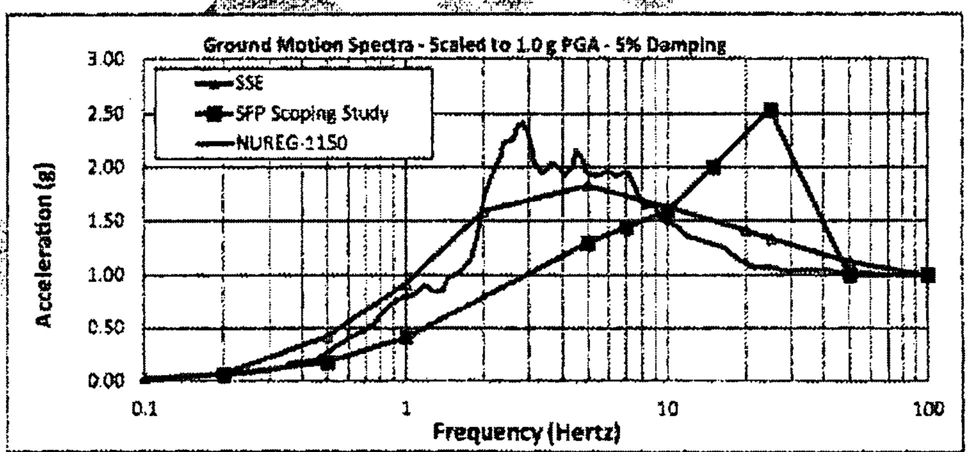


Figure 9: Response spectrum for 5-percent damping scaled to 1.0 g PGA: SSE, NUREG/CR-4550 (NUREG-1150 PRA), and this study (GI-199, USGS 2008)

As noted above, the horizontal response spectrum for the event considered in the SFP Scoping Study is the site GMRS derived using the USGS 2008 model (PGA of about 0.34 g) scaled to a PGA of about 0.7 g. Figure 10 shows the horizontal response spectra (5-percent damping) for the event considered in the SFP Scoping Study, for the SSE (0.12 g PGA), and for the response spectrum used in the NUREG-1150 PRA (for a PGA of 0.3 g) scaled to the PGA of the event for

this study. The information in Figure 10 illustrates how the ground motions considered in this study are considerably more challenging than those for the SSE. However, these ground motions are also significantly less likely as indicated above.

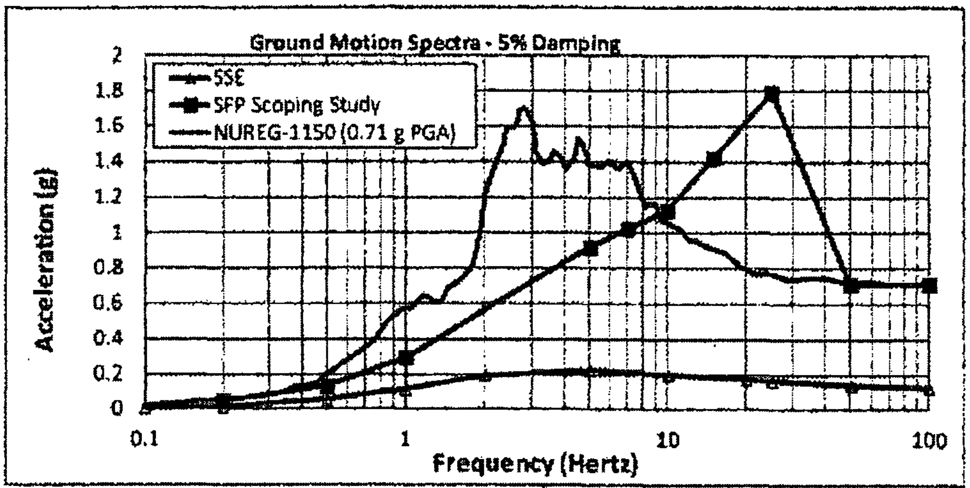


Figure 10: Horizontal response spectrum (5-percent damping): SSE, SFP Scoping Study and NUREG-1150 PRA (0.71 g PGA)

3.4. Acknowledgement of New USGS Information

As noted above, a group of stakeholders consisting of the NRC, the Department of Energy (DOE) and EPRI, and with the active participation of USGS scientists who contributed to the development of the national seismic hazard maps, is further developing and updating the USGS 2008 information in a collaborative study, which includes: (a) the seismic source zone characterization and (b) the ground motion attenuation models. In addition, the NRC is developing methods and computer codes for independent probabilistic seismic hazard analysis, which will be publicly available when completed, to combine (a) and (b). Although part (a) of this updating effort has been completed (NRC 2012b), parts (b) and (c) are still ongoing. Therefore, the SFP Scoping Study used the earlier USGS 2008 model and information instead of the ongoing update program.

References

ASCE (1999), "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," American Society of Civil Engineers (ASCE), ASCE Standard 4-98, 1999.

Bernreuter, D. L., Savy, J. B., Mensing R. W., and Chen J. C (1989), "Seismic Hazard Characterization of 69 Nuclear Power Plant Sites East of the Rocky Mountains," NUREG/CR-5250, Vols. 1-8, January 1989.

Collins, T. E. and Hubbard, G., (2001), "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, February, 2001.

Lambright, J.A., et al., (1990), "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events," NUREG/CR-4550, Vol. 4, Rev 1, Part 3, December, 1990.

McGuire, R. K., Silva, W. J., and Costantino, C. J.,(2001), "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines," NUREG/CR-6728, Prepared by Risk Engineering, Inc., Boulder, Colorado, October, 2001.

NRC (1990), "Severe Accident Risks: An Assessment for Five US Nuclear Plants, (Final Summary Report)," NUREG-1150, December, 1990.

NRC (2002), "Perspectives Gained From the Individual Plant Examination of External Events (IPEEE) Program," NUREG-1742, April 2002.

NRC (2007), "A Performance-based Approach to Define the Site-Specific Earthquake Ground Motion," Regulatory Guide 1.208, NRC, March, 2007

NRC (2008), "Risk Assessment of Operational Events, Volume 2 – External Hazards," Handbook, Revision 1.01, U.S. Nuclear Regulatory Commission, January, 2008.

NRC (2011), "Virginia Earthquake Frequently Asked Questions," Virginia Quake - NRC Actions, http://www.nrc.gov/about-nrc/emerg-preparedness/virginia-quake-info.html

NRC (2012a), "Generic Issue Management Control System Report for Fiscal Year 2012 2nd Quarter," Office of Nuclear Regulatory Research, NRC, 2012. (available at http://www.nrc.gov/reading-rm/doc-collections/generic-issues/quarterly/index.html)

NRC (2012b), "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," NUREG 2115, Department of Energy (DOE) Report DOE/NE-0140, Electric Power Research Institute Report EPRI 1021097, 2012 (available from http://www.ceus-ssc.com).

Petersen, M. D., Frankel, A. D., Harmsen, S. C., Mueller, C. S., Haller, K. M., Wheeler, R. L., Wesson, R. L., Zeng, Y., Boyd, O. S., Perkins, D. M., Luco, N., Field, E. H., Wills, C. J., and Rukstales, K. S., 2008, "Documentation for the 2008 Update of the United States National Seismic Hazard Maps" U.S. Geological Survey (USGS) Open-File Report 2008–1128, 2008.

Prassinos, P. G., et al., (1989), "Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants," NUREG/CR-5176, January, 1989.

Sobel, P., (1994), "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," NUREG-1488, April, 1994.

Toro, G. R., McGuire, R. K., Jacobson, J. P., O'Hara, T. F., Silva, W. J., (1989), "Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue," Electric Power Research Institute (EPRI), Report NP-6395-D, Prepared for the Seismicity Owners Group (SOG), April, 1989.

4. STRUCTURAL ANALYSIS AND RELATED INITIAL DAMAGE CHARACTERIZATION

This section documents the structural analysis performed to estimate the related initial damage states for the accident progression analysis. It provides the objective, approach including assumptions, structural modeling and analyses as well as the related potential damage states and their relative likelihoods for the seismic event considered.

The objective of the structural assessments was to provide damage states that might result from the seismic event described in the previous section and constitute the initial conditions for the accident progression analysis. Structural and related damage states have been divided into two major categories as listed in the following.

- Structural damage to the spent fuel structure with the potential locations of leakage from cracking of the concrete and related liner tearing.
- Other damage states to include:
  - Amount of water, if any, displaced by sloshing of the water in the spent fuel pool.
  - Damage to refuel gate, support systems and penetrations, and qualitative assessment of damage to spent fuel racks and spent fuel assemblies.
  - Damage to the reactor building and other relevant structures.

Most of the analytical effort focused on assessing potential structural damage to the spent fuel structure, namely concrete distortions, concrete cracking, and liner strains at the bottom of the pool. This is based on the review of past studies which indicates that damage to the spent fuel pool in those locations, if it were to occur, would be the more significant damage state in terms of loss of coolant.

This section includes a review of the performance of spent fuel pools at four nuclear power plants during two major recent earthquakes in Japan, namely: the March 3, 2011, Tohoku earthquake (with moment magnitude $M_w = 9.0$) and the July 16, 2007, Niigataken Chuetsu-Oki earthquake ($M_w = 6.6$). The power plants considered in the review are: Fukushima Daiichi, Onagawa, Fukushima Daini and Tokai for the Tohoku earthquake; and Kashiwazaki-Kariwa for the Niigataken Chuetsu-Oki earthquake. An essential part of this review is a comparison to relevant aspects of the seismic scenario for the SFP Scoping Study. Although these review and comparison are preliminary and use information available at the time of the execution of the SFP Scoping Study, they assist in the interpretation of the results obtained for the seismic scenario and spent fuel pool considered in this study.

The remainder of this section consists of two major parts (subsections). Subsection 4.1 addresses the damage states for the SFP structure. This part includes the approach used, seismic loads, a description of the SFP structure, the loads on the SFP including hydrodynamic loads, the finite element modeling and analysis of the SFP structure under these loads, and the estimation of potential damage states and their relative likelihood given the seismic event. This part ends with a review of the performance of spent fuel pools in two major recent earthquakes in Japan together with a comparison of loads from those events and from the scenario for the SFP Scoping Study. Subsection 4.2 presents the assumptions, approach and damage estimates for the other damage states listed above.

4.1. Damage States for the Spent Fuel Pool Structure

4.1.1. Approach and Seismic Loads

The general approach for the estimate of the damage states follows, with appropriate modifications for the specific needs of this study, the approach reported in NUREG/CR-5176 (Prassinos, 1989). The analyses reported in NUREG/CR-5176 were conducted in conjunction with research activities related to Generic Issue 82 (GI-82), "Beyond Design Basis Accidents in Spent Fuel Pools," (NRC, 2012a). Appendix 2 of NUREG-1738 (Collins and Hubbard, 2001), a technical study of spent fuel pool accident risk at decommissioning nuclear power plants, provides a review of the approach and results in NUREG/CR-5176 (Prassinos, 1989). A conclusion of that review is that the overall approach in NUREG/CR-5176 provides an acceptable means to estimate seismic structural damage to the spent fuel pool.

Approach

The overall approach used in the SFP Scoping Study to assess damage to the SFP structure, namely concrete cracking, concrete distortions, liner strains and liner tearing, for the earthquake event considered, consists of the following 9 steps:

1. Obtain free-field ground motion response spectra (horizontal and vertical) for the site considered (a rock site and a reactor building with small embedment). For this scoping study, the response spectra are based on the USGS 2008 model used for Generic Issue 199 (GI-199) (NRC 2012b) as indicated in Section 4.
2. From reliable and well-documented past studies, obtain in-structure response spectra (ISRS) (also called floor response spectra) for the vertical and horizontal directions at the elevation of the base of the spent fuel pool (Elevation 195 ft). For reference, the elevation at the top of the refueling floor is Elevation 234 ft and the elevation at the top of the foundation slab is Elevation 92 ft 6 in.
    a. The SFP Scoping Study used the median-centered ISRS calculated for the Peach Bottom Atomic Power Station (PBAPS) for the seismic PRA performed for the NUREG-1150 study (NRC, 1990) and reported in Volume 4, Part 1, Revision 1 of NUREG/CR-4550 (Lambright, 1990).
3. Estimate in-structure response spectra for the ground motions of interest for this study at the elevations of interest (Step 2a) by scaling the ISRS from previous studies (Step 2a) to account for differences in the response spectra shapes of the free-field ground motions considered in the NUREG-1150 seismic PRA and in this study (see Figure 5 when both response spectra are scaled to 1.0 g PGA).
4. Use the scaled ISRS from Step 3 to estimate equivalent static forces to be applied, in conjunction with dead loads, to the floor and walls of the SFP as input for a static nonlinear pushover analysis. These equivalent static forces account for: (i) peak vertical and horizontal accelerations of the floor and walls of the SFP structure (seismic coefficients); (ii) peak vertical and horizontal hydrodynamic impulsive pressures on the floor and walls of the SFP from the water in the pool; and (iii) vertical dynamic forces on the SFP floor from the dynamic response of the racks and spent fuel assemblies. At this stage of the analysis also estimate vertical displacement of the water surface from sloshing.
    a. The SFP Scoping Study used a simplified three-dimensional (3D) finite element model of the SFP structure to estimate or verify these loads. Specifically, it used

elastic solid elements and special fluid elements to model the water, in order to estimate natural frequencies and mode shapes for the SFP structure. This model was then used to calculate the spatial distribution of peak vertical and horizontal accelerations of the structural components using the ISRS from Step 3 as input.

b. Hydrodynamic impulsive pressures, peak vertical and horizontal pressures, were calculated on the basis of simplified methods (Housner, 1963; AEC, 1963; and Malhotra, Wenk and Wieland, 2000). The 3D finite element model used in Step 4a together with the ISRS from Step 3 as input were also used to estimate peak hydrodynamic pressures. This provided a verification of the suitability of the pressures calculated using simplified methods, which were the ones used in the analyses.

c. The 3D finite element model used in Step 4a with the ISRS from Step 3 as input was also used to estimate vertical displacements of the water surface from sloshing. The estimated displacements were small when compared to the depth of water in the SFP as indicated below.

5. Perform a 3D static nonlinear pushover analysis of the SFP structure using a detailed 3D finite element model of the SFP structure that includes nonlinear modeling of concrete including cracking as well as modeling of the steel reinforcement, embedded steel floor beams and the SFP liner. Perform the static nonlinear pushover analysis for adequate combinations of the vertical and horizontal ground motions to account for the fact that maximum vertical and horizontal accelerations do not occur simultaneously (NRC, 2007). Accordingly, perform the nonlinear static pushover analysis as follows:

   a. Incrementally apply the dead loads to the SFP structure to calculate initial stresses and strains. Dead loads considered for this study consist of: the weight of the pool structural components, weight of water, weight of the spent fuel assemblies and weight of the spent fuel racks.

   b. Follow Step 4a with an incremental application of adequate combinations of the horizontal and vertical equivalent static forces estimated in Step 4. The incremental application is needed to track development and effects of concrete cracking, concrete strains, steel yielding and liner strains.

   c. In this study, peak vertical seismic loads were combined with 40-percent of the peak horizontal loads. Alternatively, peak horizontal loads on both directions were combined with 40-percent of the vertical loads. Preliminary analyses indicated that the load combination involving peak vertical loads and 40-percent of the horizontal loads would be the most severe combination for the SFP structure analyzed. Accordingly, this was the combination studied in more detail in the remainder of the study.

   d. Use best-estimate material properties for all materials, e.g., concrete, reinforcement, steel beams and liner, based on best available information for similar materials used in nuclear plants and other guidance for the assessment of extreme events and for seismic fragility assessments. Also take into account the effect of aging on the concrete strength as recommended for the assessment of beyond design basis, extreme events.

6. Review and process the calculated structural distortions (as measured by the displacement of nearby nodes), structural deformations, concrete strains and liner strains in order to:

a. Assess the possible development of cracks through the floor or walls (the analyses indicated that critical concrete cracking such as this would only develop at the base of the walls along the intersection of the SFP walls with the SFP floor), and then estimate crack lengths and average crack width.

b. Assess liner strains at the intersection of the base of the walls and floor slab in order to assess the potential for liner tearing. Take into consideration details of the attachment of the liner, in discrete locations, to the concrete floor and walls.

7. Define three initial states for the subsequent accident progression analysis as follows:

   a. A state with no leakage, and no loss of coolant, from the bottom of the SFP. This state corresponds to concrete cracking at the base of the walls (estimated to be through wall cracking for the event considered as shown in subsequent subsections) but without tearing of the liner.

   b. A state with moderate leakage rate from the bottom of the SFP, corresponding to through wall concrete cracking at the bottom of the walls and extensive tearing of the liner that propagates to an extent such that water leakage is controlled by the size of the cracks in the concrete.

   c. A state with small leakage rate from the bottom of the SFP, corresponding to through wall concrete cracking at the bottom of the walls and tearing of the liner that remains localized to those parts where the floor liner is attached to the SFP floor near the walls.

8. For the two damage states with leakage, estimate the leakage rate at the base of the SFP walls. When the rate is controlled by the cracking in the concrete (moderate leakage rate) use recent large scale test data for the flow of water though thick concrete slabs together with the estimated crack width and length to estimate the leakage rate. When the rate is controlled by localized liner tearing use empirical date from leakage through cracks in steel pipes to estimate the leakage rates.

9. Use data for ultimate strains in the types of steel used for SFP liners, together with uncertainties in the calculated liner strains (accounting in approximation for uncertainties in the estimation of the in-structure loads and concrete properties) to estimate the relative likelihoods for the three initial damage states listed in Step 8: no leakage, moderate leakage rate and small leakage rate.

As noted above, this approach parallels part of the approach used in conjunction with the resolution of GI-82 (Prassinos et al., 1989). It augments the earlier approach in that it uses modern finite element methods in Steps 4 and 5. Use of finite element analyses in Step 4 is necessary to obtain a more accurate assessment of the natural frequencies of the SFP structure itself, to estimate the spatial distribution of seismic coefficients and to verify the estimated hydrodynamic impulsive loads on the floor and walls of the SFP. Use of finite element analyses in Step 5 is necessary to track the development of cracking and liner strains and then related those to damage states, leakage rates and their relative likelihoods.

The approach described above has potentially conservative aspects that may overestimate the damage to the spent fuel structure. A possible conservative aspect is that generally ISRS accelerations do not necessarily increase proportionally (linearly) from low PGA events to an event with a PGA as high as that considered in this study. As the load increases, the structure of the reactor building may crack and dissipate energy thus dampening the response of the building. This was taken into account, in approximation, in the scaling of the ISRS from previous

studies (using 0.36 g PGAs) by considering an increase on the damping of the reactor building structure.

Another potentially conservative aspect of the analysis is that the scaling of the ISRS does not take into account reductions on the high frequency (greater than 10 Hz) spectral accelerations that may result, under some circumstances, from ground motion incoherency, wave scattering, soil-structure interaction effects and wave passage effects. This is accounted for, in part, in the calculation of the relative likelihood of the various damage states by considering a small range of reduction in the response and associated uncertainties, as discussed below.

Other approximations of note include: (i) the scaling ISRS calculated from a ground motion with response spectra markedly different at high frequencies (greater than 10 Hz) from the ground motion spectra considered for this study; (ii) the decoupling of the response of the SFP from the response of the reactor building; and (iii) neglecting the small embedment of the reactor building (especially for the calculation of horizontal ISRS). Follow-subsections address these approximations. Another approximation is the use of a static pushover-like analysis to estimate damage to the SFP. More detailed approaches involving the use of sampled time-histories, including sampling of incoherent ground motions used in conjunction with 3D models of the entire reactor building and soil-structure interaction analysis, to calculate loads on the SFP would provide additional insights on some of these possible conservatisms. However, this would be contradictory with the scoping nature of the study.

The weight of other SFP equipment and appurtenances on the dead loads, and thermal stresses are not accounted for in the estimation of the initial stresses in the SFP components (Step 5a). The weight of other equipment is small in comparison to the other loads in the pool. Thermal stresses are not accounted for under the assumption that concrete cracking will relieve the thermal stresses. Moreover, increase in the temperature of the water, if it were to occur, would not happen until several hours after the termination of the ground shaking.

<u>Seismic Loads</u>

Section 3 provides and discusses the bases for the free-field ground motion response spectra for the seismic event considered for the SFP Scoping Study. As noted in Section 3 other free-field response spectra of interest in this study are those documented in NUREG/CR-4550 (Lambright, 1990) and used in the probabilistic risk assessment (PRA) for NUREG-1150. That report documents and provides median-centered ISRS (for 5-percent damping) for the Peach Bottom reactor structures calculated using time-history analysis and an ensemble of free-field ground motions time-histories. Section 3 provides a comparison of the median-centered free-field ground motion response spectra for that ensemble of time-histories to the ground motion response spectrum for the seismic event considered in this study.

The free-field response spectra and ISRS reported in Lambright (1990) form a set of reliable and well-documented response spectra for the Peach Bottom reactor buildings. Specifically, that report provides ISRS at various elevations of interest in the reactor building, namely at the bottom elevation of the SFP (Elevation 195 ft) and at the refueling floor (Elevation 234 ft). In addition, the report also provides estimates of frequencies of vibration for the reactor building, which are listed in Table 6. These frequencies help understand the shape of the ISRS for the Peach Bottom reactor building. It is noted that the frequencies of vibration of the SFP structure (discussed in the following section) and of the SFP floors and walls range from about 18 to 25 Hz depending on the cracking of the concrete, boundary conditions (restraints) and how the mass of the water is accounted for in the calculation of these frequencies. These frequencies

are remote (detuned) from the frequencies for the horizontal mode of vibration for the reactor building but are close to its vertical frequency.

Table 6: Estimated frequencies of vibration for the Peach Bottom reactor buildings (Lambright, 1990)

Direction	Frequency (Hz)	% Mass
Horizontal (NS)	7.1	68
Horizontal (EW)	7.6	71
Vertical	18.5	72

Using simplified scaling procedures, the ISRS in Lambright (1990) were scaled to estimate floor vertical and horizontal ISRS at the elevation at the bottom of the SFP as well as horizontal ISRS at the mid height of the SFP walls (by averaging scaled spectra at Elevation 195 ft and Elevation 234 ft). The scaling was done by estimating the ground motion amplification factors from the ground motion to the ISRS and then applying those factors to the response spectra for the SFP scoping study. This scaling was done using the reported median-centered ISRS for 5-percent damping and the EW components for the ISRS (examination of the charts indicates that this components tend to have the higher spectral accelerations). The SFP Scoping Study considered identical horizontal ISRS for both directions. Note that for the SFP studied the horizontal components of the ground motion are not those with the greatest damage potential. Justification for not considering reduction of the high frequency spectral accelerations is provided at the end of this subsection.

Figure 11 shows a comparison of the vertical ISRS for the elevation at the bottom of the SFP, estimated as indicated above to the corresponding ISRS (smoothed) for the NUREG-1150 seismic PRA. Likewise, Figure 12 provides a similar comparison for the horizontal ISRS estimated at the mid-height of the SFP structure (average of the ISRS at Elevation 195 ft and Elevation 234 ft).

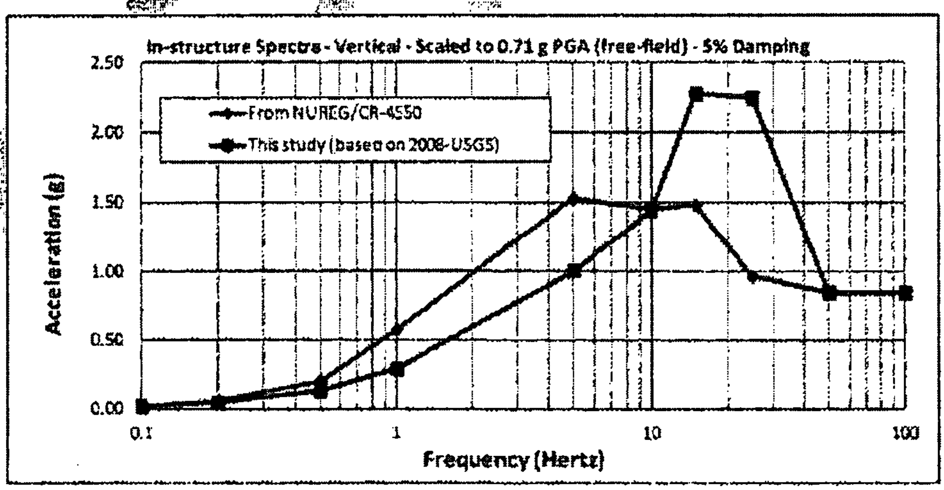


Figure 11: Vertical ISRS for 5-percent damping at Elevation 195 ft (bottom of the SFP)

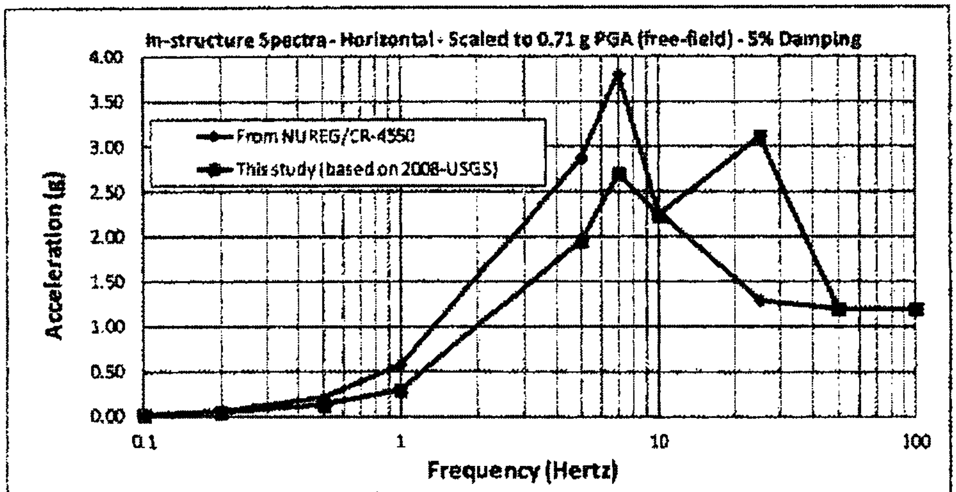


Figure 12: Horizontal ISRS for 5-percent damping midway between Elevation 195 ft and Elevation 234 ft (mid-height of the SFP)

The spectra shown in Figure 11and Figure 12 are for 5-percent damping for the reactor building and equipment. Calculation of seismic load coefficients for the SFP floors and walls considered a higher damping ratio for the reactor building but not the equipment. This is done on the assumption that for the intense ground motion of the event considered, the reactor building will undergo more cracking than that estimated for the design basis motion (SSE) which will absorb and dissipate energy and damp the response. Accordingly, spectral accelerations for frequencies greater between about 1 Hz and 50 Hz were scaled for a structural damping of the main building of about 10-percent of the critical. The scaling is done on the basis of spectral scaling factors for various damping ratios shown in NUREG/CR-0098 (Newmark and Hall, 1978). The resulting spectra are as shown in Figure 13 and Figure 14.

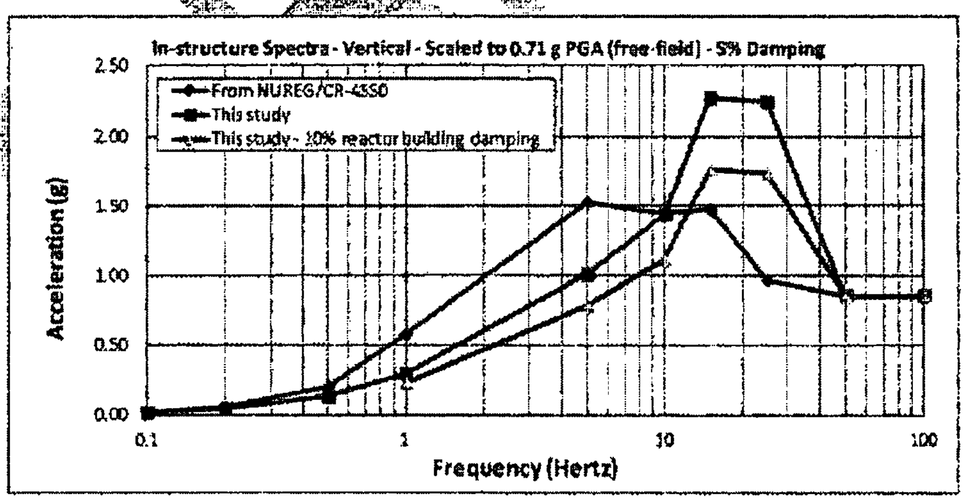


Figure 13: Vertical ISRS for 5-percent and 10-percent reactor building damping at Elevation 195 ft (bottom of the SFP)

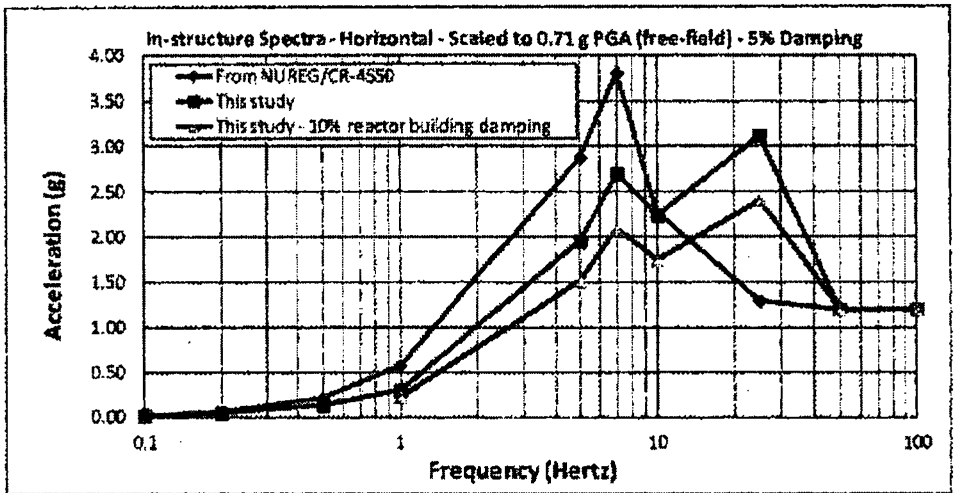


Figure 14: Horizontal ISRS for 5-percent and 10-percent reactor building damping midway between Elevation 195 ft and Elevation 234 ft (mid height of the SFP)

A possible criticism of the scaling utilized (on the basis of it being conservative) is that it does not take into account reductions on high frequency (greater than 10 Hz) spectral accelerations. These reductions would result, under some circumstances, from ground motion incoherency, wave scattering, soil-structure interaction effects and wave passage effects.

The plant dimensions of the reactor building are about 150 ft by 120 ft above Elevation 135 ft (ground elevation) and about 150 ft by 150 ft below Elevation 135 ft. The building foundation consists of a 4 ft 4 in. reinforced concrete slab lying on top of sound rock with an elevated rock pedestal about 64 ft in diameter near the center for the drywell foundation (see Figure 15). The foundation slab above this rock pedestal is still a RC slab about 4 ft thick. The main structure of the reactor building extends from the top of the foundation at Elevation 92 ft 6 in. to the refueling floor at Elevation 234 ft, which is topped by a structural steel crane bay (rated at 120 tons). For this relatively complex and relatively flexible foundation, justification for large reductions on high frequency ISRS spectral accelerations is not clear. In addition, the distance between the supports of the SFP structure, which provide direct pathways from the vertical ground motions of the rock to the SFP, is of the order of about 65 ft. This distance is less than the distance that has been considered appropriate for justifying large reductions of high frequency ISRS spectral accelerations (ASCE 1999).

The above notwithstanding, results of past studies justify consideration of some reduction of the high frequency ISRS spectral accelerations even without further analysis. Possible reduction of high frequency ground motions is accounted for, in part, in the subsequent calculation of the relative likelihood of the various damage states. This is done by considering a narrow range of reduction in the response and associated uncertainties, as discussed in Section 4.1.5.

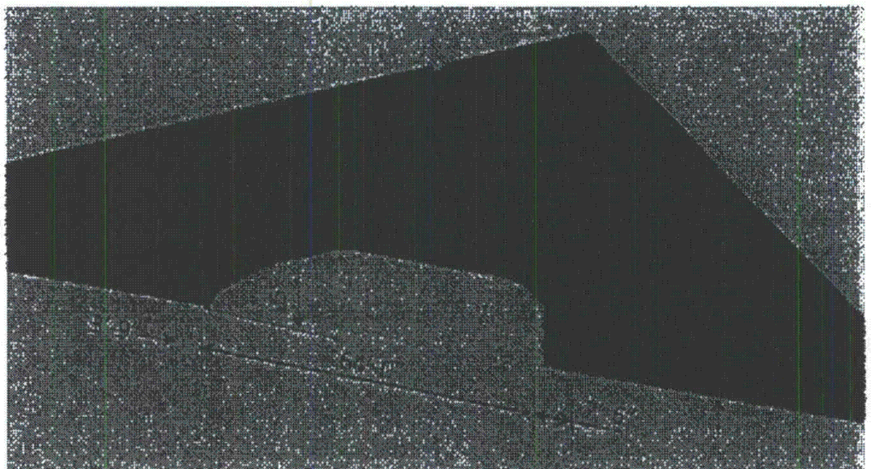


Figure 15: Schematic diagram of the reactor building foundation near the drywell

4.1.2. Description of the Spent Fuel Pool Structure

This section provides a brief description of the spent fuel pool structure and its relation to the main reactor building. The description identifies the main structural components and other aspects of relevance for this study.

The Final Safety Analysis Report (FSAR) for the Peach Bottom Atomic Power Station (Units 2 and 3) describes the SFP and the dryer-separator storage pool as a large channel-shaped beam (approximately 40 ft wide at the SFP structure). This channel beam is supported at the center by the biological concrete shield structure around the drywell and the ends by RC exterior walls on opposite sides of the reactor building. Figure 16 is a 3D representation of the SFP structure and dryer-separator storage pool. Figure 17 shows cutouts of 3D models of the reactor building that show the location of the SFP in relation to the remainder of the building. The 3D model at the left ends at the elevation of the refueling floor (Elevation 234 ft) while the model on the right shows the crane bay located above the refueling floor (but not the crane itself).

The detailed 3D finite element model of the SFP structure itself (see Figure 18) serves to identify the walls of the pool for further reference in this study. The East (E) and West (W) walls extend from the biological concrete shield to the outer wall of the reactor building. These walls, which are about 40 feet deep (above the top of the SFP floor) and about 6 feet thick in their lower half, support the entire weight of the SFP, which includes their own weight, the weight of the floor, water, spent fuel assemblies and racks, partition wall (South, S, wall). The E and W walls are supported at the thick RC biological shield building on the North (N) side and at the exterior wall of the building. An additional cavity exists between the SFP itself and the outer wall of the reactor building.

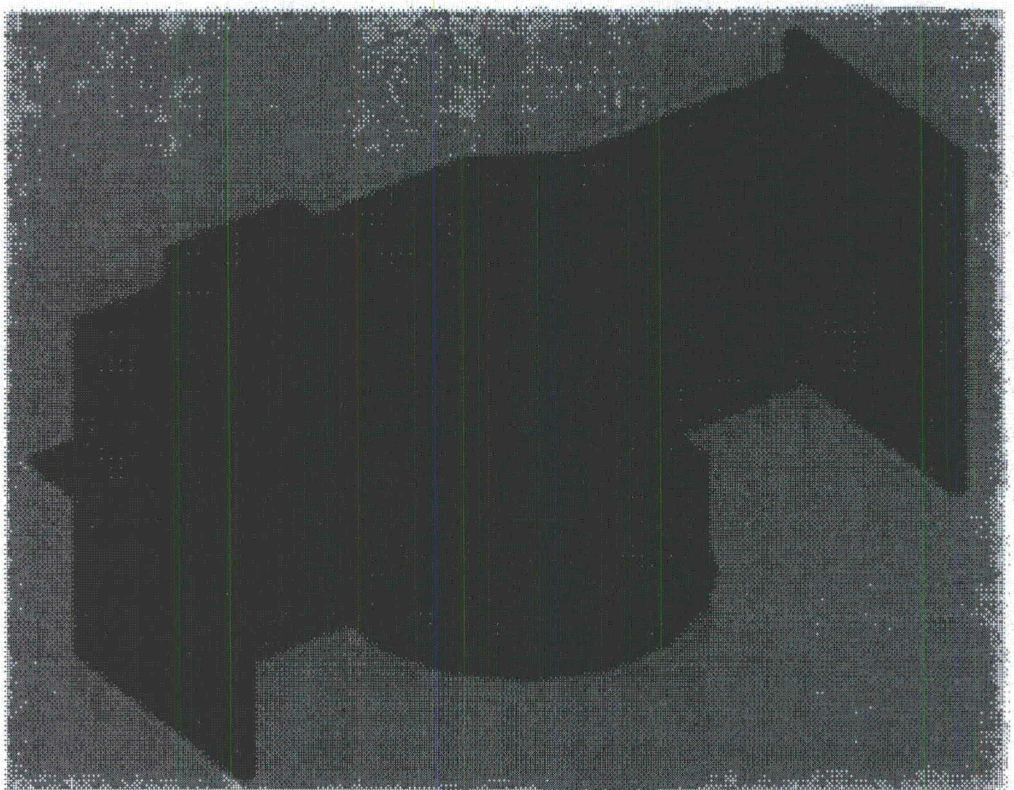
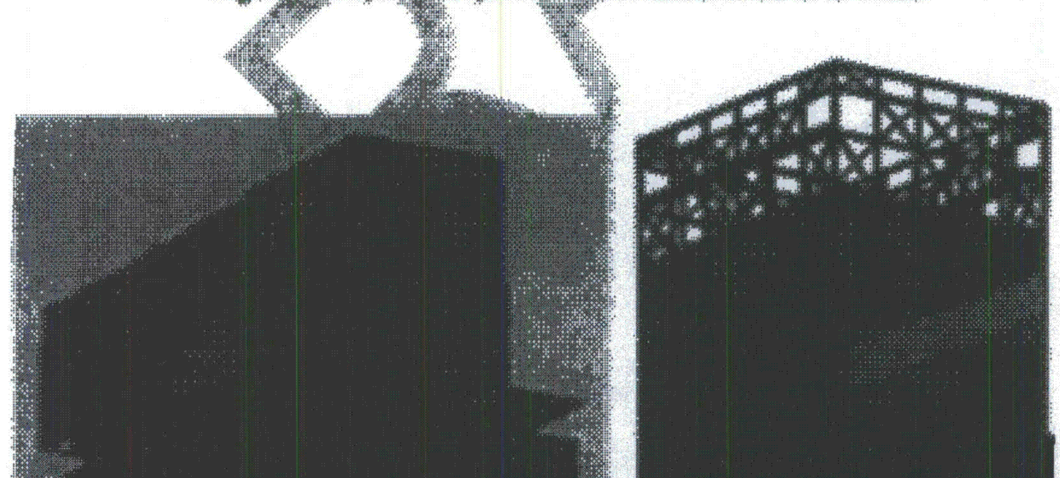


Figure [illegible] Spent fuel pool details in cutout of 3D CAD model



[redacted]

Figure 18: Finite element model of the spent fuel pool structure with labels for the floor and walls

Of interest for the scoping study is to assess damage and cracking to the walls identified in Figure 18
as well as to the floor of the pool from the challenging, low probability, seismic event considered in this study. The concrete walls are RC walls with vertical and horizontal layers of reinforcing steel bars near each face as well as near the mid surface of the walls.

The SFP floor consists of a 6ft 3 in. thick RC slab with embedded heavy steel W-Shapes (I beams) as shown in Figure 16. This floor framing was used during construction and designed to carry the weight of the wet concrete but were left embedded in the concrete floor to the depth of the lower flange of the shapes. The beams extending from the biological concrete shield to the out wall are W-36x300 (3 ft deep beams weighing about 300 pounds per foot) and those extending from one wall to the other are W-36x230 (3 ft deep and weighing about 230 pounds per foot). The floor is reinforced with steel rebar layers in two directions at the top of the floor and with a complex reinforcing pattern in between the steel girders within the clear span of the floor as well as in the portion of the floor under the side walls of the SFP.

The floor and walls of the are covered with a stainless steel designed to preclude inadvertent loss of water, which attached to the concrete using steel anchors and welds to steel plates and shapes embedded in the concrete floor and side walls. Figure 19, which is an outline of the 3D finite element model of a portion of the liner and its attachments to the concrete floor and walls, serves to identify some of these attachments. Interconnected drainage paths are provided behind the liner for drainage of small amounts of water that might leak through small cracks to a drain.

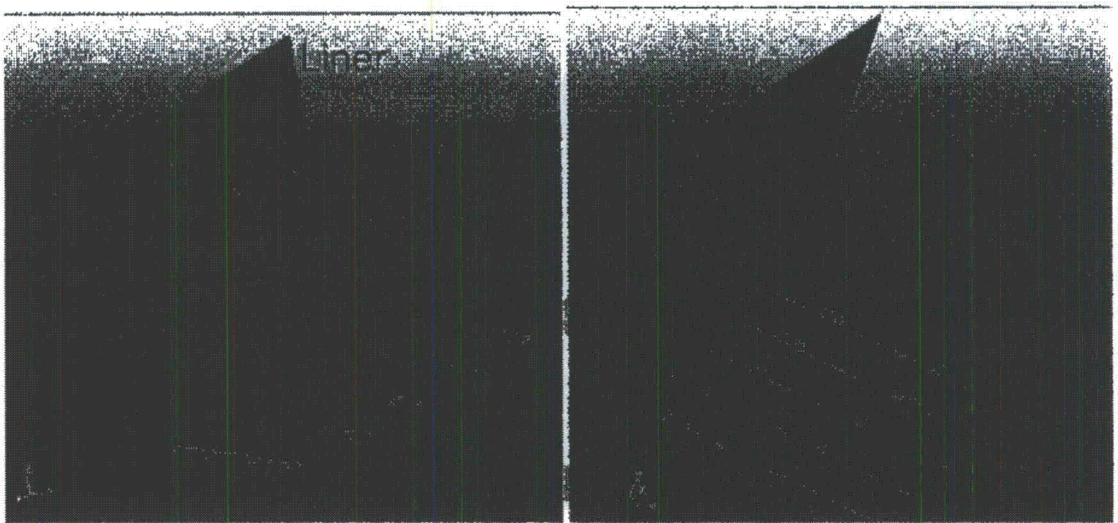


Figure 19: Outline of detailed finite element model of the SFP liner representing attachments to the SFP floor and walls

According to the Final Safety Analysis report, there are no connections to the SFP that would allow water to drain below the refueling gate or below 10 ft above the top of active fuel. The FSAR further states that lines below the levels in the previous paragraph are equipped with siphon breaker holes to prevent inadvertent drainage. In addition, the systems for maintaining water quality and quantity are designed so that failure or inappropriate operation of these systems does not cause uncovering of the fuel.

The refueling channel (in red on Figure 16) is covered by concrete blocks and closed by two steel gates that provide redundancy in the case of malfunction of a single gate. Each gate consists of steel plates with steel stiffeners. Each gate has seals around its perimeter that are kept under pressure by mechanical means by the locking system of the gates. It is not a pneumatic system that requires pressurization provided by electric systems.

4.1.3. Finite Element Analysis Model Description

Step 5 of the approach described in Subsection 4.1.1, the nonlinear pseudo-dynamic analysis of the SFP analysis under the combined dead loads and seismic loads, requires a detailed finite element model of the entire SFP structure in order to estimate concrete cracking and liner strains for the estimation of leakage areas. The LSDYNA finite element software was used for the analysis (LSTC, 2007). Figure 18 above already shows the overall detailed finite element model used for that purpose. The model has about 600,000 elements and uses 16 elements through the thickness of the E and W walls and equally refined detail for the SFP floor.

The finite element model included all the major reinforcing bars for the floor and walls of the SFP structure as well as of the outer walls and biological concrete shielding. This model also considered all steel shapes embedded in the floor of the SFP which were modeled using LSDYNA shell elements. In addition, the finite element model also include the steel liner in on the inside surface of the SFP. Figure 20 shows some of the components included in the finite element model.

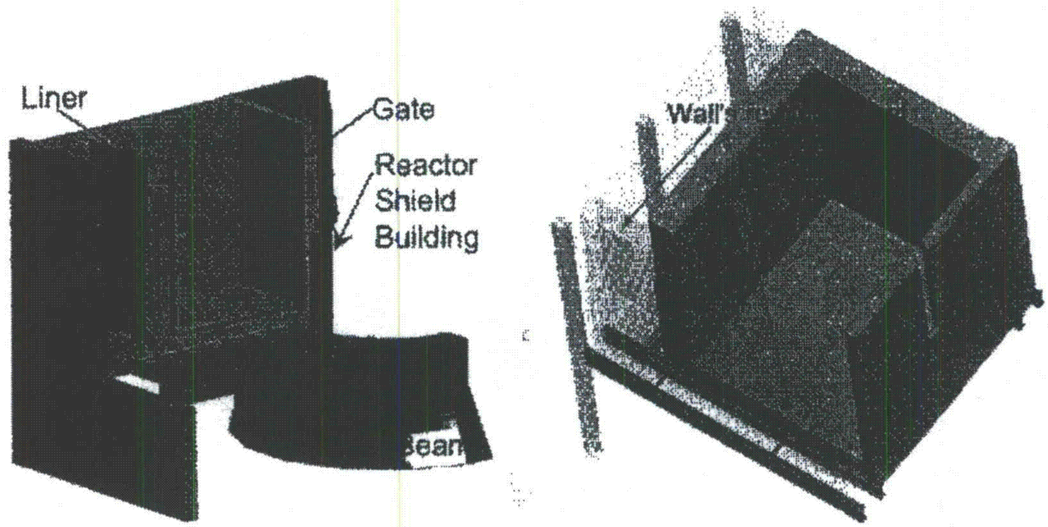


Figure 20: Cutouts of 3D finite element model showing components included in the model

Given the complexity of the structure, rather than using node to node modeling for the embedded shell elements of the steel shapes, the model used the "Constrained Lagrange in Solid" option available in LSDYNA to represent the coupling between the embedded elements and the concrete. In addition, the finite element model also include the steel liner on the inside surface of the SFP. For the steel liner, two modeling considerations were used. In the calculation of the overall response of the SFP to the combined loads, liner shell elements of the size of the underlying concrete elements were used and the liner was assumed to be bond to the concrete (node to node connections). A more detailed model of sections of the liner (see Figure 19 above) was subsequently used as an embedded gage to assess strain concentrations in the liner steel at the intersection of the floor and walls as discussed in the following section.

The finite elements in the model for the nonlinear analysis are as follows:

- Reinforcing bars – LSDYNA beam elements with the truss option.
- Concrete – Constant stress LSDYNA solid elements (reduced integration).
- Shell elements – Belytschko-Tsay shell elements.

Two material models were used as follows:

- Concrete – LSDYNA material model 159, the Continuous Surface Cap Model (CSCM) (FHWA, 2007). The analysis used the option of specifying a minimum number of

material properties, namely the unconfined compressive strength and aggregate diameter and allowing the model to calculate the other material properties of interest.

- Steel – LSDYNA material model 3, called plastic kinematic, which was used for all steels but with different material properties.

Table 7 provides a summary of the material properties used in the nonlinear finite element analyses. The properties for the concrete and steel reinforcement, assumed to the materials that would influence the most the overall response of the SFP, were taken to be best estimates of the median material properties. In the case of concrete, the unconfined compressive strength of the concrete was estimated based on recommendations used for the analysis of extreme events, namely aircraft impact assessment (NEI,) and a nominal concrete strength of 4,000 psi. For the other materials, the table primarily lists nominal properties.

Table 7: Material properties for the nonlinear finite element analyses

Material	Properties		
Concrete	Unconfined compressive strength	6400 psi	(44.6 MPa)
	Aggregate diameter	1.5 in.	(38 mm)
	Unit weight (and density)	145 lb/ft^3	(2.33 g/cm^3)
Rebars	Yield strength (Grade 40)	47850 psi	(330 MPa)
	Yield strength (Grade 60)	69000 psi	(475 MPa)
	Young's modulus	31x10^7 psi	(2.15x10^5 MPa)
	Tangent modulus	15x10^4 psi	(1000 MPa)
	Unit weight (and density)	479 lb/ft^3	(7.7 g/cm^3)
	Failure strain	0.10	
Liner and steel plate anchorages	Yield strength (Grade 40)	36000 psi	(250 MPa)
	Young's modulus	30x10^7 psi	(2.07x10^5 MPa)
	Tangent modulus	15x10^4 psi	(1000 MPa)
	Unit weight (and density)	479 lb/ft^3	(7.7 g/cm^3)
	Failure strain	N/A	
Beams	Yield strength	36000 psi	(250 MPa)
	Young's modulus	30x10^7 psi	(2.07x10^5 MPa)
	Tangent modulus	25x10^4 psi	(1700 MPa)
	Unit weight (and density)	479 lb/ft^3	(7.7 g/cm^3)
	Failure strain	0.10	
Anchor studs	Yield strength	36000 psi	(250 MPa)
	Young's modulus	30x10^7 psi	(2.07x10^5 MPa)
	Tangent modulus	25x10^4 psi	(1700 MPa)
	Unit weight (and density)	479 lb/ft^3	(7.7 g/cm^3)
	Failure strain	0.10	

The SFP Scoping Study also used a simpler model of the SFP structure, namely a simpler version of the model used for the nonlinear analysis. This is a model used in conjunction with the finite element analyses related to Step 4 of the approach described in Section 4.1.1. Specifically, the simpler model was used to estimate frequencies of vibration for the SFP structure, to estimate seismic load coefficients and to verify hydrodynamic impulsive pressures with the ANSYS (version 13) finite element software. The simplified finite element was used with linear analyses appropriate for its intended use, had fewer elements through the thickness of the walls and floor, and had a simpler representation of the concrete biological shielding.

This finite element model used solid, elastic finite elements to represent the structure of the SFP (concrete only) and fluid element to represent the water. Specifically, it used the ANSYS SOLID185 element, a 3D structural solid element, and the ANSYS FLUID80 element for the modeling of the water. Material properties considered with this model are as follows

- Concrete: (i) Young's modulus of 3.6×10^6psi (varied to account for cracking effects on stiffness); (ii) unit weight of 145 lb/ft^3; and a Poisson ratio of 0.15.
- Water: (i) bulk modulus of 3.16×10^5 psi; (ii) unit weight of 62.4 lb/ft^3; (iii) Poisson ratio of 0.0; and (iv) a viscosity of 1.64×10^{-4}.

The simplified finite element model was used in conjunction with the following analyses:

- Estimation of frequencies and modes of vibration for the SFP including the effects of water using Householder reduced methods for the low frequency modes and the Block Lanczos method for the high frequency modes.
- Related deterministic spectrum analysis using single point spectral accelerations at the supports together with the Complete Quadratic Combination (CQC) rule for the combination of modal responses. These analyses estimated seismic load coefficients for structural components and verified the magnitude of the hydrodynamic pressures on the SFP walls.

<u>Summary of Dead and Seismic Loads for the Finite Element Analysis</u>

As indicated in Section 4.1.1, the dead loads considered for the nonlinear seismic analysis are the weight of structural materials (concrete, reinforcement, steel beams, liner and other steel plates), the vertical and horizontal hydrostatic pressures of the water, and the weight of the spent fuel assemblies and racks. The weight of the structural elements was applied as gravity loads on the finite element analysis. Hydrostatic pressures were applied as vertical and horizontal pressures on the inside surfaces of the floor and walls of the SFP. Vertical loads on the SFP floor from the weight of the spent fuel assemblies and racks were also applied as pressures SFP floor. Table 8 lists approximate values of the dead loads on the SFP floor in terms of an equivalent vertical pressure on the SFP floor for the purpose of comparing the magnitude of these loads with those imposed by the earthquake. Table 9 has approximate values of peak equivalent seismic static loads (vertical) expressed in terms of an equivalent vertical pressure on the SFP floor.

Table 8: Approximate dead loads on SFP floor in terms of an equivalent vertical floor pressure

Load	Equivalent floor pressure (approximate) (lb/ft^2)
Weight of the floor	900
Vertical hydrostatic pressure	2300
Weight of spent fuel assemblies and racks	1700
Total	4900

Table 9: Approximate peak equivalent seismic loads in terms of an equivalent static vertical floor pressure

Load	Equivalent floor pressure (approximate) (lb/ft^2)
Hydrodynamic impulsive vertical pressure	4840
Floor slab acceleration	1400
Dynamic forces from spent fuel assemblies and racks	1750
Total	7990

The results shown in Table 8 and Table 9 indicate that the seismic loads (in terms of equivalent vertical pressures on the SFP floor) are approximately twice as large as the dead loads and that the hydrodynamic impulsive pressures on the SFP floor are the largest of all forces considered.

Finite element analyses with the simplified finite element model described above were used to estimate and verify the seismic forces listed in Table 9 using deterministic response spectrum analysis. The seismic input for this analysis was a single point spectral acceleration at the supports using vertical and horizontal ISRS like those described in Section 4.1.2. It is noted that the natural frequencies of the SFP, considering a reduction of the concrete Young's modulus to 80-percent of its original value and the mass of the water range from about 16 to 30 Hz. These are frequencies of interest for the estimation of both hydrodynamic impulsive pressures (vertical and horizontal) as well as peak accelerations of the floor (vertical) and walls (horizontal). Comparison of these natural frequencies with the free-field response spectra shown in Section 3 for this study shows that these frequencies are similar to those for which the ground motions for this study have spectral accelerations higher than those from the SSE when scaled to the same PGA.

Figure 21 shows contours of the peak vertical accelerations of the SFP floor obtained using the deterministic response spectrum analysis described in the previous paragraph with the vertical ISRS as a single point spectral acceleration input at the supports. The results shown in are for a free-field PGA of 1.0 g and were multiplied by 0.71 to estimate the peak accelerations (seismic coefficients) to be used as input for the nonlinear finite element analysis. To obtain corresponding forces for the nonlinear analysis, the area of the SFP floor was divided into a 5x5 grid and the peak vertical accelerations were sampled at the center of each element of this grid. These sampled peak accelerations were then used to calculate equivalent nodal forces for the nodes of the detailed LSDYNA finite element model for the nonlinear analysis. Estimation of equivalent nodal forces for the walls, both horizontal and vertical used a procedure analogous to that described for the vertical forces on the SFP floor.

Vertical hydrodynamic forces, which are proportional to the vertical spectral accelerations at the base of the SFP, are the largest seismic forces in Table 9. This is because the natural frequencies of the SFP structure, including the water, are resonant to the high frequency content of the free-field ground motion. Given the significance of these pressures, deterministic response spectrum analysis with the simplified ANSYS finite element model of the SFP was used to verify their magnitude. Figure 22 shows peak hydrodynamic vertical pressures calculated in this manner for the vertical ISRS at the supports of the SFP (taken to be the same at each support). The pressures shown in Figure 22 are for a free-field PGA of 1.0 g and need to be multiplied by 0.71 for comparison with the values in Table 9.

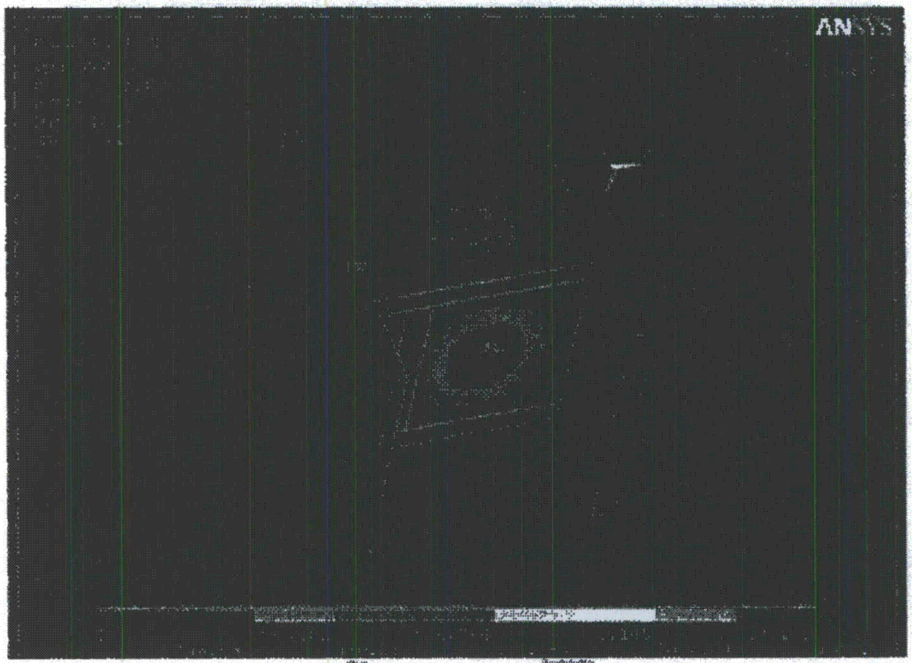


Figure 21: Estimated peak vertical accelerations of the SFP floor calculated using deterministic response spectrum analysis and vertical ISRS as input (1.0 g PGA)

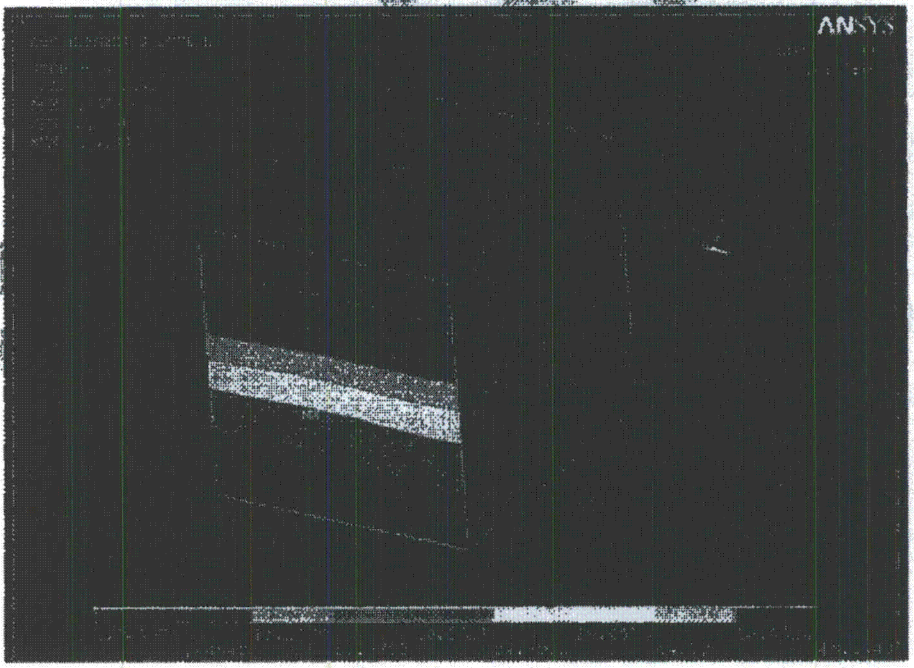


Figure 22: Estimated peak vertical hydrodynamic pressures on the SFP floor calculated using deterministic response spectrum analysis and vertical ISRS as input (1.0 g PGA)

4.1.4. Finite Element Analysis Results for the Spent Fuel Pool

This section presents a summary of the results obtained with the nonlinear finite element model described in the previous section for the loads described in Step 5 of the approach and described in Section 4.1.3 as well. The principal objective of the analysis was to track the deformation of the SFP structure, concrete cracking and liner strains to estimate potential leakage rates.

The analysis used the LSDYNA software which is an implicit dynamic finite element code. Since this is an equivalent static analysis, the analysis used mass scaling (with small changes in total mass of the model) together with slow ramping of the loads in order to minimize spurious dynamic effects. Specifically, the analysis slowly (with respect to the periods of vibration of the SFP structure) and proportionally incremented all dead loads until it reached their full values. Subsequently, the analysis slowly and proportionally applied all the equivalent seismic static loads until it reached their full values. Full values of the peak seismic loads were kept constant for some time in order to verify the stability of the response. Subsequently, the seismic forces were slowly and proportionally removed and then applied again to verify the nature of the response for an unloading and reloading path. The reloading path resulted in the same final condition as the original loading path.

Figure 23 shows vertical displacement contours for the load combination consisting of the dead loads, 100-percent of the vertical seismic loads and 40-percent of all horizontal seismic loads. The maximum displacements are near the center of the SFP floor and small, of the order of 15 mm (about 0.6 in.). Small displacements are a result of the high stiffness of the SFP structure which consists of thick RC slabs and walls (of the order of 6 ft) and comparatively short spans (from about 35 ft in the NS direction and about 40 ft in the EW direction).

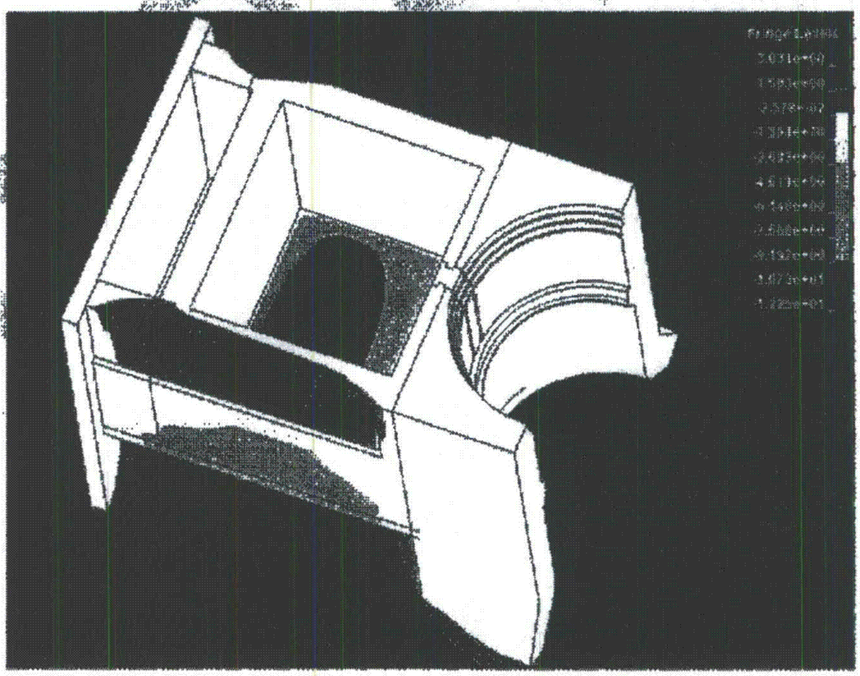


Figure 23: Contours of vertical displacements of the SFP floor and walls

Figure 24 shows vertical displacement along the outside face of the W wall. Of special interest in Figure 24 are the discontinuities of vertical displacement at the bottom of the SFP wall at the top of the SFP floor, which are identified by the transition between the blue and green contours near the center of span at the bottom of the wall. Discontinuities of vertical displacements in this region are of interest because, on the inside of the SFP, this is the region of strain concentrations in the SFP liner as shown in Figure 25. Finally, Figure 26 shows (with the red contour) the region of the SFP, at the bottom of the SFP walls and at the top of the SFP floor where the tensile strain of the concrete is exceeded and a crack starts to develop. The crack starts as a flexure crack and develops into a tension-flexure crack though the thickness of the wall.

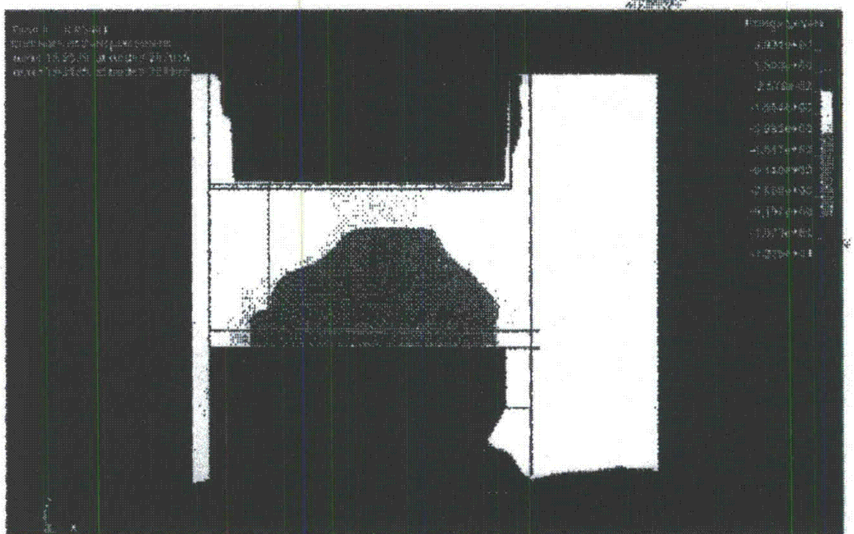


Figure 24: Contours of vertical displacement of the SFP walls

Figure 25: Liner strains (overall response not fully accounting for strain concentrations)

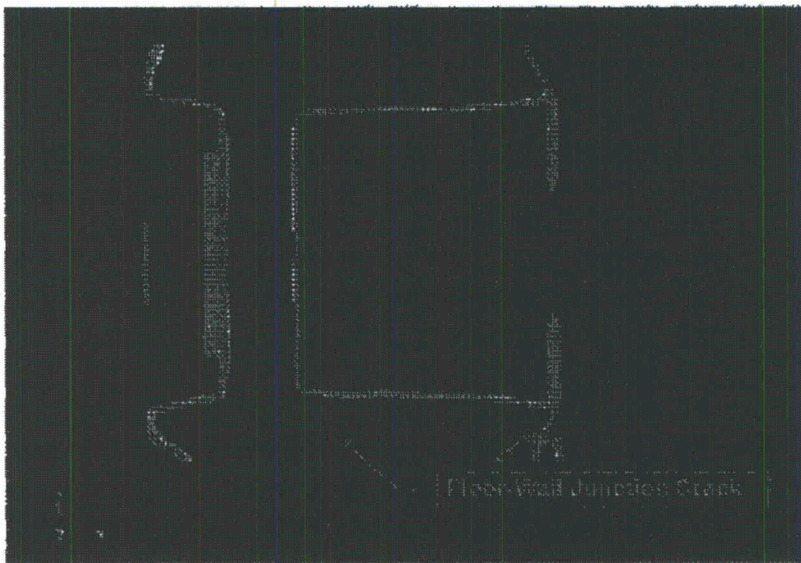


Figure 26: Region of concrete cracking initiation at the floor-wall junction

The higher liner strains in Figure 25 are, as expected, at the intersection of the SFP wall with the SFP floor, which is a region strain concentrations. Although this is a region of strain concentrations, the liner strains shown are small, of the order of $5x10^{-4}$ to $1.9x10^{-3}$. Which are less than or about 60-percent greater than the nominal liner yield strain of $1.2x10^{-3}$. The mesh size for the liner for this overall finite element analysis is not sufficiently small to fully capture strain concentrations in the liner. The main objective of this analysis was to obtain the overall deformation of the structure and the development of concrete cracking which is not expected to depend significantly on the liner modeling.

To assess strain concentrations in the liner, a detailed finite element model of the liner which includes the main details of its attachments to the floor and wall concrete was developed and is shown in Figure 19. The fine mesh of this liner inset uses elements as small as 0.15 in. (about 3.7 mm) at the transition from the floor to the wall. The analysis then used this detailed liner insert to estimate the liner strains. Specifically, the detailed insert was embedded into the original nonlinear finite element model of the structure. The SFP structure was then analyzed with the embedded detailed model of the liner (using the "Constrained_Lagrange_In_Solid" option in LSDYNA and appropriate contact definitions) to assess strain concentrations in the liner. Note that the liner is attached to the concrete only at a few discrete locations and is in contact with the concrete elsewhere. Specifically, at the junction with the wall, the liner is attached to concrete only near the backup plates between the floor and wall (see Figure 19) and is in contact with the concrete elsewhere along the floor-wall junction. For this reason, high strain concentrations are expected to develop only near the backup plates.

Figure 27 shows results of the analysis of the SFP with the embedded liner in a portion of the wall near the region where strain concentrations are expected to be the largest. The results show that the presence of the embedded liner as a gage did not affect the overall response of the SFP in a significant manner. However, it permitted an estimation of the strain concentrations in the liner.

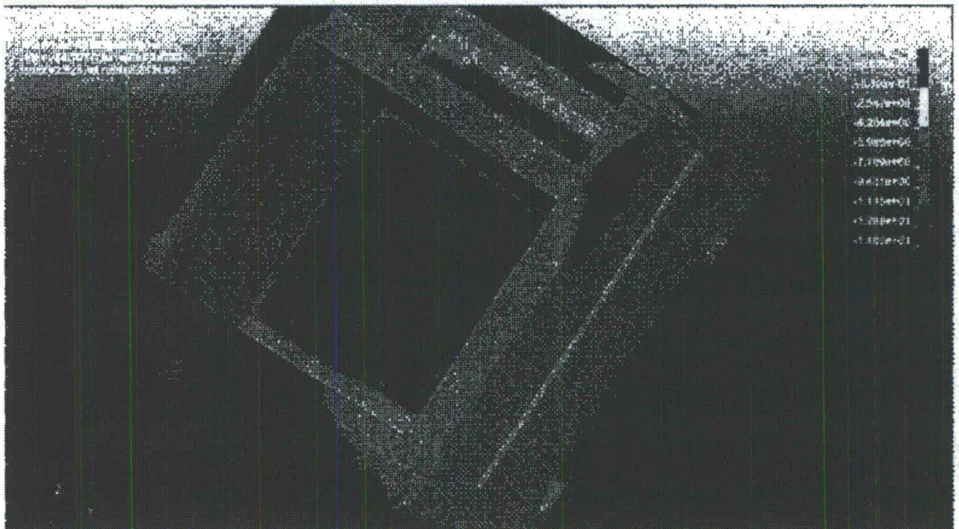


Figure 27: SFP displacement with detailed liner insert

Figure 28 shows the strain concentrations in the liner calculated using the detailed liner insert as indicated above. As expected strain concentrations are localized to the region of the liner near the backup plates at the lines where the liner is attached to the shapes embedded in the SFP floor. Elsewhere the liner strains remain small as indicated by the overall analysis with the coarser model. The maximum membrane effective strain in Figure 28 is about 3.7 percent (0.037). The following section uses these strains as well as estimates of the width and extend of the concrete cracking (see Figure 26) to assess liner tearing likelihoods for the scenario considered.

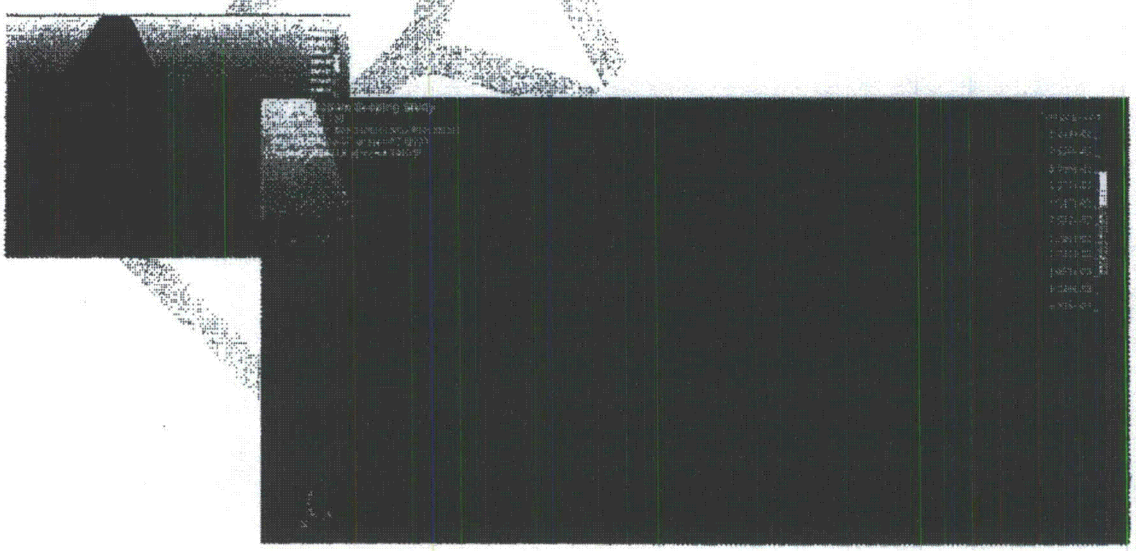


Figure 28: Strain concentrations in the SFP liner

4.1.5. Damage States

This section documents the results for Steps 6 to 9 of the approach defined in Section 4.1.1, which uses results from the nonlinear finite element analysis described in Section 4.1.5 to estimate leakage rates. These leakage rates are then used in the accident progression analysis to define the rate of loss of water from leakage at the bottom of the SFP.

Concrete cracking and moderate leakage rate

Post-processing of the displacements at the top of bottom nodes of the horizontal layer of concrete finite elements at the top of the SFP floor provides an estimate of the width and length of the cracking at the bottom of the SFP walls. This first step of this processing is the sampling of vertical displacements at the top and bottom nodes of this layer of concrete elements at various locations along the perimeter and through the depth of the wall. This is achieved by dividing the length of the base of the wall into segments and sampling those quantities at locations across the wall thickness near the center of each segment. The next step consists of subtracting the displacements of the top and bottom nodes for a first estimate of the crack width at the sampled locations. This estimate is then corrected by the vertical displacement of those nodes implied by the tensile strain of the concrete at cracking, which is comparatively small. A main assumption in this process is that a major single concrete crack (flexure-tension crack for this SFP) develops at the floor-wall junction rather than a set of closely spaced minor cracks. The next step averages the sampled crack widths though the thickness of the walls for each sampled segment at the base of the walls. Finally, the processing combines the crack areas estimated in this manner to estimate an average crack width and length which are as follows: average crack width of about 3.6 mm (about 0.14 in.) and average crack length of about 33,000 mm (about 108 ft), which describe a concrete crack with a non-smooth and non-uniform surface.

Estimation of the flow through this crack used recent experimental data for the flow of water through thick cracked concrete sections for hydraulic pressures similar to those in the SFP (Kanitkar et al., 2011). Crack widths and water pressures for those tests bracketed the average crack width estimated for the SFP and the water pressures in the SFP. The thickness of the concrete slabs is about half of the thickness of the SFP walls, meaning that these are large scale tests. Main results of that study are: an equation to estimate the leakage flow rate through concrete cracks that involves a friction factor determined by the experiments. Specifically, the study recommends the use of the following equation derived from the Navier-Stokes equations for incompressible flow of a Newtonian fluid:

$$\frac{P}{\rho g} = \frac{v^2}{2g} + f\frac{v^2}{2g}\frac{d}{2w}$$

where P is the pressure, ρ is the fluid density, g is the acceleration of gravity, v is the flow velocity, d is the crack depth (concrete thickness), and f is a friction factor. The results reported indicate that a friction factor of 0.8 is adequate for the average crack width estimated above.

Using the equation above for the leakage flow, and a friction factor of 0.8, assuming no initial loss of water and using the crack width and length estimated above, the leakage flow with time for the SFP was calculated as shown in Figure 29 in terms of the change of the water height in the SFP with time. The flow rate in that figure represents a moderate flow rate condition. However, for this condition to occur it is necessary that the liner strains exceed failure strains for the liner material at the region of strain concentrations (near the backup plates), that these tears

become unstable and the liner tearing spreads to an extent such that the leakage rate through the liner is greater than the leakage rate through the concrete crack. In this case, the concrete crack controls the leakage rate from the SFP. This is further discussed below in conjunction with the liner strains and liner failure criteria as well as the estimation of the relative likelihoods for the three damage states considered.

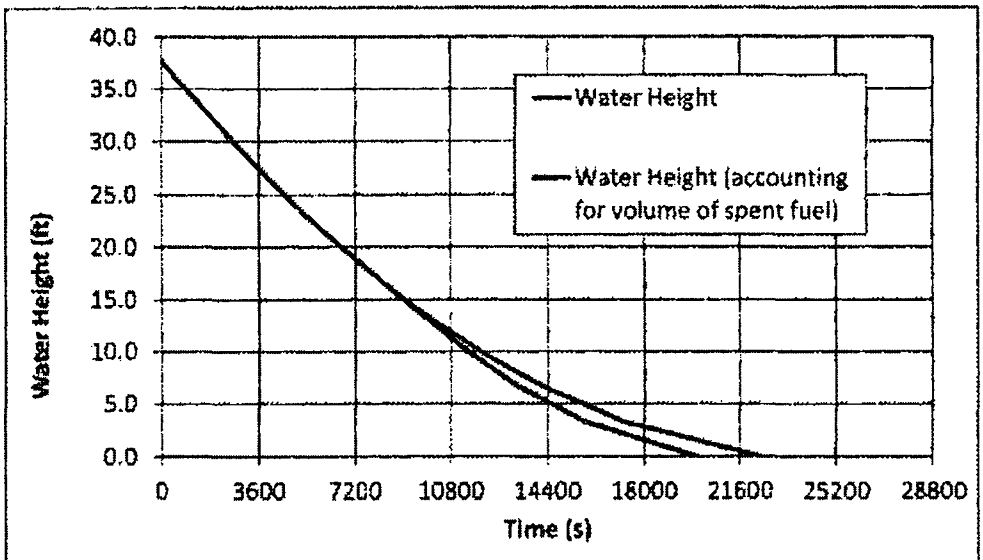


Figure 29: Moderate leakage flow rate (through concrete cracks)

Liner Strains and Small Leakage Rates

Maximum effective membrane liner strains from strain concentrations at the floor-walls junction are of the order of 0.037 (3.7 percent). These strains are localized at the backup plates, which are spaced two feet apart along the length of the E and W walls. Attachment details along the S wall are different, imposing less compliance of the liner to the concrete deformations, and are not expected to lead to strain concentrations as large as those at the base of the E and W walls. In addition, liner strains near the biological concrete shielding structure are small and liner tearing is not expected at the base of that wall. Accordingly, tearing of the liner, if it were to occur, would be along the base of the E and W walls.

Failure criteria for steel plates used in reinforced concrete containments included liners with corrosion damage, is used here to estimate failure strains for the SFP liner (Cherry, 1996). Failure criteria for liners without corrosion damage reported by Cherry (1996) are used in this study to estimate failure strains for the stainless steel SFP liner. On the base of the reported failure criteria, this study assumed a somewhat conservative estimate for the liner failure strain from the point of view of leakage rate in order to characterize the leakage rate for a damage state with small leakage flow rate.

Specifically, a liner strain at failure of 0.10 (10-percent) was assumed to estimate the width of a steel crack located at the floor-junction and localized at the location of the backup bars. The crack width was then calculated by multiplying this strain at failure by the width of the finite element with the maximum effective strain, which is approximately equal to 3.7 mm (0.15 in.) as indicated above. The resulting crack width for a liner tear localized at the location of the backup

bar is then estimated at 3.7x0.10 = 0.37 mm (0.015 in.). The crack length at each location is taken to be equal to the width of a backup bar which is equal to 4.0 in. (101.6 mm). Given that the spacing of the backup bars is 2 ft, a total of 40 backup bars (20 on each wall) are used to estimate the sum of all localized cracks at 4x40 = 160 in. The estimated width for each crack, if it were to occur, is then 0.015 in and the depth of the crack is the depth of the liner which is equal to 0.25 in.

Given the estimated width, length and depth for each localized liner tear and their number, it is still necessary to estimate the leakage rate through these tears. Estimation of this flow rate uses the following assumptions: (i) the flow rate can be estimated using an equation similar to that used for flow through the concrete cracks; and (ii) the friction factor for that equation can be calculated on the basis of test results for leakage rates through cracks in pipes. These assumptions are not validated at this time. Therefore, considerable uncertainty exists for the resulting leakage rate estimate.

Estimation of a friction factor was made using data in Paul (1994) for leakage through cracks in steel pipes. Back calculation of friction factors from data presented in this reference shows a large variability in the calculated friction factor. In particular, the friction factor appears to depend heavily on the smoothness of the crack surface. Also, the fluid in the pipe is at high temperature and the driving pressures are much higher than those applicable to the SFP. Review of other flow models reported in Paul (1994) indicates that for relatively smooth cracks friction will be low. On the bases of this information, the equation for the flow through concrete cracks was used for flow through steel cracks with a small friction factor (0.11) for an initial estimate of the leakage flow. On that basis, the leakage flow though the steel cracks (small leakage flow) was calculated as shown in Figure 30. It is important to emphasize that considerable uncertainty exists in the calculation of reported leakage rate given the localized liner tears.

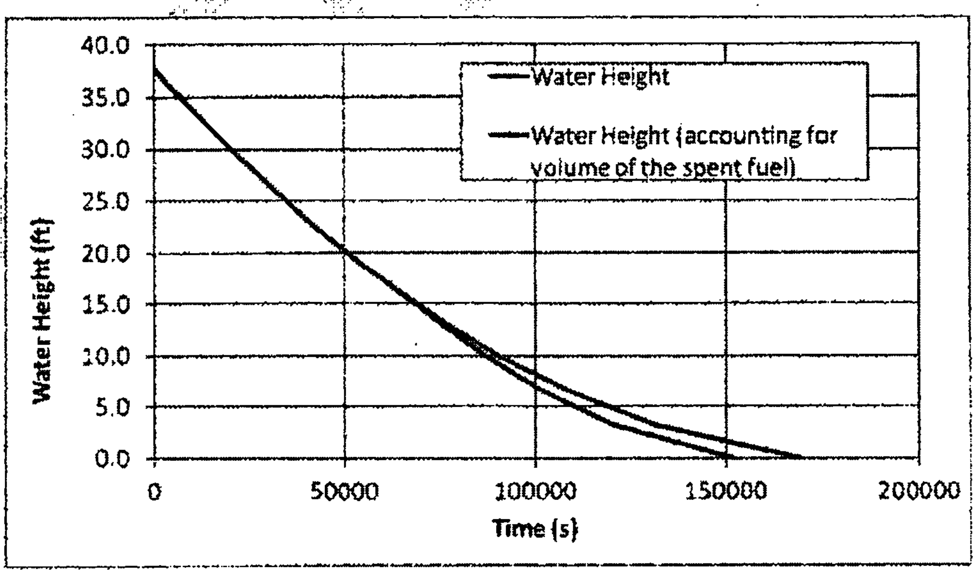


Figure 30: Small leakage flow rate (through localized steel tears)

Damage States and Relative Likelihoods

Step 6 of the approach (Section 4.1.1) defined three initial damage states as follows:

a. No leakage - A state with no leakage from the bottom of the pool. This state corresponds to concrete cracking at the base of the walls (estimated to be through wall cracking for the event considered as shown in subsequent subsections) but without tearing of the liner.

b. Moderate leakage rate - A state with leakage from the bottom of the SFP, corresponding to through wall concrete cracking at the bottom of the walls and tearing of the liner that propagates to an extent such that water leakage is controlled by the size of the cracks in the concrete. The estimated leakage flow rate for this damage state is shown in Figure 29.

c. Small leakage rate - A state with leakage from the bottom of the SFP, corresponding to through wall concrete cracking at the bottom of the walls and tearing of the liner that remains localized to the where the floor liner is attached to the SFP floor near the walls. The estimated leakage rate for this damage state is shown in Figure 30.

This study uses strain criteria, including uncertainties, for tearing of reinforced concrete steel liners (Cherry, 1996) together with uncertainties in the calculated liner strains (accounting in approximation for uncertainties in the ISRS estimation and concrete properties) to estimate the relative likelihoods for the three initial damage states listed above.

The results of these estimates indicate that the state with no leakage is the most likely with a relatively likelihood in excess of 90-percent. The relative likelihood of the two states with leakage from the bottom of the SFP is less than 10-percent. Estimation of the relative likelihood of the two damage states with leakage is subject to considerable uncertainties. Accordingly, the SFP Scoping Study assumes that both states are equally likely.

4.1.6. Review of Spent Fuel Pool Performance under Recent Major Earthquakes in Japan

Five Japanese nuclear power plants with a combined total of 20 reactors and 20 spent fuel pools were subjected to severe ground motions from major earthquakes in the past 5 years as follows:

- March 3, 2011, Tohoku earthquake (with moment magnitude M_w= 9.0)
  - Fukushima Daiichi (5 BWR Mark 1 SFPs)
  - Onagawa (3 BWR SFPs)
  - Fukushima Daini (4 BWR SFPs)
  - Tokai (1 BWR SFP)
- Niigataken Chuetsu-Oki earthquake (M_w= 6.6)
  - Kashiwazaki-Kariwa (7 BWR SFPs)

Other than some loss of coolant from sloshing, e.g., unit 6 a loss of 1.2 cubic meters from the SFP at Unite 6 of Kashiwazaki-Kariawa (Kawamura, 2008), no leakage flow near the bottom of the SFPs has been reported for any of the SFPs in those nuclear power plants.
This section provides a comparison of ground motion indices and ISRS spectral accelerations considered for this study and observed at the various units of those nuclear power plants. Although these review and comparison are preliminary and use information available at the time

of the execution of the SFP Scoping Study, they assist in the interpretation of the results obtained for the seismic scenario and spent fuel pool considered in this study.

It is noted that the seismic design loads for the various reactors considered in this comparison differ, for the most part, from the design basis loads for the site considered in the SFP Scoping Study. A possible exception to this would be the Unit 1 at Fukushima Daiichi, which initially considered comparable seismic design basis loads. However, seismic design basis loads for this reactor were subsequently revised upwards (those are the design loads reported in this comparison). This difference in the seismic design basis loads and the fact that details of the reinforcement (e.g., out of plane shear reinforcement if any) and construction of the various SFPs affected by the recent past experience in Japan are not known add uncertainties to the comparisons. Another source of uncertainty for this comparison is that the recorded ground motions and related PGAs at the various sites are not for the most part free-field ground motions and, therefore, not directly comparable to the free-field PGA considered in the scoping study. Additional sources of uncertainty are the type of reactor (e.g., several of the reactors are Mark 2 reactors instead of Mark 1 reactors), site conditions (e.g., soil vs. rock sites), reactor building foundation and reactor building embedment.

Table 10 to Table 14 show horizontal and vertical PGAs observed at the foundation slab of the various units for each of the nuclear power plants. Those tables also list the design PGAs for each of the reactors. For comparison, the vertical and horizontal PGAs for the free-field ground motion considered in this study are about 0.7 g. On the basis of the values reported on those tables, the following observations are possible:

- Horizontal PGAs at the foundation slabs of all reactors are less than those considered in the scoping study with the exception of that for Unit 2 of Kashiwazaki-Kariwa.
- Vertical PGAs at the foundation slabs of all reactors are for the most part less than horizontal PGAs with the exception of Unit 1 at Fukushima Daiini and the reactor at Tokai.
- Vertical PGAs at the foundation slabs of all reactors are less than those considered in the scoping study. The difference between the recorded PGAs and the PGA for the scoping study is greater for the vertical accelerations than for the horizontal accelerations.
    - It is noted that for the scoping study the vertical PGA is taken to be approximately equal to the horizontal PGA on the assumption that the dominant seismic event for this scenario will be an earthquake at a distance of about 15 km or less.

Table 10: Fukushima Daiichi, measured and design (DBGM Ss) PGAs at foundation slab (Tohoku, 2011 earthquake)

Unit	Reactor	Measured			Design Values		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark 1	460	447	258	487	489	412
2	Mark 1	348	550	302	441	438	420
3	Mark 1	322	507	231	449	441	429
4	Mark 1	281	319	200	447	445	422
5	Mark 1	311	548	256	452	452	427
6	Mark 2	298	444	244	445	448	415

Table 11: Onagawa, measured and design (DBGM S_S) PGAs at foundation slab (Tohoku, 2011 earthquake)

Unit	Reactor	Measured			Design Values		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1		540	587	439	532	529	451
2		607	461	389	594	572	490
3		573	458	321	512	497	476

Table 12: Fukushima Daiini, measured and design (DBGM S_s) PGAs at foundation slab (Tohoku, 2011 earthquake)

Unit	Reactor	Measured			Design Values		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark 2	254	230	305	434	434	512
2	Mark 2	243	196	232	428	429	504
3	Mark 2	277	216	208	428	430	504
4	Mark 2	210	205	288	415	415	504

Table 13: Tokai, measured and design (DBGM S_s) PGAs at foundation slab (Tohoku, 2011 earthquake)

Unit	Reactor	Measured			Design Values		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark 2	214	215	189	393	400	456

Table 14: Kashiwazaki-Kariwa, observed and design PGAs at foundation slab (Chuetsu-Oki, 2007 earthquake)

Unit	Reactor	Measured			Design Values		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark 2	311	680	408	274	273	235
2	Mark 2	304	606	282	167	167	235
3	Mark 2	308	384	311	192	193	235
4	Mark 2	310	492	337	193	194	235
5	Mark 2	277	442	205	249	254	235
6	ABWR	271	322	488	263	263	235
7	ABWR	267	356	355	263	263	235

Another aspect of interest for this comparison is the frequency content of the ground motions as characterized by response spectra. The site chosen for the scoping study is a rock site and the ground motion response spectra for the seismic scenario considered shows high spectral amplifications for frequencies greater than about 10 Hz.

Figure 31 shows vertical response spectra for 5-percent damping at the foundation slab of Unit 1 (the case with an horizontal PGA of about 0.7 g) and Unit 4 of Kashiwazaki-Kariwa and the corresponding response spectrum for the vertical ground motion for the scoping study. The comparison indicates that the ground motion for this study has higher vertical spectral accelerations at the frequencies of vibration for the SFP structure and its components. The results shown are typical of those for the other reactors at Kashiwazaki.

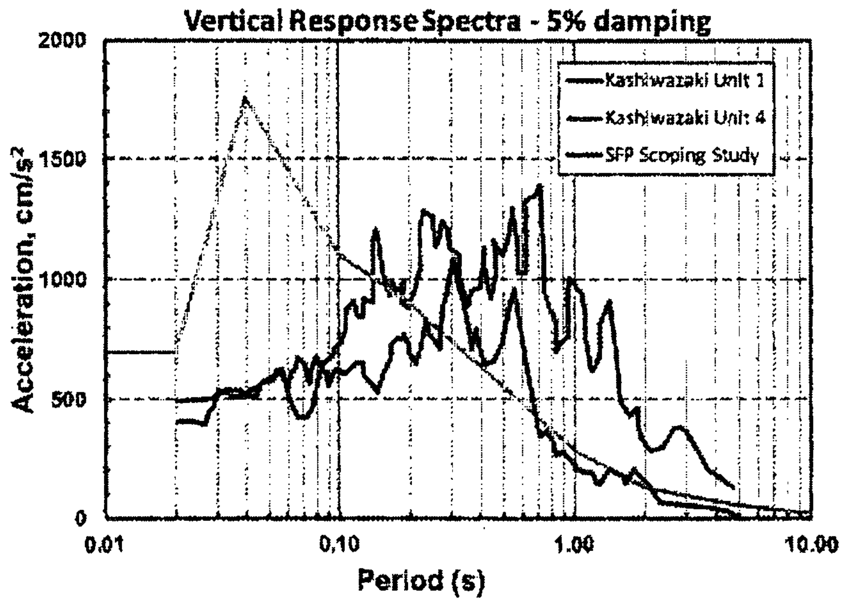


Figure 31: Vertical response spectra: Kashiwazaki-Kariwa Units 1 and 5 (foundation) and SFP scoping study (free-field)

Vertical response spectra for the other reactors were not available at the time of the scoping study so the comparison of the frequency content of the ground motions is made using horizontal spectra. Figure 32 shows horizontal response spectra for 5-percent damping at the foundation slab of Unit 1 (the case with an horizontal PGA of about 0.7 g) and Unit 4 of Kashiwazaki-Kariwa and the corresponding response spectrum for the horizontal ground motion for the scoping study. The comparison indicates that the ground motion for this study (rock site) has higher horizontal spectral accelerations at the frequencies of vibration for the SFP structure and its components. The results shown are typical of those for the other reactors at Fukushima Daiichi.

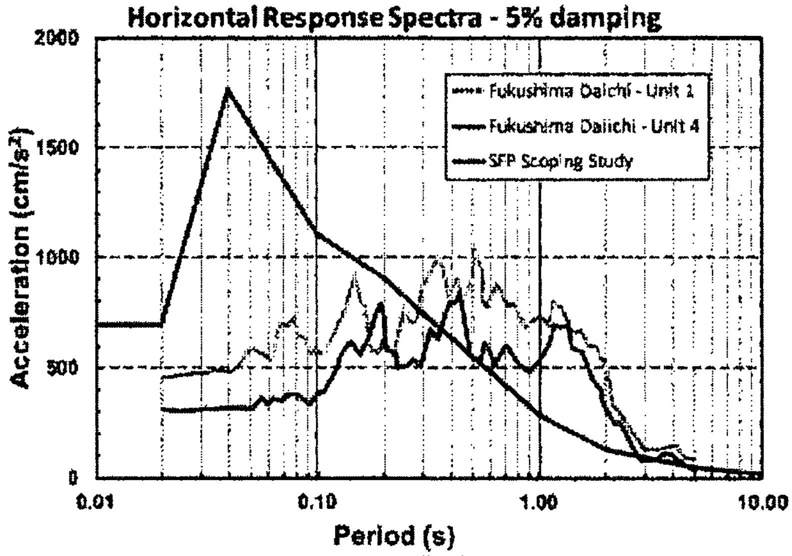


Figure 32: Horizontal response spectra: Fukushima Daiichi Units 1 and 4 (foundation) and SFP scoping study (free-field)

Figure 33 shows vertical ISRS at an elevation at about mid height of the refueling floor Unit 1 and Unit 4 of Kashiwazaki-Kariwa together with the vertical ISRS for the scoping study. ISRS for the scoping study are shown for 5-percent and 10-percent damping of the reactor building. In both cases, the ISRS for the scoping studies is higher than the observed ISRS for frequencies close to the frequencies of the SFP for the scoping study. However, for 10-percent damping for the reactor building, the ISRS for Unit 4 approaches that for the scoping study at frequencies equal to about 18 Hz. The comparison with the data for Unit 1 is more typical of that for the other units. Vertical ISRS were not available at the time of the scoping study for comparison with the ISRS for this study.

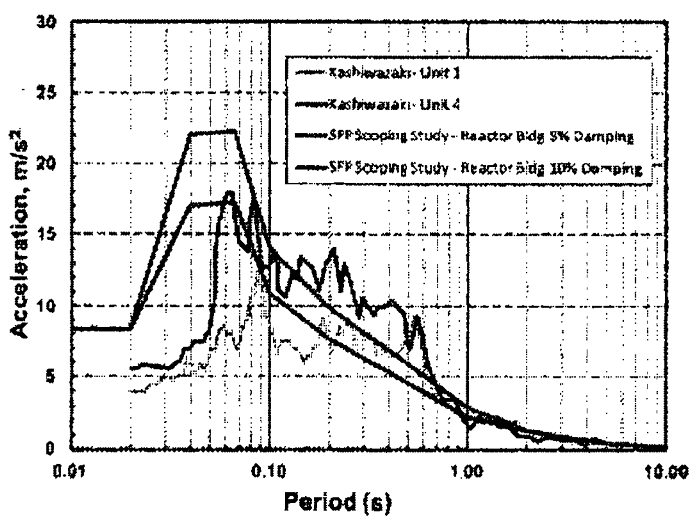


Figure 33: Vertical ISRS for kashiwazaki-Kariwa Units 1 and 3 and for the SFP Scoping Study

The review and comparison show indicate that the seismic loads for this study tend to be more challenging to the SFP than those for the observed events. One reason for this conclusion is the higher frequency content of the ground motions for the rock site chosen for the scoping study at frequencies near the natural frequencies of the SFP structure and its components. Another reason is that the vertical PGA for the scoping study reason is substantially greater than that for the observed events in Japan.

A conclusion from the review is that for the challenging events considered, leakage from the bottom of the SFPs of 20 BWR reactors was not reported. However, possible differences in the design and construction of the SFPs, including higher design basis seismic loads, for the Japanese plants leads to uncertainties in this preliminary comparison.

4.2. Other Damage States

Assessment of other damage stages is primary based on: (i) finite element deterministic response spectra analysis to estimate maximum vertical displacements of the water surface (sloshing); (ii) seismic fragilities used in conjunction with the NUREG-1150 seismic PRA study (Lambright, 1990); (iii) the examination of design details for certain appurtenances such as the refueling gate; and (iv) maximum displacements (vertical and horizontal) of the SFP floors and walls under the applied loads.

Loss of Water from Sloshing

Vertical displacements of the water surface (sloshing) that may lead to the ejection of some water from the SFP depend on the low frequency components of the motions at the base of the SFP. Simplified analytical methods (AEC, 1963; Malhotra, 2000) show that the natural frequency of the sloshing modes for the SFP is of the order of 0.25 Hz for a period of about 4 seconds. The free-field ground motion specified for the scoping study does not have high spectral velocities and accelerations at these frequencies. Consequently, sloshing amplitudes are expected to be small. Deterministic response spectrum analyses with the simplified ANSYS finite element model of the SFP were done to estimate the sloshing amplitude. These analyses used as input the horizontal ISRS at mid height of the SFP (for the frequencies of interest to sloshing) scaled to a damping ratio of 0.5-percent. The results indicate that the sloshing will not exceed about 18 in. Given that the water at the pool is about ½ to 1 foot below the top of the SFP, sloshing is not expected to cause more than 1 foot of water loss. Accordingly, an initial 1 foot decrease in the height of the water is considered at the end of the earthquake event for the subsequent accident progression analysis.

Damage to Refuel Gate, SFP Penetrations, Spent Fuel Assemblies and Racks

Refuel gate - A site visit and examination of the refueling gate structural drawings revealed the following:

- The steel gate next to the water is backed by a similar gate.
- Each of these gates consists of a steel-plated decking with steel stiffeners.
- Each gate has a polymeric seal around its perimeter that is pressed against the concrete by passive mechanical means that are not expected to be lost during the seismic event. Since these are passive mechanical means the effectiveness of the seals does not depend on the availability of AC power.

- Tolerances around the seals are sufficient to accommodate the already small distortions of the biological concrete shielding in the refueling area from the seismic event.

Based on the above, it is concluded that the refueling will not fail under the earthquake and will continue to maintain its intended function during the accident progression.

SFP penetrations - According to the Final Safety Analysis report, there are no connections to the SFP that would allow water to drain below the refueling gate or below 10 ft above the top of active fuel. The FSAR further states that lines below the levels in the previous paragraph are equipped with siphon breaker holes to prevent inadvertent drainage. In addition, the systems for maintaining water quality and quantity are designed so that failure or inappropriate operation of these systems does not cause uncovering of the fuel. In addition, results of the nonlinear finite element analysis also indicate that overall distortions of the pool walls are small (of the order of a few millimeters). These distortions are not expected to lead to seismically induced damage of the penetrations that would leak to potential leakage.

Spent fuel racks and assemblies - Damage to the spent fuel assemblies and racks was not calculated as part of this study. The assumption was made that under the applied seismic loads a coolable configuration would be maintained. This assumption is consistent with the seismic assessments made in conjunction with the resolution of GI-82 and reported in NUREG/CR-5176 (Prassinos et al., 1989). As in the case considered in GI-82, the spent fuel racks for the site considered are allowed are allowed to slide, which tends to reduce the magnitude of the seismic accelerations on the racks and partially decouple their dynamic response from the response of the SFP.

<u>Damage to the Reactor Building and Other Relevant SSCs</u>

According to the fragility analysis for the NUREG-1150 seismic PRA (Lambright, 1990), the median fragility for the reactor building is about 1.6 g. The response of the reactor building structure is expected to be more sensitive to the horizontal ground motions than to the vertical ground motions. Natural frequencies of vibration at these frequencies are about 7 Hz, i.e., frequencies at which the spectral accelerations of the ground motion for the scenario considered are less than those for the ground motions considered in the evaluation of the median fragility. On these bases, failure of the reactor building would not be expected for the seismic scenario considered.

Examination of structural drawings for the Peach Bottom reactor buildings together with a simple kinematic analysis indicates that if the crane bridge were to lose support at one of its as a consequence of the ground shaking, that end of the crane bridge would not fall inside the SFP. Depending on which side would lose support that end might fall a few feet from the pool but not inside the pool.

Loss of offsite power is expected for the seismic scenario considered. Median fragilities for loss of offsite power, in terms of PGA, are less than half the PGA for the seismic motion considered in this study. Review of the fragilities estimated for NUREG-1150 (Lambright, 1990) study indicates a high probability of loss of onsite AC (about 0.84). This high probability estimate is based on either direct failure of onsite emergency diesel generators (sensitive to spectral accelerations around the 20 Hz frequency) or failure of either the emergency service water or the emergency cooling water which provide cooling water for the diesel generators.

References

Ang, A. H-S. and Tang, W. H., (2006), "Probability Concepts in Engineering: Emphasis on Applications to Civil and Environmental Engineering," 2nd Edition, Wiley, 2006.

Ang, A. H-S. and Tang, W. H., (1984), "Probability Concepts in Engineering Planning and Design: volume II - Decision, Risk and Reliability," Wiley, 1984.

AEC (1963), "Nuclear Reactors and Earthquakes," US Atomic Energy Commission (AEC), Reactor Technology Report TID-4500, 22nd Edition, Prepared by Holmes and Narver, Inc., August, 1963.

ASCE (1999), "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," American Society of Civil Engineers (ASCE), ASCE Standard 4-98, 1999.

ANSYS (2011), "ANSYS Mechanical, Release 13," Ansys Inc., Cannonsburg, PA, 2011.

Cherry, J., 1996, "Analyses of Containment Structures with Corrosion Damage," PVP- Vol. 343, Development, Validation, and Application of Inelastic Methods for Structural Analysis and Design, American Society of Mechanical Engineers (ASME), pp. 15-23.

Collins, T. E. and Hubbard, G., (2001), "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, February, 2001.

FHWA (2007), "Evaluation of LSDYNA Concrete Material Model 159," Federal Highway Administration (FHWA), Publication Number FHWA-HRT-05-063, McLean, VA, 2007.

Housner, G. W., (1963), "The Dynamic Behavior of Water Tanks," Bulletin of the Seismological Society of America, Vol. 53, No. 2, pp. 381-387, 1963.

Kanitkar, R., Liu, W. W., Houlahan, G. and Kianoush, R., (2011), "Water flow through crack in thick concrete sections," Concrete International, April, 2011.

Kawamura, Shinichi, (2008), "Investigations and Analyses of The Niigataken Chuestu-oki Earthquake and Kashiwazaki-Kariwa Nuclear Power Station," The Tokyo Electric Power Company, Inc. (TEPCO), June, 2010.

Lambright, J.A., et al., (1990), "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events," NUREG/CR-4550, Vol. 4, Rev 1, Part 3, December, 1990.

LSTC (2007), "LSDYNA® keyword User's Manual," Version 971, Livermore Software Technology Corporation, Livermore, California, 2007.

Malhotra, P. K., Wenk, T., and Wieland, M., (2000), "Simple Procedure for Seismic Analysis of Liquid-Storage Tanks," Structural Engineering International, International Association of Bridges and Structural Engineering (IABSE), 2000.

NEI (2011), "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Nuclear Energy Institute (NEI), NEI Report NEI 07-13, Revision 8P, April, 2011.

NERH (2011), "The Accident at TEPCO's Fukushima Nuclear Power Stations," Report of the Japanese Government to the IAEA Ministerial Conference on Nuclear Safety, Prepared by the Nuclear Emergency Response Headquarters (NERH), Government of Japan, June, 2011.

Newmark, N.M. and Hall, W.J., (1978), "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," NUREG/CR-009, May, 1978.

NRC (1990), "Severe Accident Risks: An Assessment for Five US Nuclear Plants, (Final Summary Report)," NUREG-1150, December, 1990.

NRC (2012a), "Resolution of Generic Safety Issues: Issue 82: Beyond Design Basis Accidents in Spent Fuel Pools (Rev. 3) (NUREG-0933, Main Report with Supplements 1–34)," Available from http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0933/sec3/082r3.html.

NRC (2012b), "Generic Issue Management Control system Report for Fiscal Year 2012 2nd Quarter," Office of Nuclear Regulatory Research, NRC, 2012. (available at http://www.nrc.gov/reading-rm/doc-collections/generic-issues/quarterly/index.html)

Paul, D. D., et al., (1994), "Evaluation and Refinement of Leak Rate Estimation Models," NUREG/CR-5128, June, 1994.

Prassinos, P. G., et al., (1989), "Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants," NUREG/CR-5176, January, 1989.

Sato, Gatu, (2010), "Experience of NCO Earthquake and Restart of Kashiwazaki-Kariwa NPP," First Kashiwazaki International Symposium on Seismic Safety of Nuclear Installations, November, 2010. (Available from http://www.jnes.go.jp/seismic-symposium10/presentationdata/content.html)

5. SCENARIO DELINEATION AND PROBABILISTIC CONSIDERATIONS

5.1. Representative Operating Cycle Characterization

This section captures initial and boundary conditions related to the assumed operating cycle, as well as other related assumptions about the contents and layout of the spent fuel pool. Specifically, Table 15 captures these boundary and initial conditions for the high-density loading configuration and the alternate low-density loading configuration. Information about the operating cycle length and outage length are based on averages of this information for the last 5 operating cycles at the selected plant.

Table 15: Remaining Boundary and Initial Conditions

Item	High-density loading	Low-density loading (if different)
General:		
Operating cycle duration	23 months	-
Rack geometry:		
Support leg height	18.41 cm (7.25 in.)	-
Cell pitch	15.95 cm (6.28 in.)	-
Open vs. closed cell	Closed cell	-
# of storage locations	3819	-
Fuel loading		
Min. assem. during outage[4]	3819 – 764 – 284 = 2771	284*2 = 568
Max assem. during outage	3819 - 764 = 3055[5]	284*3 = 852
# of assem. after outage	3819 - 764 = 3055	284*3 = 852
Newer fuel (< 5 years)	GE14 / GNF2[6]	-
Older fuel (> 5 years)	Actual, based on 2003 info.	N/A
Pattern for "hot" fuel	pre-arranged in 1x4[7]	1x4 "with empties"
Coherent downcomer area	Yes (as is the case in reality)	-

[4] Here it is assumed that a full core offload capability (an industry commitment as opposed to a regulatory requirement) is maintained. Further, it is assumed that 284 assemblies are offloaded each outage (roughly 37% of the core) based primarily on the information in [Exelon, 2011], with a slight change from 270 to 284 assemblies for MELCOR modeling convenience.

[5] 60 of these rack locations may be reserved for storing guide tubes. This situation is not addressed here, but is expected to have a very minor effect on the results. By assumption, these 60 rack cells are filled with very low decay heat fuel, and represent less than 2% of the overall SFP inventory (and less than 2% of the radionuclide inventory available for release).

[6] See [Exelon, 2011] for more information.

[7] See Section 9.6 for a discussion of how the use of contiguous (uniform) arrangements would affect the results.

Item	High-density loading	Low-density loading (if different)
Outage specifications:		
Shuffling vs. full core offload	Shuffling (roughly 1/3 core)[8]	-
Removal of weir gate	2 days (after subcriticality)	-
Start of defueling	2 days	-
Completion of defueling	8 days	-
Start of refueling	14 days	-
End of refueling	20 days	-
Replacement of weir gate	20 days (modeled as 25 days)	-
End of outage	25 days	-
Cycle length	700 days (23 months)	-

The above table includes prescription of a 1x4 storage pattern for the recently discharged fuel, based on regulatory requirements. The plant studied actually exceeds this expectation, in that recently-discharged fuel is stored in a 1x8 pattern. Such a pattern would result in slower heatup of the fuel, were it to become uncovered. Because this arrangement is believed to be highly atypical (relative to the fleet), it is not considered in this study. However, if resources permit, a sensitivity study will be performed to demonstrate the benefit of this arrangement. The figure below shows what the different patterns look like. Section 9.6 provides a discussion of how the use of contiguous (uniform) arrangements would affect the results.

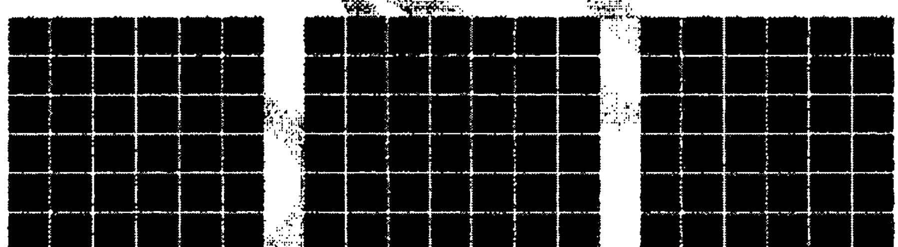


Figure 34: Illustration of SFP Patterns
From left to right: uniform/contiguous; 1x4; 1x8
Red = a recently discharged assembly; Blue = older, lower decay heat assembly

5.2. Operating Cycle Phase Specification

As described in Section 1.5, there are constant changes to the conditions in the spent fuel pool that affect the consequences of a postulated accident (changes in the decay heat, changes in the inventory of fuel in the pool, etc.) As such, it is necessary to discretize this continuous behavior in to a manageable set of discrete quasi-steady snapshots. Further, it must be recognized that the number of quasi-steady snapshots (or Operating Cycle Phases – OCPs as they are termed throughout this report) has roughly a linear scaling effect on the number of MELCOR analyses that must be performed. As such, the definition of the OCPs becomes a minimization / optimization problem (i.e., it needs to minimize the number of OCPs while optimizing the resulting OCPs' accuracy in representing the above pool-reactor configurations / spent fuel loading configurations / decay heat levels).

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8 Note that the decay heat from the fuel left in the reactor is considered when the pool and reactor well are hydraulically connected.

Based on the above considerations, timing associated with the movement of fuel and key changes in plant configuration were combined with the peak assembly and whole pool decay heat curves to arrive at set of five OCPs as outlined in Table 16.

Table 16: OCP Definition for a "Typical" Peach Bottom Operating Cycle

OCP #	OCP Description	Time (days)	% of operating cycle	Pool-reactor configuration	Spent fuel config. for high-density loading	Total decay power	Peak assembly power
1	Defueling of the reactor (~ 1/3 core)	2 – 8	0.9%	Refueling	Contiguous OR 1x4	Existing[2] + (27% of offloaded assemblies) @ 4 days[1]	Highest powered offloaded assembly @ 4 days[1]
2	Reactor T&M / inspection and refueling	8 – 25	2.4%	Refueling	Contiguous OR 1x4	Existing[2] + (offloaded assemblies) @ 13 days[1]	Highest powered offloaded assembly @ 13 days[1]
3	Highest decay power portion of non-outage period	25 – 60	5%	Unconnected	1x4	Existing[2] + (offloaded assemblies) @ 37 days[1]	Highest powered offloaded assembly @ 37 days[1]
4	Next highest decay power portion of non-outage period	60 – 240	25.7 %	Unconnected	1x4	Existing[2] + (offloaded assemblies) @ 107 days[1]	Highest powered offloaded assembly @ 107 days[1]
5	Remainder of operating cycle	240 – 700; 0 – 2	66%	Unconnected	1x4	Existing[2] + (offloaded assemblies) @ 383 days[1]	Highest powered offloaded assembly @ 383 days[1]

1 These times are based on mean decay heat load (as opposed to mean time) during the specified phase (see text for additional discussions); time zero is set to the time of reactor shutdown.

2 Refers to the fuel residing in the SFP at t = 0 (prior to offload).

There are several key assumptions in the above OCP definition that warrant highlighting:

- Offloading of older fuel in to casks (as part of the normal fuel management practices as opposed to an expedited fuel movement program) is not explicitly treated. Rather, a stylized assumption is made that the 284 assemblies that would be loaded in to dry casks during the operating cycle are instantaneously removed from the pool just prior to the outage.

- New fuel is not treated. This fuel would be placed in to the spent fuel pool just prior to the outage (the subject plant does not use a separate new fuel vault). Thus, the fuel would only be present for a very short portion of the operating cycle. During the time that the new fuel is in the SFP, it would affect the amount of zirconium present to participate in a propagating zirconium fire, but would have a negligible effect on the source term[9].

- The calendar time at which the snapshots are evaluated is based on the mean decay heat during the OCP, as opposed to the mean calendar time. Recall that the OCP snapshots are intended to represent a continuous function of possible consequences. While the likelihood of a seismic event occurring is constant in time within one of these OCPs, the consequences associated with the event are not. Furthermore, the exponential decay heat function better represents the change in the post-accident timeline within an OCP than does a linear function, and provides a better mean estimate of the OCP's expected consequences. Therefore, the exponential functional form is used to determine the time within the OCP that is used for the quasi-steady evaluation. In the case of OCP#1, a minor adjustment is made from 4.4 to 3.9 days for modeling convenience (the model is nodalized such that having 88 recently-discharged assemblies can be more readily represented, and 3.9 days is the point at which this many assemblies would have been offloaded given the outage assumptions previously discussed).

5.3. Treatment of Mitigation

This study does not attempt to rigorously quantify the likelihood of successful execution of different mitigative actions that might take place. Rather, this study takes the approach of assuming two different situations with respect to mitigation, each of which is described in the following paragraphs.
The first situation is one in which all onsite mitigative actions are unsuccessful for an extended period of time, meaning that there is no credit for repair or recovery of damaged systems (e.g., offsite power) and no credit for successful deployment of 50.54(hh)(2) equipment. Regarding offsite support for these situations, the study presently assumes the following:

- At 24 hours, offsite support arrives
- At 24-48 hours, ad hoc actions are planned and staged
- At 48 hours, if the fuel is not uncovered and the pool can be reflooded (which is true for the cases with no leak or a small leak), the sequence is truncated.
- Otherwise, the sequence is run to 72 hours in acknowledgement of the additional complexities of accessing the area of the pool when the fuel is uncovered, stopping an ex-containment release-in-progress and/or performing a large leak repair.

Section 5.3.2 provides some further discussion on the rationale for developing results for this situation.

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9 The radioactive material that is of concern during an accident is the fission products generated while the fuel is in the reactor. The UO_2 present in fresh fuel would not contribute noticeably to the source term, and in particular, not in a SFP accident where the temperatures during a postulated accident are lower than those during a reactor accident.

The second situation is one in which (i) mitigative actions associated with the regulatory requirements of 10 CFR 50.54(hh)(2) are successfully deployed, (ii) additional onsite capabilities are used to extend the use of this equipment, and (ii) arrival of offsite resources allows this equipment to be utilized for an extended period of time (i.e.., days) until onsite capabilities can be recovered. Each situation is summarized in Table 17.

Table 17: Summary of Mitigation Assumptions

Item	Situations with successful deployment of onsite mitigation	Situations without successful deployment of onsite mitigation
Normal accident mitigation equipment	Damaged by the event; recovery/repair not credited	
50.54(hh)(2) equipment	Successfully deployed 2 hours after diagnosis	Not credited
Other onsite resources	Successfully deployed to extend operation of 50.54(hh)(2) equipment	Not credited
Offsite resources	Successfully deployed for terminating the accident at 48 or 72 hours (see associated text)	
Emergency Preparedness	Effective	Effective
Mitigation equipment being considered under NRC Order EA-12-049, dated March 12, 2012	Not considered; may be substantively similar to 50.54(hh)(2) capabilities within the context of this study	

The following sections describe the further assumptions made in carrying out this approach, provide the rationale for why the situation without successful deployment of onsite resources is useful from a decisionmaking standpoint, and provide some thoughts as to how future work can use human reliability analysis to go beyond this simplified approach.

5.3.1. Approach Details and Assumptions

For scenarios that credit successful deployment of the 10 CFR 50.54(hh)(2) measures, there are assumptions that must be made about how that deployment is carried out. In general, this study utilizes some of the limits associated with these capabilities that are contained in NEI-06-12, Revision 2 (which has been endorsed by the NRC[10]). For instance, the time at which the mitigative capability is assumed to commence (meaning that it has been deployed and is starting to operate) is 2 hours after diagnosis. The guidance in NEI-06-12, Revision 2 does include a provision that allows for a deployment time of 5 hours after diagnosis for spray, if the fuel has been favorably configured. That provision is not invoked here because the site in question strives to deploy the equipment within 2 hours regardless of the fuel configuration, and the existence of cases without successful deployment of mitigation envelope this effect. The assumption associated with the time to deployment of sprays when the fuel is favorably configured will be addressed as part of a sensitivity study. Future work could be aimed at developing performance-based or timeline-oriented deployment times based on mitigative

[10] This document was originally endorsed for operating reactors by the NRC by letter dated December 22, 2006 [redacted], and this endorsement was carried forward in the Statement of Considerations for the associated rulemaking (see "Power Reactor Security Requirements, Final Rule," published in the Federal Register on March 27, 2009).

capability demonstrations or task analysis. Such work might support an earlier deployment time, or conversely, effects associated with external hazards (for which the mitigative capabilities were not explicitly designed) might suggest later times. Either way, potential upcoming changes to the regulatory requirements along with the scope of this particular study, prompted the guidance-based approach taken.

The flow rates associated with the two modes of delivery considered (spray and makeup) are assumed to be the minimum amounts required (200 gallons delivered per minute for spray and 500 gallons per minute delivered for makeup). For the selected plant, the capacities of the available equipment are higher. The use of 500 and 200 gpm here attempts to account for uncertainties in the speed at which the pumps would actually be run, as well as spray that goes outside the boundary of the pool[11]. As a result, no additional "penalty" is given for inefficiencies associated with spray coverage (i.e., the spray flow rate is applied uniformly across the pool cross-sectional area without further reduction).

As with the time to deployment, timeline-oriented times to diagnosis could have been developed based on task analysis. However, even a detailed analysis would potentially have considerable uncertainty given the nature of the event being studied. Instead, a set of logical, albeit stylized, criteria were established. These were:

- No AC power AND
- Spent fuel pool level decrease by 1.5 meters (5 feet), keeping in mind that 0.5 meters (1.5 feet) is lost due to sloshing, AND
- 30 minute delay associated with manual observation/decision-making

These criteria were developed with consideration of the plant-specific procedures for problems associated with the spent fuel pool, though these specific criteria don't exist in those procedures and they are not intended to represent a specific procedural pathway. It is also important to note that for the plant studied, the various procedures related to loss of spent fuel pool cooling or loss of spent fuel pool inventory do refer plant personnel to the guidelines for use of the 50.54(hh)(2) equipment, even if the cause of the event is not a loss of large area of the plant. More specifically, if control room alarms are available, the loss of inventory would cause an alarm that would direct the operators to a local panel on the refuel floor. The alarm procedure would also start a procedural pathway that would explicitly lead to consideration of the use of the 50.54(hh)(2) equipment. If control room alarms are not available, the special event procedure related to an earthquake directs the operators to inspect the status of the SFP and its cooling systems. The special event procedure also triggers a procedural pathway that would explicitly lead to consideration of the use of the 50.54(hh)(2) equipment.

The above criteria could be conservative or non-conservative depending on what priorities the operators have, and different criteria would clearly be more applicable to other scenarios, namely those that did not include loss of offsite and onsite power at time zero. The assumption that pool elevation must drop 5 feet can lead to long diagnosis time periods for slowly progressing events, thus leading to debatably exaggerated timeline for mitigative action. However, these same slowly developing scenarios are the ones that are least important for

[11] MELCOR does not model the details of the spray delivery from the nozzle(s) to the SFP. Rather, it assumes a uniform flux of water at the top of the SFP. A system flow rate of greater than 200 gpm is necessary to achieve this uniform 200 gpm-equivalent spray flux, to allow for water striking the pool deck or walls and not entering the pool. This inefficiency effect has been accounted for in the regulatory implementation of 10 CFR 50.54(hh)(2).

offsite consequences (i.e., are less severe and less likely to lead to a release). The use of a 2 hour deployment time, as opposed to a 5 hour deployment time allowed in some situations, has a compensating effect for some scenarios.

Regarding implementation mode, for cases where the water level in the pool is greater than 0.9 meters (3 feet) above the top of the racks (a surrogate for high radiation levels on the refueling floor near the edge of the spent fuel pool – see Section 5.4) at the earliest time the sprays/makeup are ready for initiation (i.e., 2 hours after diagnosis), makeup will be utilized. Otherwise, sprays will be utilized. This represents one possible approach to the decision point in Figure 2-1 of NEI-06-12, Revision 2 regarding whether SFP leakage is excessive. In some respects it is a more complicated approach than might be used, but is arguably a more straight-forward approach to enact in the absence of instrumentation. In practice, both approaches end up prompting the same implementation mode for most scenarios studied in this report. The exception is for the "moderate" hole for OCP #1/#2, where (due to the larger volume of water since the SFP is connected to the reactor well and separator/dryer pool), the water level has not reached the 3 foot mark (above the top of the racks) by the time mitigation is deployed. In these cases, makeup is deployed even though the leakage rate actually exceeds 500 gpm. A sensitivity study may be run to address this assumption.

Whichever mode is initiated (spray vs. makeup), it is assumed to be used for the duration of the event (i.e., no later switching to a different mode).

Practically speaking, the above set of assumptions leads to the following process when establishing mitigation timeline boundary conditions in the MELCOR analyses (recall that this only applies for half of the studied sequences since each scenario has an unmitigated calculation):

- Start of calculation / earthquake occurs
- When spent fuel pool level has decreased by 1.5 meters (5 feet), and 30 (diagnosis delay) plus 120 (initial deployment delay) additional minutes have transpired, then:
  - If the water level is > 0.9 meters (3 feet) above the top of the fuel:
    - 500 gallons per minute of makeup in to the top of the pool commences
  - Else if the water level is < 0.9 meters (3 feet) above the top of the fuel (thus indicating excessive leakage) then:
    - 200 gallons per minute of spray at the top of the pool commences

The above assumptions are characterized as being optimistic, relative to the unmitigated (pessimistic case). However, it is important to note that there are aspects of these assumptions that assume failures where they may not occur. For instance, the above set of assumptions only credit a single successful spray/makeup strategy, whereas multiple strategies may be deployed. Along these lines, there are several other ways to recover makeup to the spent fuel pool, several of which have much higher capacities than the mode selected. These alternatives are captured in the Table 10.3.1 of the FSAR, and range from capacities of 25 gpm to 18,000 gpm. For each of the modes capable of delivering more than 200-500 gpm (the mode selected here), these modes require either multiple manual alignments in the vicinity of the SFP and reactor, the availability of AC power for valve manipulations, and/or the use of equipment that might be involved in reactor recovery (most notably a residual heat removal pump). Finally, as mentioned above, the selected set of assumptions does not allow for switching from one mode of makeup/spray to the other.

5.3.2. Rationale for Producing Unmitigated Results

NRC licensees that operate nuclear power plants are required to maintain the facility in a manner that makes the occurrence of a severe accident unlikely. This is achieved through a number of mechanisms involving facility design and operator training, and by applying the concept of defense-in-depth. Even so, there are uncertainties associated with the response to a well-beyond-design-basis seismic event, and its associated effects on the spent fuel pool, which make consideration of unmitigated scenarios prudent from an informed decisionmaking standpoint. Some specific considerations at play for the situation considered in this report include:

- The regulatory requirements for 10 CFR 50.54(hh)(2) equipment are currently focused on use of this equipment for responding to a loss of large area of the plant from explosion or fire. Ongoing regulatory activities related to the NRC's response to the March 2011 accident at the Japanese Fukushima-Daiichi site will alter this situation (e.g., see NRC Order EA-12-049, dated March 12, 2012).

- The large seismic event could damage onsite (and offsite) infrastructure designed to facilitate accident response, as well as causing general disruption at the site.

- If circumstances led to the uncovery of fuel in the SFP, radiation fields on the refueling floor might hamper mitigative actions. Shielding analyses to inform this aspect of the accident analysis are described later in this section. For reference, these analyses suggest projected doses under certain circumstances which could cause personnel implementing 50.54(hh)(2) equipment on the refueling floor of the reactor building to receive doses in excess of 25 rem (a value above which actions can be taken to save lives or protect large populations, on a voluntary basis, as defined in Table 2-2 of EPA 400-R-92-001). For the majority (but not all) of scenarios considered in this report, the above is not the case for the initial deployment of the equipment. Note that as part of the implementation of 50.54(hh)(2), the licensee has committed to an ability to carry out the required mitigative actions even in such situations (e.g., using portable shielding or implementing from a location other than the refueling floor itself).

- A concurrent reactor event, or an ongoing accident at the other unit's SFP, could hamper mitigative actions by reducing accessibility due to radiation fields, impeding accessibility due to other hazards such as hydrogen accumulation, or diverting resources (both personnel and equipment).

- Accessibility could be reduced if an inadvertent criticality event in the SFP were to occur. See Section 9.5 for more information about inadvertent criticality events.

- Identification of human failure events and quantification of human error probabilities for ex-control room actions under beyond-design-basis conditions is not a mature field.

For these reasons, this study is presenting results for cases where accident mitigation efforts are unsuccessful for some period of time.

5.3.3. Considerations for Future Human Reliability Analysis

For schedule and resource reasons, human reliability analysis (HRA) is not included within the scope of this study. In the absence of HRA, no quantitative basis exists for assigning relative likelihood between accident progression scenarios which include or preclude successful deployment of mitigation. The two main roles that HRA could play in a study of this type and scope are:

- The identification and quantification of human failure events that lead to heavy load drops, and
- The quantification of human error probabilities related to mitigative actions.

Both items present a significant challenge, in that no state-of-the-practice exists for either. For the former item, the most relevant recent work is that documented in NUREG/CR-7016 [NRC, 2011b] and NUREG/CR-7017 [NRC, 2011c], which describe qualitative HRA practices applied to dry cask storage operations. Specifically, this work applied aspects of the ATHEANA (A Technique for Human Event Analysis) HRA approach to examine how unsafe actions could lead to a cask drop event or the misload of a cask. The work developed scenarios involving unsafe actions and error-forcing contexts to develop human performance vulnerabilities. The work focuses on two specific casks designs, used at a generic BWR and PWR.

The above work reviewed relevant past information, which included: NUREG-1774 [NRC, 2003b], relevant 10 CFR 21 notifications, Generic Issue 186, NRC inspection reports, NUREG-1864 [NRC, 2007a], and RIS-2005-25, Supplement 1 [NRC, 2007b]. RIS-2005-25, Supplement 1 provides a brief summary of the regulatory history of heavy load drop requirements at US nuclear power plants. For the terminology proposed in that report, the events of interest here are a "cask drop" or "MPC drop" (MPC stands for multi-purpose canister). Other events such as an unplanned descent would cause a less bounding effect on the SFP, and events such as cask hang-ups would only be relevant if they led to a cask drop (as is the case with one of the postulated scenarios). If the current effort proceeds past the single seismic event under consideration to cask drop events, the above work could be used as the launching point for the associated HRA.

With regard to the second main role for HRA stated above, the fact that the mitigative actions of relevance will be outside the scope of both the Emergency Operating Procedures and the Severe Accident Management Guidelines (as well as being outside the current scope of the Extensive Damage Mitigation Guidelines if the event is not treated as a loss of a large area of the plant) is an important consideration. The application of HRA within PRA modeling is fundamentally based on the notion that one of the important drivers for quantification is the time available to take an action, which is in turn reliant on the identification of the specific cue that prompts an action relative to the time at which the action will no longer be effective (or in the worst case might be detrimental) even if it is taken. The treatment of actions outside the scope of the EOPs and SAMGs greatly increases the uncertainty in the quantification of the probability of success/failure of these actions.

In this case, two recent activities may form the logical starting points for any HRA performed in follow-on phases of the SFPSS. These are (i) screening human error probability evaluations performed for incorporating basic events associated with the 10 CFR 50.54(hh)(2) equipment and procedures in to the agency's SPAR models (as a de-activated portion of the model for use in sensitivity studies) and (ii) ex-control room HRA studies, most notably in the area of fire HRA.

[redacted]

~~The latter~~ work is documented in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines." [NRC, 2012] It includes treatment of qualitative HRA, screening HRA quantification, scoping HRA quantification, and detailed quantification. For more general aspects of HRA quantification, the document also refers back to NUREG-1880, "The ATHEANA User's Guide," [NRC, 2007c] and NUREG-1842, "Evaluation of Human Reliability Analysis Methods Against Good Practices." [NRC, 2006]

Additional items that would require attention in the area of HRA are: (i) pre-initiators, (ii) the interrelationship between accessibility, accident progression, and HRA, and (iii) the truncation of consideration of mitigative actions at later times (e.g., 48 hours) due to sequence explosion and/ or very high uncertainties.

5.4. Refueling Floor Dose Rate Analysis Using SCALE

Due to the approach of assuming that deployment of mitigation is either successful or unsuccessful, it is not necessary to take in to account the complexities of accessibility in the determination of mitigation reliability. Even so, an opportunity existed to employ methods and models that were readily available for predicting the radiological conditions on the refuel floor for a range of conditions associated with uncovery of the fuel. Note that the analyses described in this section only account for the radiological conditions stemming from beta and gamma "shine" from exposed radioactive material, not the concern of radiological conditions associated with the release of that material following fuel heatup. It is expected that if a radiological release of fission products from the spent fuel pool commenced, radiation fields in the vicinity of the pool would be very high.

The analyses described, which were performed by the Oak Ridge National Laboratories, looked at a range of conditions. As of this writing, all completed and reviewed analyses are for the "as is" condition, that is an SFP with approximately 3,000 assemblies. The times following discharge that were considered are the same as those associated with the different operating cycle phases. This portion of the analyses is plant specific for Peach Bottom, and utilized 2001 vintage information for representing the fuel design/characteristics in the SFP. Calculations were performed using the ORIGEN and MAVRIC modules of the SCALE code suite. MAVRIC in turn used BONAMI, CENTRM, and DENOVO routines, along with the FW-CADIS methodology. The 1977 flux-to-dose conversion factors were used. The 200 neutron group and 47 gamma group cross sections based on ENDF/B-VII distributed with SCALE 6.1 were used. Results of the analyses can be summarized as follows:

- For water depths of 3 meters (10 feet) above the top of the racks, projected dose rates are very, very low. This is consistent with Regulatory Guide 1.13 which uses this water depth as a conservative measure of adequate shielding.
- Doses for the maximally-exposed location on the refueling floor once the fuel has become uncovered are very high (on the order of 1,000 rem/hr as the fuel first becomes uncovered).
- The point at which projected doses at the maximally-exposed location on the refueling floor surpass 25 rem/hr (a value above which actions can be taken to save lives or protect large populations, on a voluntary basis, as defined in Table 2-2 of EPA 400-R-92-001) is on the order of 0.3 to 0.9 meters (1 to 3 feet).
- Dose rates elsewhere on the refueling floor could be significantly lower than those at the maximally-exposed location.

Additional analysis is ongoing, and this section will be updated to include this analysis once it is complete.

5.5. Discussion of Repair and Recovery

For this version of the report, no attempt has been made to account for repair or recovery of onsite equipment or offsite power. This is a simplifying assumption, and is motivated in part by the lack of quantitative information available to support such a determination, for the large seismic event being considered here. Procedures would direct the operators to attempt to recover failed equipment, and pursue alternate means of establishing AC power such as the ability to obtain AC power from a station blackout cross-tie line to the Connowingo Dam. The assumption here is that onsite and offsite electrical distribution systems are damaged enough by the earthquake so as to significantly delay these recoveries until the 48/72 hour truncation times.

5.6. Scenario Development

5.6.1. Identification of Key Events

The major assumptions that went in to the scenario development, based on the structural analysis documented in Section 1 or other consideration, are:

- All offsite and onsite power is lost as a direct result of the seismic event.
- The 50.54(hh)(2) equipment (when credited) is available for the duration of the event (following delays associated with diagnosis and deployment).
- Initial water loss due to "sloshing" will be 1-2 feet
- Tearing of the SFP liner is not the most probable outcome, but is possible.
- There is no failure of penetrations (including the refueling transfer canal gate).
- The overhead structures (building debris, crane) do not pose a threat to the SFP in terms of failure due to the initiating event.

- Inadvertent criticality, including seismic effects on the integrated poison rack material, are not treated.

The resulting simplified set of key events/conditions is:

1. The initiating event occurs
2. Size of leak (if any) located at the bottom of the pool – 0 (none) or value representing a crack (small) or value representing a propagating tear of the liner (moderate) – these damage states envelope a leak introduced elsewhere in the system (e.g., from the separate/dryer pool during OCP #1 - 2, from the fuel transfer canal gate during OCP #3 – 5).
3. Assembly in a lifted position (only applicable for OCP #1 and #2 – and not explicitly treated)
4. 50.54(hh)(2) mitigation successfully deployed (if credited)
5. Accident sequence is terminated at 48 hours if fuel is still covered at that time, or 72 hours otherwise. (Note that for all simulations considered here, if the fuel is still covered at 48 hours, it does not become uncovered in the ensuing 24 hours.)

Item #3 (assembly in a lifted position) is not explicitly included. At the time of the start of accident progression analysis, there was insufficient information on dose rates in this situation to make assertions about the effect on accessibility. Also, the treatment of optimistic and pessimistic mitigation scenarios envelopes this effect. Finally, the accident progression models used are not well-equipped to model this situation. As a result, the effects of a single assembly becoming exposed and undergoing a zirconium fire will be studied as a stand-alone sensitivity study, if resources permit.

5.6.2. Scenario Calculation Matrices

The following tables show how the combinations described thus far translate to the scenarios considered, for situations during and following an outage.

Table 18: Scenario Table During Outage

Case #	Scenario Characteristics		Radioactive Release Commences Prior to 72 hours?	
	SFP Leakage Rate?	Mitigation?	High-Density Loading – 1x4	Low-Density Loading
1	None	Yes	See later sections of the report for results	
2		No		
3	Small	Yes		
4		No		
5	Moderate	Yes		
6		No		

Table 19: Scenario Table Post-Outage

Case #	Scenario Characteristics		Radioactive Release?	
	SFP Leakage Rate?	Mitigation?	High-Density Loading	Low-Density Loading
1	None	Yes	See later sections of the report for results	
2		No		
3	Small	Yes		
4		No		
5	Moderate	Yes		
6		No		

5.6.3. Refresher on Event Split Fractions

As described previously, the available seismic hazard information has been considered to obtain an initiating event frequency of approximately 1 event in 60,000 years.

Table 20: Refresher on the Seismic Hazard Estimates

Seismic Bin #	PGA Range (g)	Geometric Mean Accel. (g)	Likelihood based on PGA (yr)	Likelihood based on PGA (/yr)	Potential for damage to SFP liner?
1	0.1 to 0.3	0.2	1 in 2,000	$5.2 \cdot 10^{-4}$	Damage not expected
2	0.3 to 0.5	0.4	1 in 40,000	$2.7 \cdot 10^{-5}$	Damage not expected
3	0.5 to 1.0	0.7	1 in 60,000	$1.7 \cdot 10^{-5}$	Damage possible
4	> 1.0	> 1.0	1 in 200,000	$4.9 \cdot 10^{-6}$	Damage possible

Regarding the probability of losing AC power from this particular seismic event, the results described earlier in this report are summarized below.

Table 21: Refresher on AC Fragility

Item	Relative Likelihood	Comments
Loss of offsite and onsite AC	0.84	Direct failure of the onsite emergency diesel generators or indirect failure owing to failure of emergency service water or emergency cooling water (based primarily on information from the NUREG-1150 study).

As also described previously, the structural assessment led to SFP leakage estimates as follows:

Table 22: Refresher on SFP Leakage Conditional Probabilities

Damage State	Relative Likelihood	Comments
No leakage	0.9	Significant damage to concrete; no rupture of SFP liner
"Small" leakage	0.05	Small rupture of SFP liner; drains pool in 10s of hours
"Moderate leakage	0.05	Tearing of SFP liner; damaged concrete limits outflow; drains pool in ones of hours

Finally, since a seismic event is equally likely to happen throughout the operating cycle, the conditional probability for its occurrence during a specific OCP is simply the duration of that OCP divided by the duration of the operating cycle. These weights range from 1% for OCP #1 to 66% for OCP #5 (recall that the OCPs were intentionally "front-loaded" because the outage is when the most change in SFP conditions is occurring):

Table 23: Refresher on the OCP Fractional Contributions

OCP #	Time window / (Time of evaluation) [days]	Fraction of operating cycle	Pool-reactor configuratio n	Spent fuel configuration for high-density loading
1	2 – 8 (5)	0.01	Refueling	2 alternatives considered
2	8 – 25 (13)	0.02		
3	25 – 60 (37)	0.05	Unconnected	Dispersed
4	60 – 240 (107)	0.26		
5	240 – 700 & 0 – 2 (383)	0.66		

The above conditional probabilities are combined, algebraically, to provide likelihoods associated with each of the different sequences treated. At times, sequences are grouped (e.g., those that lead to a release versus those that don't), in order to assign scenario-specific release frequencies, scenario-specific individual risk of a latent cancer fatality, etc. It is important to keep in mind that all such frequencies only consider the particular large seismic event studied in this report.

6. ACCIDENT PROGRESSION ANALYSIS

6.1. Modeling SFPs with MELCOR

6.1.1. Overview and Experimental / Analytical Basis

The MELCOR computer code [Gauntt, 2005] represents the current state of the art in severe accident analysis which has been developed through NRC and international research performed since the accident at Three Mile Island in 1979. MELCOR is a fully integrated, engineering-level computer code and includes a broad spectrum of severe accident phenomena with capabilities to model core heat up and degradation, fission product release and transport within the primary system and containment, core relocation to the vessel lower head, and ex-vessel core concrete interaction.

The MELCOR code is composed of an executive driver and a number of major modules, or packages, that together model the major systems of a reactor plant and their generally coupled interactions. Plant systems and their response to off-normal or accident conditions include:

- Thermal-hydraulic response of the primary reactor coolant system, the reactor cavity, the containment, and the confinement buildings
- Core uncovering (loss of coolant), fuel heatup, cladding oxidation, fuel degradation (loss of rod geometry), and core material melting and relocation
- Heatup of reactor vessel lower head from relocated fuel materials and the thermal and mechanical loading and failure of the vessel lower head, and transfer of core materials to the reactor vessel cavity
- Core-concrete attack and ensuing aerosol generation
- In-vessel and ex-vessel hydrogen production, transport, and combustion
- Fission product release (aerosol and vapor), transport, and deposition
- Behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere such as particle agglomeration and gravitational settling
- Impact of engineered safety features on thermal-hydraulic and radionuclide behavior
- The various code packages have been written using a carefully designed modular structure with well-defined interfaces between them. This allows the exchange of complete and consistent information among them so that all phenomena are explicitly coupled at every step. The structure also facilitates maintenance and upgrading of the code.

MELCOR modelling is general and flexible, making use of a "control volume" approach in describing the thermal hydraulic response of the plant. No specific nodalization is provided, which allows a choice of the degree of detail appropriate to the task at hand. Reactor-specific geometry is imposed only in modeling the reactor core. The MELCOR code has been modernized (source code upgrade to Fortran95) to provide an efficient code structure for ease of maintenance, resulting in the release of MELCOR version 2.1. The new upgraded version of the code architecture supports advancements in computer hardware and software, and the code numerics improvements are underway to carry out reasonable execution times. The input structure for MELCOR 2.1 is completely different from MELCOR 1.8.6. MELCOR is an ideal tool for this type of application because (1) its capabilities have been recently developed and validated for treating spent fuel pool accidents and (2) it is able to model the accident progression, and radionuclide release and in-building transport/retention.

As part of NRC's post-9/11 security assessments, Spent Fuel Pool (SFP) modeling using detailed thermal-hydraulic and severe accident progression models integrated into the MELCOR code were developed and applied to assess the realistic heatup of spent fuel under various pool draining conditions. The analyses was performed for a reference BWR (Peach Bottom) as documented in References [redacted] with additional supporting analyses for separate effects and fluid flow modeling [redacted]. The MELCOR analyses were performed using an earlier version of the code (MELCOR 1.8.5 Version RP) which is no longer maintained. Some of the modeling improvements in MELCOR 1.8.6 include revised modeling of the lower plenum to account for the curvature of the lower head (not relevant for SFP), and formation and convection of stratified molten pools.

MELCOR 1.8.5 Version RP included two modeling enhancements applicable to BWR SFP modeling, (1) a new rack component, which permits better modeling of a SFP rack and (2) a new oxidation kinetics model. The new BWR spent fuel pool rack component permits proper radiative modeling of the SFP rack between groups of different assemblies. The new oxidation kinetics model predicts the transition to breakaway oxidation in air environments on a node-by-node basis. All these models have been integrated into the latest versions of MELCOR (1.8.6 and 2.1). These new SFP features can be used to perform two types of SFP calculations: (1) partial loss-of-coolant inventory accident, and (2) a complete loss-of-coolant inventory accident. A complete loss-of-coolant inventory accident is characterized by the draining of the water to uncover the bottom of the racks leading to air circulation patterns inside the pool and associated air oxidation of the cladding (pre- and post-breakaway) and enhanced Ru release. A partial loss-of-coolant inventory or boil-off accident could involves no or late uncovery of the bottom of the racks. Boil-off of the coolant leads to steam generation and steam oxidation of the cladding and hydrogen generation that could lead to hydrogen combustion.

<u>Breakaway Oxidation Model</u>

Argonne National Laboratory (ANL) has performed oxidation kinetics testing on Zr-based alloys including Zircaloy-4 which is similar to the Zircaloy-2 alloy. The testing showed that air oxidation can be observed at temperatures as low as 600 K. In the tests, a specimen was held at constant temperature and the weight gain due to oxidation as a function of time was measured. The reaction rates for air oxidation are described by parabolic kinetics similar to the ones used to describe steam oxidation. The general form of the equation is,

$$— \qquad (1)$$

where, w is oxide scale thickness or, alternatively, in the MELCOR convention, reacted metal mass. The rate of oxidation was initially steady versus the square root of time at a particular temperature. However, the rate of oxidation increased after some time and persisted for the remainder of the test. The ANL pre- and post-breakaway Zr-4 oxidation correlation are provided below.
Steam pre-oxidized, wide-temperature pre-breakaway Zr-4 oxidation correlation [Natesan,2004]

(2)

Steam pre-oxidized, wide-temperature post-breakaway Zr-4 oxidation correlation [Natesan,2004]

(3)

The new oxidation model was implemented in MELCOR by adding a breakaway lifetime calculation. The model calculates an oxidation "lifetime" value for Zircaloy components in each cell using the local Zircaloy cladding temperature.

$$— \qquad (4)$$

$$(5)$$

$$(6)$$

where P_{Lox} is the MELCOR fit of the timing for the transition from pre-breakaway to post-breakaway oxidation reaction kinetics for Zirlo and Zircaly-4 in the ANL experiments.

The air oxidation model was benchmarked against experimental data from the Sandia National Laboratories SFP facility [redacted]. The calculations with and without breakaway oxidation kinetics showed different heatup rates following breakaway. Both the data and the calculation with breakaway kinetics show a sharp increase in the heatup rate following breakaway. The new breakaway kinetics model provided a better prediction of the measured data including a transition to accelerated post-breakaway oxidation kinetics.

Hydraulic Resistance Model

MELCOR modeling approach for flow paths connecting control volumes includes constitutive relationships to specify form losses (i.e., minor losses) and wall friction losses (i.e., major or viscous) along a flow path as a hydraulic flow loss term to the momentum equation. The format of the user-specified input for MELCOR is defined from the sum of the local viscous and major pressure drops,

$$— \quad — \qquad (7)$$

where ρ is the fluid density, v is the fluid phase velocity, L is the inertial flow path length, D is a representative hydraulic diameter, and K is the form loss coefficient. The laminar friction factor (f) is given as:

$$(8)$$

where S_{LAM} is a user-specified MELCOR input parameter, Re is the Reynolds number ($\rho vD/\mu$), and μ is the fluid dynamic viscosity.

Hydraulic resistance measurements were performed on a Global Nuclear Fuel (GNF) 9x9 BWR assembly at Sandia National Laboratories [Durbin,2005] to obtain the required frictional and form loss coefficients including the effects of grid spacer and partial rods. The empirical loss factors were used directly to model the flow path resistance parameters in the security assessment analysis [redacted] where the axial nodalization of the core cells roughly corresponded to various regions in the experimental setup.

6.1.2. Heat Transfer Modeling Within the SFP and to Surrounding Walls

The MELCOR core models calculate the thermal response of the core. The core is nodalized into a number of axial levels and radial rings (each ring represents a collection of assemblies). All important heat transfer processes are modeled in each core cell including thermal radiation within a cell and between cells in both the axial and radial directions as well as radiation to boundary heat structures. Each core cell is hydraulically interfaced to a control volume to obtain the necessary boundary conditions (e.g., water level, flow velocity) and in turn supplies the calculated heat and mass transfer to the control volume. Each core cell may contain a number of components including fuel, cladding, canister (BWRs), and other structures (e.g., control rods).

The new spent fuel pool rack component permits separate modeling of the SFP rack and radiative heat transfer between the rack and existing components in the core. The new air oxidation kinetics model predicts the transition to breakaway oxidation kinetics in air environments on a node-by-node basis. The SFP racks and the lower gap region below the SFP racks can be modeled using the existing core and lower plenum components. The MELCOR core model is designed in two-dimensional cylindrical geometry and nodalization of the SFP must fit within this framework. Implicit in this framework is the assumed direction of heat and mass transfer between adjacent rings and adjacent elevations. For SFP models, the user can take advantage of this pre-existing framework and arrange the fuel-rack cells in a similar ring pattern.

The heat transfer paths modeled within the core are appropriate for conventional commercial light water reactors. The capability has been added to define arbitrary ("generalized") additional heat transfer paths between core components to allow for more flexible intra-cell radiation or conduction, but the user is responsible for defining a single input parameter that captures the geometry of the heat transfer path. The heat transfer paths within a ring and also across a ring boundary is depicted in Figure 35. For radiation between different core rings, the user adjusts the view factors and the surface areas.

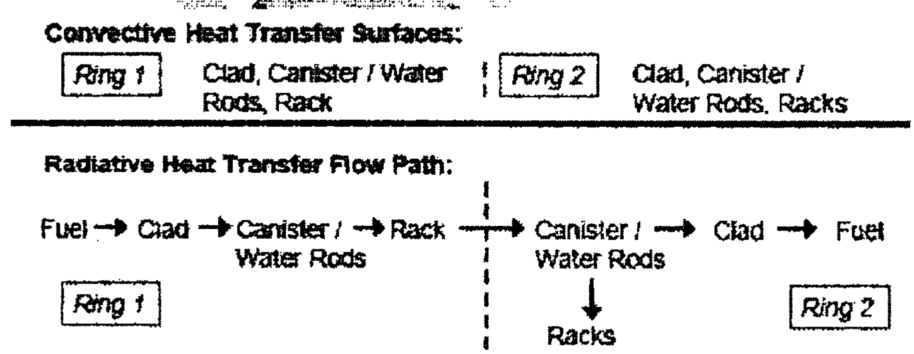

Figure 35: MELCOR modeling of heat transfer paths

The core models radiative heat transfer from the outermost ring components (if present) to the core boundary specified as a heat structure. The SPF wall is modeled as a heat structure composed of a steel liner and concrete which can receive radiative energy from the core as well as convective heat transfer from the adjacent control volume.

6.1.3. Modeling of Mitigative Sprays

The MELCOR containment spray model was used to calculate the thermal response of the fuel for the mitigated scenarios involving spray activation. The spray model mechanistically models the interaction of the spray droplets with the containment atmosphere, and includes droplet heat and mass transfer and fission product removal capabilities. A droplet size of 1250 microns was used for all calculations. The spray was positioned at the top of the spent fuel pool (elevation of the refueling bay) allowing the droplets to be directed into the assemblies and open spaces based on their respective cross-sectional areas.

The penetration of the spray water into the assembly is controlled by the interphase momentum model, which replicates the Wallis flooding curve. Once the spray water enters the assembly, the spray is assumed to form a thin film on the fuel structures in the assembly, which drains downward. The MELCOR simplified flow regime model identifies the spray flow as a film in contact with the fuel rods (see Figure 36). Heat transfer takes place between the fuel rods and water in core cells where the flow regime model is active. Nucleate or film boiling heats the water film to saturation conditions as it drains down the assembly. Simultaneous heat transfer from the rods and surrounding gas causes the spray flow to boil. The spray film travels downward in contact with the fuel rods until the local control volume void fraction becomes greater than 99.8% (i.e., $\alpha > 0.998$). Due to numerical considerations, the residual water is converted into a shallow pool where the liquid heat transfer area is apportioned by the depth of the pool in the control volume. Typically, the remaining water boils away in the first core cell after the flow regime model is disabled.

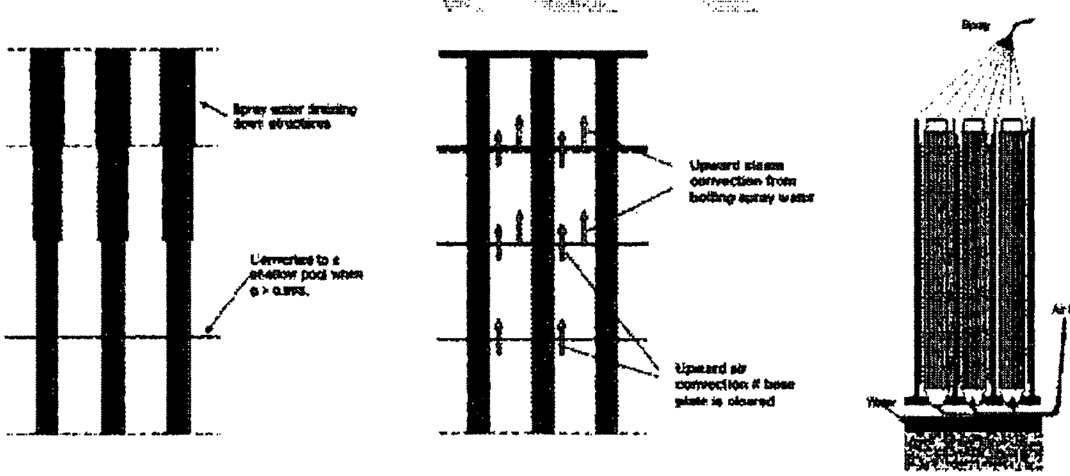

Figure 36: Spray model for SPF analysis

It should be noted that the MELCOR thermal hydraulic model interprets the liquid film as a small pool at the bottom of each control volume (see Figure 36). Due to the high void fraction, the phasic resistance of the steam or air flowing through the pool is relatively insignificant, which is the expected impact of a liquid film. Similarly, the depth of the spray water penetration is controlled by the heat transfer rate from the fuel rather than the momentum solution. Axial, stepwise heat transfer from the core cells limits how far the spray water penetrates into the assembly. A possible limitation of the thermal hydraulic representation is the relatively small heat transfer area between the two phases (i.e., heat transfer through the pool and the surface versus a film). However, the rate of heat transfer from the gas to the water film is minor in

comparison to the nucleate and film boiling heat transfer on the surface of the fuel rods. A detailed nodalization is used to track the water as it penetrates into the assembly that permits a better local representation of the fluid conditions and the location of the spray dryout. Parametric calculations will be performed to show the impact of this modeling parameter (i.e., flow regime model active or inactive).

6.1.4. Modeling of Fuel Collapse and Baseplate Failure

Fuel collapse is based on user defined cumulative fuel damage fraction logic, where the fuel failure time is defined as a function of cladding temperature and only applied if the unoxidized zircaloy cladding thickness is less than 0.0001 m. The failure logic calculates the fuel damage fraction for the current timestep, if the unoxidized cladding thickness criteria is met, and adds that fractional damage to any previously calculated damage. When the cumulative fuel damage fraction exceeds unity, the fuel is failed in the SFP MELCOR model. This lifetime damage model eliminates the threshold behavior present in the other fuel failure criteria and predicts accumulating damage if the fuel remains above the melting temperature of Zircaloy but below the absolute threshold collapse criteria of 2500 K.

All components other than fuel rods (fuel and cladding) will be immediately converted to particulate debris whenever the unoxidized metal thickness is reduced below a user-defined minimum value. The minimum thickness criterion for the two MELCOR canister components is 0.0001 m. The unoxidized metal thickness is reduced both by oxidation and by melting and candling of metal. Molten Zircaloy held up by an oxide shell is released from the fuel rods at 2400 K and from the canister at 2100 K (i.e., just above the melting temperature of the Zircaloy). Particulate debris will be formed for canister components following the release of the molten Zircaloy or if the temperature of the component reaches the melting temperature of the associated oxide.

Baseplate failure is defined by the grid-supported or egg-crate plate model in MELCOR. In general, the beams that form the grid have sufficient strength that their failure is not an issue, and the interest is in failure of the web between them. Upon failure of the plate, the capability to support particulate debris or intact components is lost; however, the plate will remain in place until it melts. This model calculates baseplate failure based on the maximum stress in a plate of user defined thickness supported by beams of user defined spacing with a total load on the area of the ring. In the SFP model, the thickness of the baseplate is defined as 0.0127 m with grid spacing of 0.07 m. The melting temperature of the plate is 1700 K.

6.1.5. Radionuclide Transport Modeling and Treatment of Hydrogen

In MELCOR, the release and transport of fission product vapors and aerosol (referred to as radionuclides) is modeled by the RN package. Release of radionuclides can occur from the fuel-cladding gap by exceeding a failure temperature criterion or losing intact geometry, from material in the SFP using various empirical release correlations based on fuel temperatures. After release to a control volume, masses may exist as aerosols and/or vapors, depending on the vapor pressure of the radionuclide class and the volume temperature.

Aerosol dynamic processes and the condensation and evaporation of fission product vapors after release from fuel are considered within each control volume. Aerosols can deposit directly on surfaces and water pools, or can agglomerate and eventually fall out by gravitational settling. Aerosols deposited on surfaces can be vaporized (if volatile) but cannot currently be resuspended in MELCOR. All deposition mechanisms are mechanistically modeled. Aerosols and vapors are transported between control volumes by bulk fluid flow of the atmosphere and the pool.

For tracking purposes, the radionuclides are combined into material classes, which are groups of elements (and their isotopes) with similar chemical and transport behavior. Radionuclide masses include both the radioactive and non-radioactive mass to properly model the transport of fission products. In the SFP MELCOR model, there are fifteen default material classes and two user defined classes to model the behavior of cesium iodide and cesium molybdate as shown in Table 24.

The fuel release model is based on the CORSOR-Booth model that more accurately predicts the release rates from the Phebus and VERCORS experiments [Gauntt,2010]. The default MELCOR radionuclide package input was modified to accommodate new insights from the Phebus experimental program. The cesium, iodine, and molybdenum radionuclide classes were reconfigured as follows,

Class 4 – Characteristic released compound is I_2 with the default inventory wholly transferred to Class 16.

Class 7 – Characteristic released compound is Mo with the default inventory reduced by the amount allocated to Class 17.

Class 16 - Characteristic released compound is CsI with the default inventory representing all of Class 4 and sufficient Cesium from Class 2 to form CsI.

Class 17 – Characteristic released compound is Cs_2MoO_4 using the remainder of the cesium not in the gap (already included in Class 2) or not already combined with the iodine in Class 16. Sufficient molybdenum is included from Class 7 to Class 17 to form Cs2MoO4. The released vapor pressure and compound mass is consistent with Cs_2MoO_4.

An approach for the estimation of increased Ruthenium release under air-oxidation conditions is proposed in [Gauntt, 2010]. Ru (class 6) has the lowest of vapor pressures in the default MELCOR model that prevents prediction of large releases[12]. There is evidence of higher volatility of ruthenium oxides (many orders of magnitude higher than the default MELCOR). It is assumed [Gauntt, 2010] that there is always air present leading to formation of a moderately hyper-stoichiometric fuel ($UO_{2.15}$) and release of RuO_2. The default vapor pressure parameters in MELCOR are adjusted for Ru class to match RuO_2 vapor pressure at 2200 K[13]. The new Ru release model is applied only to scenarios involving rapid drain down (for moderate leak rates) of the SFP pool. These cases lead to relatively early clearing of the rack baseplate and flow of air (and possibly steam) through the assemblies. It should be noted that the model does not take into account the concentration of oxygen or steam during the oxidation process.

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[12] There is a mass transfer limitation to the release from the fuel.

[13] The rationale for increased Ru class release is based only on increased vapor pressure, and requires further experimental validation.

Table 24: MELCOR Radionuclide Class Composition

Class #	Class Name	Representative	Member Elements
1	Noble gases	Xe	He, Ne, Ar, Kr, Xe, Rn, H, N
2	Alkali Metals	Cs	Li, Na, K, Rb, Cs, Fr, Cu
3	Alkaline Metals	Ba	Be, Mg, Ca, Sr, Ba, Ra, Es, Fm
4	Halogens	I	F, Cl, Br, I, At
5	Chalcogens	Te	O, S, Se, Te, Po
6	Platinoids	Ru	Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni
7	Early Transition Elements	Mo	V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W
8	Tetravalent	Ce	Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C
9	Trivalents	La	Al, Sc, Y, La, Ac, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf
10	Uranium	U	U
11	More volatile main group	Cd	Cd, Hg, Zn, As, Sb, Pb, Tl, Bi
12	Less volatile main group	Sn	Ga, Ge, In, Sn, Ag
13	Boron	B	B, Si, P
14	Water	H_2O	H_2O
15	Concrete	-	-
16	Cesium Iodide	CsI	CsI
17	Cesium Molybdade	Cs_2MoO_4	Cs_2MoO_4

The gap inventory is specified in Table 25 based on NUREG-1465 [NRC,1995]. It should be noted that in NUREG-1465, it is stated that for accidents where long term cooling is maintained (e.g., postulated spent fuel handling accident), the gap release could be as low as 3%. However, in the unmitigated scenarios in this work, the fuel experiences prolonged high temperatures (and even failure in some instances). Therefore, in the present work, it is conservatively assumed that 5% applies to all scenarios.

The decay heat calculation was based on security assessment analyses [redacted] that used a decay heat program provided by the licensee. The decay heat power is calculated based on the discharge time and other parameters, such as the fuel burn-up and power history. The utility provided the program and the appropriate input files for the SFP configuration after their last offload (i.e., September of 2001) to perform decay heat calculations. Consequently, the decay heat power of every assembly was calculated as a function of time from reactor shutdown[14]. The decay heat and radionuclide package for MELCOR was conceived for reactor analysis. Therefore, all assemblies are assumed to have the same shutdown time. MELCOR calculates the initial fission product inventory from tables of inventories and specific decay power for 29 elemental groups. The elemental decay heat is normalized per unit of mass of the element and stored as a function of time after shutdown.

[14]An interpolation scheme was used to calculate the individual assemblies decay power at different times relevant to this study (the error in interpolation is typically less than 1%). Since the number of old assemblies was increased by 60 (3055 total in the pool), the decay heat for these assemblies was assumed to be an average of the older assemblies.

Table 25: Radionuclides gap inventories

Class #	Gap inventory	Class combination
1	5%	-
2	100%	Characteristic released compound is CsOH with the default inventory wholly representative of the Cesium in the fuel gap except what is already included in Class 16. Required amount of cesium not in gap of Class 16 to yield a 5% total cesium gap inventory.
3	1%	-
5	5%	-
16	5%	5% of the Class 16 inventory to yield 5% of the total Iodine inventory in the gap

Since SFP accident calculations involve fuel assemblies with multiple shutdown times, the following procedure was used to implement the batch-average decay heat results. First, the effective reactor operating power was estimated using SFP inventory burn-up. The effective operating power was calculated as the total burn-up of all assemblies in the SFP (GWD/MTU) divided by the average assembly metric tons of uranium (MTU) and the total number of days of criticality. Based on the effective operating power, MELCOR calculates the specific element, time-dependent decay heat tables and mass inventories. The aging time in the specific element decay heat tables is specified as the scenario time minus the shutdown time of the assemblies in the most recent offload. Next, the above results for element inventories (kg) times the specific element decay heat (W/kg) at the scenario time are scaled to match the total SFP decay power. This scaling procedure addresses any limitations in the relatively long-term decay heat power in the MELCOR data base. Finally, inventory scaling coefficients are used to partition the decay heat amongst the various MELCOR rings. In summary, the batch-average decay heat is explicitly conserved but the fission product inventory is not properly scaled to account for differences in the various assembly discharge dates. A post-processing routine is implemented that uses the MELCOR predicted release fractions along with actual inventories calculated for each batch.

To accommodate consequence calculations using MACCS, an extensive control system was written in MELCOR input file that tracks the fission product releases from each ring[15] and the subsequent release to the environment. Time-dependent, non-dimensional environmental release fractions are calculated for each batch (i.e., MELCOR ring) that can be multiplied by the specific batch fission product activities to evaluate the environmental source term. The following procedure was used to map the releases from MELCOR to MACCS. MELCOR activity release for each isotope (e.g., m = Cs137, Cs134, Cs136 for class 2) is given by:

(9)

MACCS activity release is given by:

(10)

[15] Ring is a collection of assemblies in the MELCOR radial nodalization.

where is defined as:

(11)

(12)

Where

r = ring number (total 6 rings)
m = radionuclide (1:M) where M is the number of ORIGEN-s isotopes in each class
t = time since start of event
RF (t) = Environmental release fraction (ring by ring from MELCOR)
= Environmental release fraction (by radionuclide group for MACCS)
A = Released Activity (Bq)
A^{o} = Initial inventory (Bq) from ORIGEN-s (69 isotopes for each MELCOR ring)

Radionuclide Inventories

The radiological inventories and decay heat for assemblies in the spent fuel pool (SFP) were calculated using information provided by the utility for all assemblies discharged to the pool through Cycle 18 (September, 2011). The information included the assembly ID, design type, initial enrichment, discharge burnup, and discharge date. The analysis basis for the high-density SFP inventory was 3055 assemblies, a number based on the pool capacity of 3819 assemblies, reduced by 764 assemblies to accommodate a full core offload capability. Information on assemblies discharged prior to Cycle 7 is not considered since the target pool inventory was achieved with the assemblies from Cycles 7 to 18.

Assembly depletion and decay calculations were performed using the ORIGEN code [Gauld, 2011], maintained within the SCALE nuclear safety analysis code system [Rearden, 2012]. The nuclear cross-section libraries used for the burnup analysis of the assemblies were those distributed in SCALE 6.1. These libraries are developed using ENDF/B-V cross sections and include representative 7×7, 8×8, 9×9, and 10×10 GE assembly designs [Ilas ,2006] used in the Peach Bottom reactor. ORIGEN calculations performed using these libraries have been validated against experimental destructive assay measurements, and calorimeter measurements of assembly decay heat have been demonstrated in previous validation studies[Ilas ,2008) to be accurate within ±2%.

For the burnup analysis, the irradiation and decay history for each of the 3055 assemblies in the pool was simulated explicitly using ORIGEN and assembly-specific design and operating history data provided by the utility. Each assembly was decayed to a reference date corresponding to the end of Cycle 18 (9/11/2001), and the assembly inventories combined into analysis groups. The groups were then further decayed to calculate spent fuel assembly activities and decay heat power for analysis cooling times of 3.6, 3.9, 5.0, 13.1, 37.0, 107.0 and 383.0 days after shutdown of the reactor. The assemblies were grouped according to the cycle they were discharged.

- Group 1 (268 assemblies from Cycle 18)
- Group 2 (272 assemblies from Cycle 17)
- Group 3 (272 assemblies from Cycle 16)
- Group 4 (276 assemblies from Cycle 15)
- Group 5 (284 assemblies from Cycle 14)
- Group 6 (1683 assemblies from Cycles 7 to 13)

This division of assemblies by group facilitated use of the data for an analysis of a low-density SFP configuration, whereby all assemblies with a cooling time greater than five years have been removed from the pool. For the present analysis, each offload was assumed to be 284 assemblies for modeling convenience and to avoid modifying the MELCOR model nodalization[16]. Therefore, the actual inventories from batches were scaled appropriately to correspond to the rings in MELCOR nodalization. For example, for the low density case, the Cycle 18 inventories were increased by 284/268 and the sum of Cycles 16 and 17 were scaled as 568/(272+272) resulting in 852 assemblies as opposed to the actual 812.

The spent fuel pool results were compiled for each assembly group and all decay times and included activities (Bq) for 69 radionuclides and decay heat (watts).

Results from the present analysis were compared with those generated previously for the Peach Bottom pool using assembly data provided by the utility through 2001 (assemblies discharged up to Cycle 13) as part of the security assessment work. A limitation of the 2001 data was that the utility did not provide the actual discharged burnup distribution of assemblies from Cycles 12 and 13. Consequently, previous analyses assumed burnup distributions for these cycles based on data from Cycles 10 and 11. Review of the actual burnup distributions included in the 2011 data, indicates that the average discharge burnup increased significantly after Cycle 12. The burnup values used in the present Peach Bottom analysis are significantly higher and therefore more representative of modern spent fuel pool inventories than earlier analyses. Previous analyses using the 2001 data are representative of discharged fuel up to about 1995.
Other differences are attributed the specific power of the assemblies which influences the decay heat power and activities of short-lived fission products in the analysis time range. Information on the specific power was not provided by the utility. Notwithstanding power up-rates for the Peach Bottom reactor, the most recent occurring in 2002, the specific power used to calculate inventories for the assemblies in the present analysis were lower than those assumed using the earlier 2001 data. The present analysis normalized the average specific power of the discharged assemblies to the reactor specific power. Previous information provided by the utility in the 2001 data included the effective full power days, used to derive slightly higher specific power values compared to those used in the present study.

The net impact of differences between the analyses performed using 2001 data and the present analysis is an increase in the inventories for cooling times longer than about 30 days, attributed to higher assembly burnup in the 2011 data. For shorter cooling times the previous analyses predicted decay heat rates about 5% larger than the current results, likely due to more conservative estimates of specific power used in the previous analyses. A comparison of the present decay heat results with values calculated by the utility in 2001 show agreement to better than 3% over all cooling times, with present results being slightly larger than utility values likely due to the increase in discharge burnup since 2001.

[16] The nodalization was based on the security assessment work up to Cycle 13. The additional data on later cycles were received after the MELCOR model had been developed and the calculations were started.

Hydrogen Burn

A burn is initiated in a control volume if the mole fraction of the reactants (hydrogen and oxygen) satisfies the burn criteria. In addition, control volumes that are specified to contain igniters are tested against different criteria than control volumes without igniters. In SFP calculation, ignition is assumed to occur in the reactor building when the hydrogen concentration exceeds 10% by volume. In addition, MELCOR checks to determine whether there is sufficient oxygen. The minimum oxygen mole fraction for ignition is 5%. The maximum diluents mole fraction for ignition (mole fraction of steam plus mole fraction of carbon dioxide) is 55%. If all these conditions are satisfied, a burn is initiated. There may be some uncertainty regarding the combustion of hydrogen, especially with regards to the timing of a spontaneous ignition. It is recognized that a hydrogen burn may occur at higher or lower concentrations of hydrogen, air, and steam that have both epistemic and aleatory uncertainties. Many SFP calculations resulted in conditions where combustion was very likely or very unlikely. Consequently, the results of cases with and without combustion are presented in the SFPSS. However, there are some cases that have conditions where the occurrence or timing of a combustion event has more uncertainty that were assumed to ignite or not ignite according to the default spontaneous combustion criteria in MELCOR

6.2. Description of MELCOR Models

The spent fuel pool, 40 feet wide by 35.3 feet long by 38 feet deep, is located on the refueling floor of the reactor building. The pool is constructed of reinforced concrete with a wall and floor lining of 1/4-inch thick stainless steel. The walls and the floor of the spent fuel pool are approximately 6'. In the northeast corner of the SFP is a cask area of 10' square.

The high density SFP racks provide spent fuel storage at the bottom of the fuel pool. The fuel storage racks are normally covered with about 23 ft of water for radiation shielding. The SFP racks are freestanding, full length, top entry and are designed to maintain the spent fuel in a spaced geometry, which precludes the possibility of criticality. The high-density SFP racks are of the "poison" type utilizing a neutron absorbing material to maintain a subcritical fuel array. The racks are rectilinear in shape and are of nine different sizes. A total of 3819 storage locations are provided in the pool. The racks are constructed of stainless steel materials and each rack module is composed of cell assemblies, a base plate, and base support assemblies. Each cell is composed of (a) a full-length enclosure constructed of 0.075" thick stainless steel, (b) sections of Bisco Boraflex, which is a neutron absorbing material, and (c) wrapper plates constructed of 0.020" thick stainless steel. The inside square dimension of a cell enclosure is 6.07". The cell pitch is 6.28". The base plate is made from 0.5" thick stainless steel with 3.8" chamfered through holes centered at each storage location, which provides a seating surface for the fuel assemblies. These holes also provide passage for coolant flow.

Each rack module has base support assemblies (i.e., 'rack feet') located at the center of the corner cells within the module and at interior locations to distribute the pool floor loading (e.g., see Figure 37). Each base assembly is composed of a level block assembly, a leveling screw, and a support pad. The top of the leveling block assembly is welded to the bottom of the base plate. SFP fuel cells are located above each rack foot. Four 1" holes are drilled into the side of the support pad. The interior of the support pad is hollow and permits flow to the opening in the base plate. The square tube cells are used to construct the rack cells, which results in an equal number of cells resulting from the square tube cell checkerboard layout. Figure 37 shows the

layout of the rack cells. There is the potential for lateral cell-to-cell flow between connected rack cells.

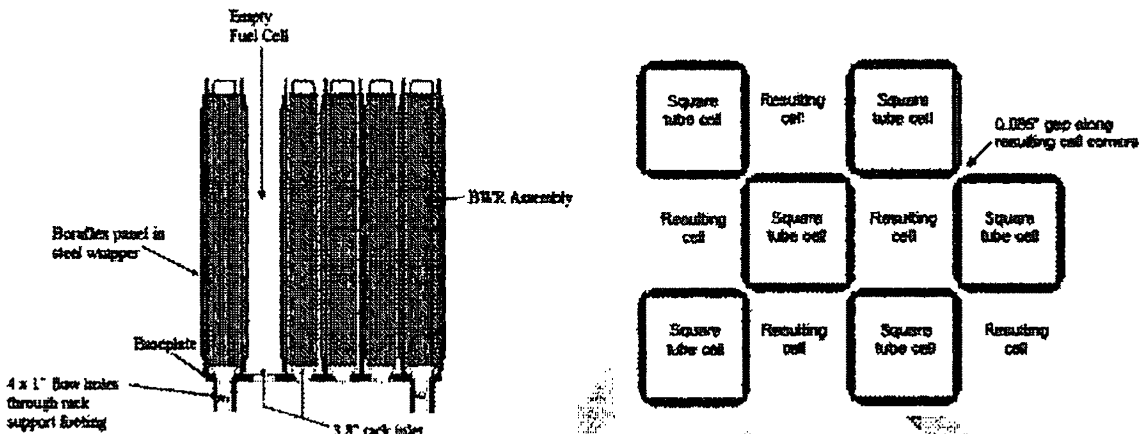

Figure 37: Typical spent fuel pool rack cut away cross sections

Figure 38 shows the control volume (CV) nodalization of the SFP region of the whole pool model. The bottom of the pool was divided into eight regions. CV299 represents all the open regions in the SFP around the racks and including the cask area. The racks are subdivided into the other 7 regions. Ring 7 (CV170 and CV171) represents the empty rack cells on the periphery of the SFP. All the assemblies in the SFP are located in Rings 1 through 6. Each ring with assemblies is further subdivided into 19 control volumes (CVs); a CV below the racks and 9 CVs inside the canister and 9 CVs in the bypass region between the rack and canister. For example, CV110, CV111 through CV119, and CV211 through CV219 represent the region beneath the rack, the region within the canister, and the bypass region between the rack and canister, respectively (see Figure 39). Similarly, core Rings 2 through 6 contain similar canister and bypass region nodalizations. The region above the pool is divided into two control volumes. Typically, flow goes down CV301 and CV299 and rises through CV300. The flow enters the bottom of the racks through CV110 through CV170. For low density configurations, the CV nodalization does not contain a bypass region (between the channel box and rack) as shown on the right hand side of Figure 39.

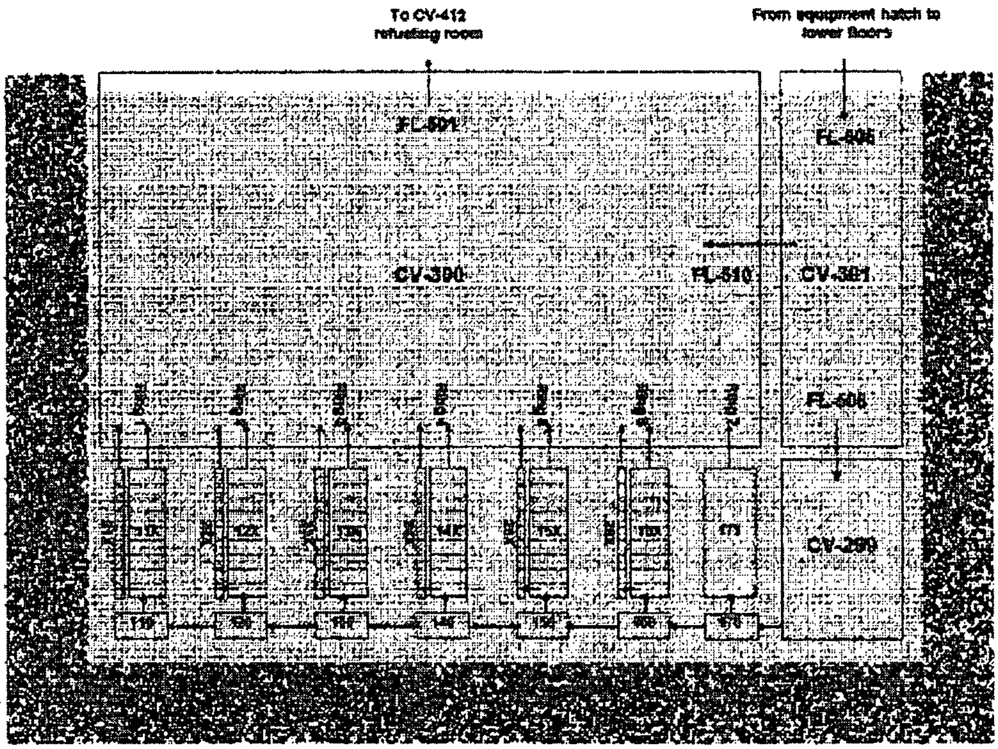

Figure 38: MELCOR nodalization of the whole pool high density model

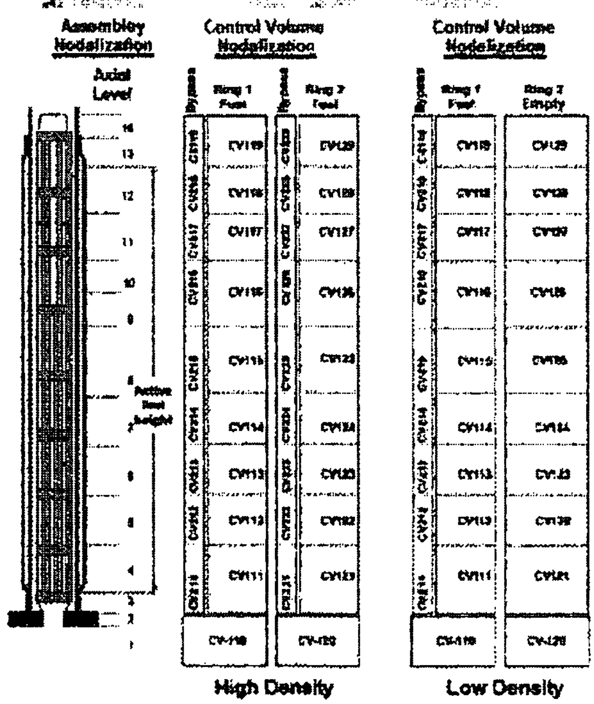

Figure 39: MELCOR nodalization of the assemblies (only two rings shown)

The hydraulic resistance was specified using the results from the SNL experimental test program [Durbin, 2005][17]. For example, for the flow path connecting CV113 and CV114 in the fully populated region, the MELCOR input values included a loss coefficient (K) of 3.8, and a friction factor (S_{LAM}) of 31.3 (=125/4 since MELCOR uses the fanning friction factor definition). The flow resistance under the racks was represented using typical contraction inertial loss coefficients and viscous losses consistent with a flow length to the center of the SFP. The BWR assembly canister is modeled with the MELCOR canister component. The rack walls are modeled with the new rack component with stainless steel and Boraflex materials. MELCOR does not include an option to model the two large water rods in the center of the assembly. Consequently, the water rod mass and surface area was included in the canister wall.

The axial channel and bypass wall blockage models were active and controlled the resistance in the respective flow paths. The blockage model monitors the porosity of the materials in the channel and bypass regions. If a debris bed forms, the flow resistance is adjusted via an Ergun flow resistance model. The canister wall radial blockage model controls flow paths between the bypass region and the assembly. Initially, the canister wall precludes flow. However, if the canister fails, a radial flow path is activated that permits flow between the two regions. Similar to the axial blockage model, the flow resistance is adjusted based on the local debris porosity.

A complete reactor building has been developed for Peach Bottom [NRC, 2012]. However, the bulk of the reactor building does not play a significant role in SFP accidents. Consequently, the reactor building model was simplified to only model the refueling room (i.e., within the red dashed line in Figure 40).

A single control volume models the refueling bay. An open hatch in the southeast quadrant connects (via a flow path) the refueling room to a boundary condition volume representing the flow connection to the lower sections of the building. The nominal reactor building leakage is modeled at the center elevation of the refueling bay and the leakage flow from elevations in the simplified model from the lower regions was tuned to match the leakage flow rate of a detailed reactor building model.

The detailed reactor building model simulated many over-pressure failure flow paths within the reactor building. The two most important flow paths were included to the simplified refueling floor model, (1) the blowout panels on the refueling room walls and (2) a pathway representing the structural failure of the reactor building roof. The refueling room blowout panels will fail if there is an over-pressure greater than 1720 Pa (0.25 psig). If the reactor building pressure rises above 3450 Pa (0.5 psig), a structural failure of the reactor building roof will occur.

MELCOR does not include models for stratification of hot gases. Each control volume is assumed to be well-mixed and have a single temperature. Large-scale natural circulation flow patterns can be predicted where the bulk temperature differences between adjacent rooms create mixing flows. However, it would be awkward or perhaps impossible to predict complex plume behavior within regions typically modeled with a single control volume (e.g., the room above the SFP). Consequently, the MELCOR calculations are expected to over-predict the amount of thermal mixing within the building. Based on insights from the Computational Fluid

[17] In the present study, the assembly nodalization is based on the GE14C 10x10 [NRC, 2012] to account for the latest offloads used in the low density configuration. Both 9x9 and 10x10 configurations have partial fuel rods. The flow area for each assembly is reduced by about 4% compared to 9x9 design. The hydraulic resistance data are assumed to apply. Frictional loss coefficient for a 10x10 array could be somewhat different since it is a function of hydraulic diameter and grid spaces design.

Dynamics (CFD) calculations [redacted], the MELCOR refueling room model nodalization included modeling features to minimize excessive mixing. The refueling room is modeled as a single control volume. However, the inlet flow into the SFP (i.e., CV301 in Figure 38), comes directly from the hatch region (see left-hand side of Figure 40). In this manner, the cool gases leaving the lower regions of the building are not brought into thermal equilibrium with gases above the SFP. Cross-flow is simulated between CV300 and CV301 as observed in CFD calculations.

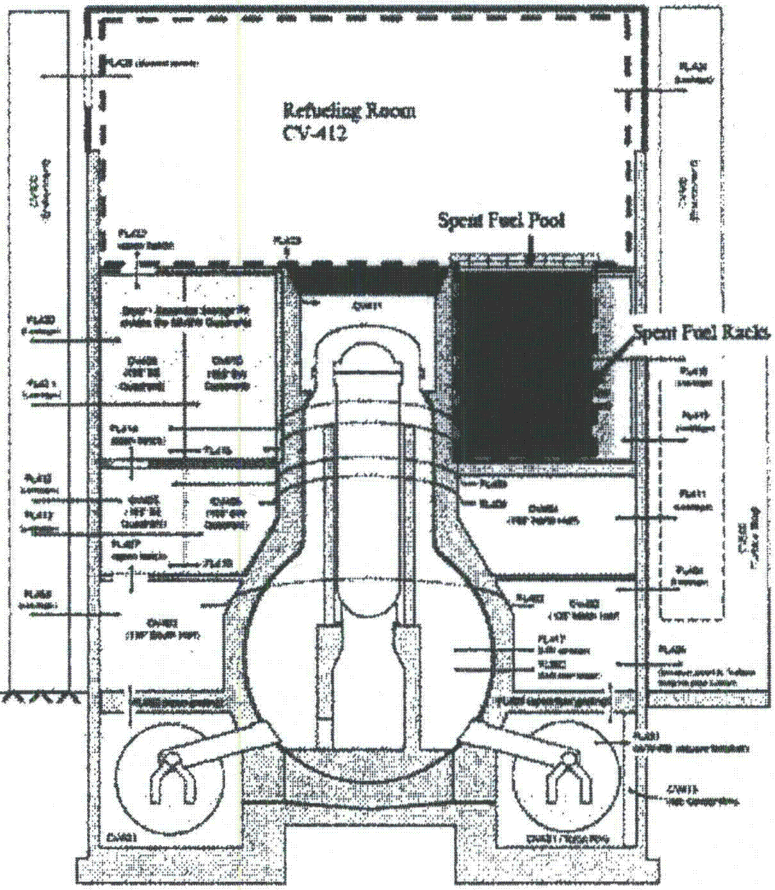

Figure 40: MELCOR reactor building model

6.2.1. High-density Loading During Outage

During outage where the SFP and reactor are hydraulically connected, a single control volume is used to represent both the reactor well and separator/dryer (S/D) pool as shown in Figure 41. The total volume of pool in CV601 is about 1900 m^3 (neglecting the dead end pool volume of 243 m^3 below the S/D gate elevation). CV601 is hydraulically connected to CV300 (see Figure 38) using two flow paths until the water level reaches the SFP gate and no more water can flow in to the SFP. The reactor power is applied as an external energy source until the pools become disconnected. The total additional volume of water above the SFP gate is about 1400 m^3.

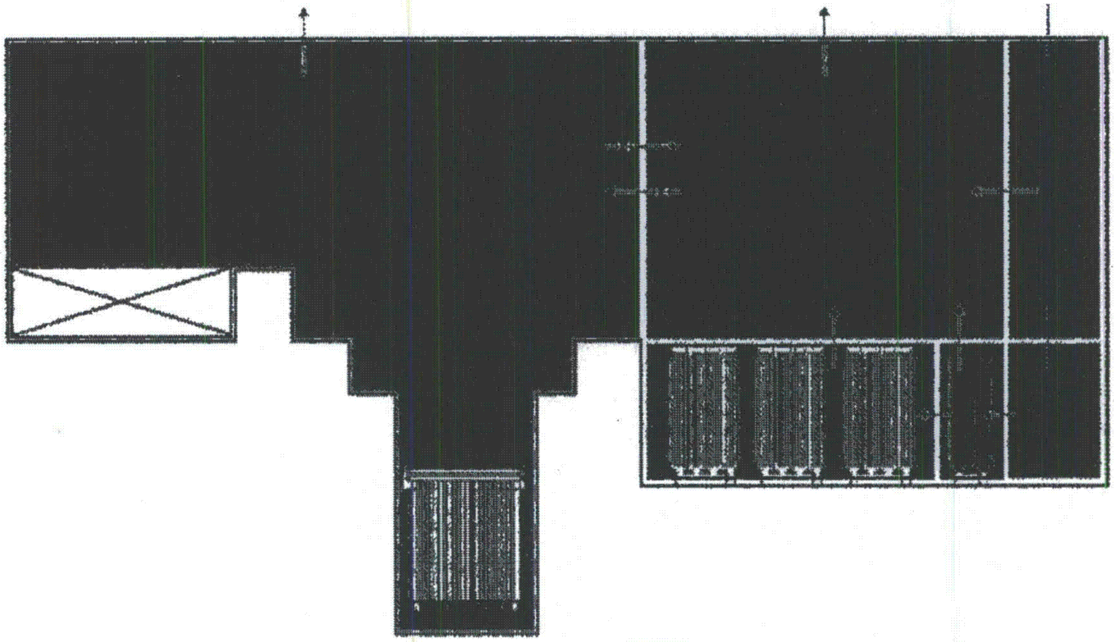

Figure 41: SFP and Reactor connection model during outage

For both OCP1 (at 4 days) and OCP2 (at 13 days), CV601 is connected to the detailed model of the SFP (Figure 38). The assembly layout for OCP1 in a 1x4 pattern is shown in Figure 42 where the assemblies are grouped into 6 types or "rings" by decay heat power and time of discharge. The 88 assemblies from the most recent offload in Ring 1 are surrounded by 352 old assemblies in Ring 2[18]. Ring 3 is empty during outage where the assemblies still reside in the reactor[19]. Ring 5 contains the last offload (284 assemblies) with an additional 31 assemblies from previous offloads[20]. Rings 2, 4, and 6 have a total of 2456 assemblies with their total decay heat distributed in each ring scaled by the number of assemblies. Within each MELCOR ring, the assembly decay heat is uniform. Consequently, for any given scenario, the decay heat in each ring is adjusted to give the average assembly power. Finally, the 764 empty cells in Ring 7 were placed around the outside of the SFP which promotes open air downflow into the SFP in the event of a complete loss-of-coolant inventory accident. The empty cells (764 in ring 7 and 196 in ring 3) have no decay heat. For the empty cells in Ring 3, the axial nodalization is detailed (see Figure 39) without the bypass control volume. This will ensure a better representation of flow through the assemblies and also modeling of heat transfer between components in various rings.

[18] All the old assemblies are smeared in MELCOR rings 2, 4, and 6, i.e., decay power per assembly is the same.

[19] The decay power for the ring 3 assemblies is added to the CV601 external power. Therefore, OCP1 has less power in the SFP since the 196 ring 3 assemblies have not been moved yet.

[20] This was simple a modeling convenience not to change the nodalization from the original model in the security assessment work.

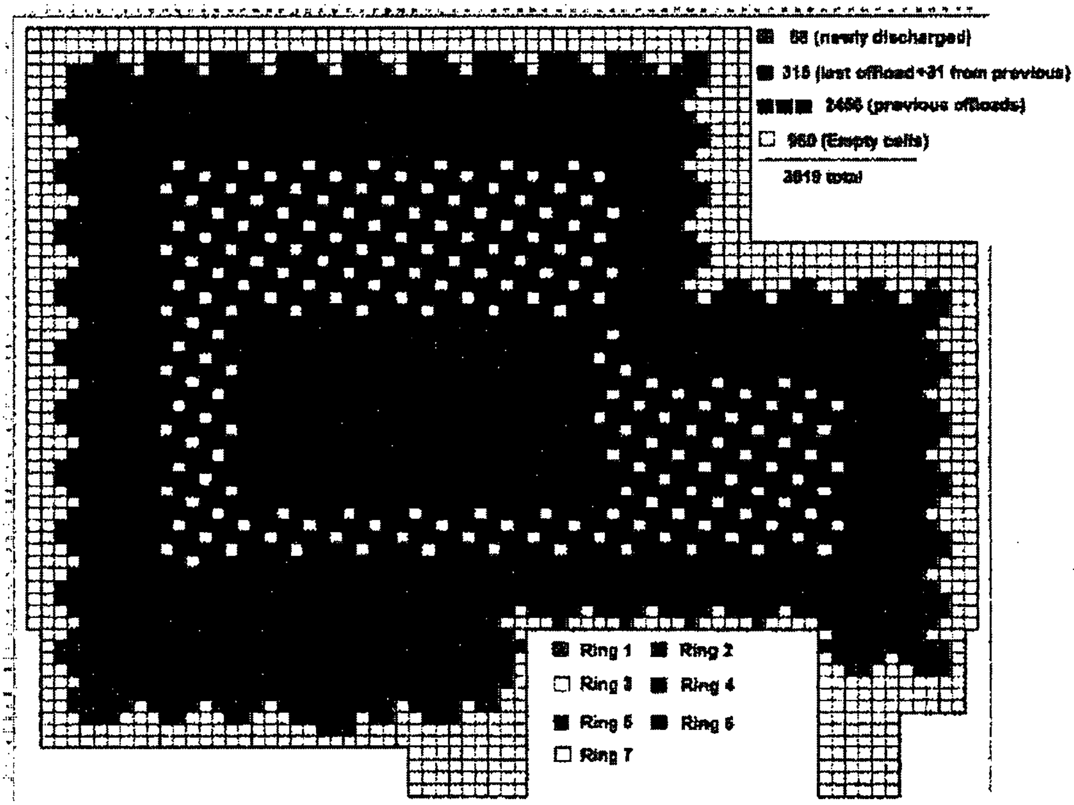

Figure 42: Layout of assemblies for OCP1 high density (1x4) model

Figure 43 shows the cell-wall radiation view factors between the various rings.[21] The resultant view factor specifies the amount of coupling from each region to another. For example, the Ring 1 cells are completely surrounded by Ring 2 cells. Hence, the view factor from Ring 1 to Ring 2 is 1.0. Similarly, Rings 3 and 4 and Rings 5 and 6 are coupled in 1x4 patterns. Using the specific layout in Figure 3.5, the special MELCOR generalized radiative heat transfer coupling model was prescribed to represent the thermal coupling between Rings 2 and 4, Rings 4 and 6, Ring 6 and 7, and Ring 7 and the SFP wall. The radial coupling for these regions was specified as the product of the area (i.e., represented as the number of coupling panels) times the view factor[22]. In OCP2, the 196 assemblies have been moved to Ring 3 as shown in Figure 44, and the radial thermal coupling is preserved as in Figure 43.

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[21] MELCOR by default models intra-cell radiation between concentric rings. To disable the radiation model for rings 2 to 3 and 4 to 5, the radial view factor area is set to zero.

[22] The view factor is assumed unity. It should be noted that there is a temperature gradient within each ring and MELCOR attempts to model a multi-dimensional geometry with a simplified two surface radiation model.

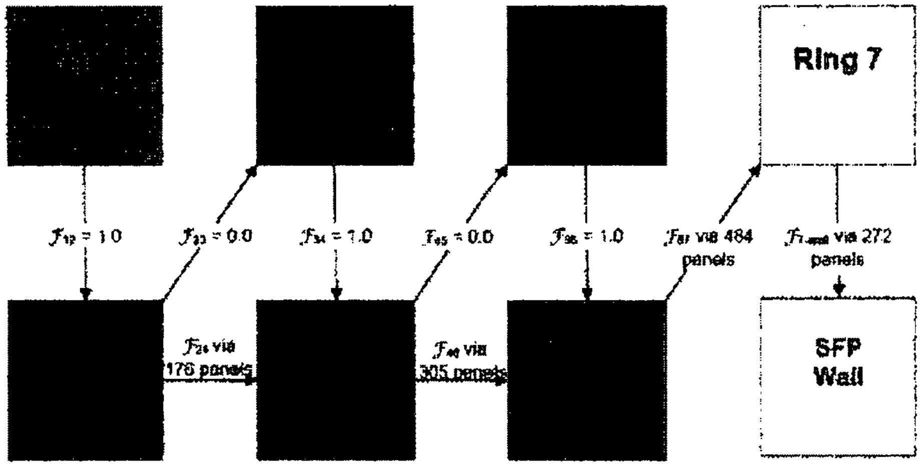

Figure 43: MELCOR radial radiative coupling scheme

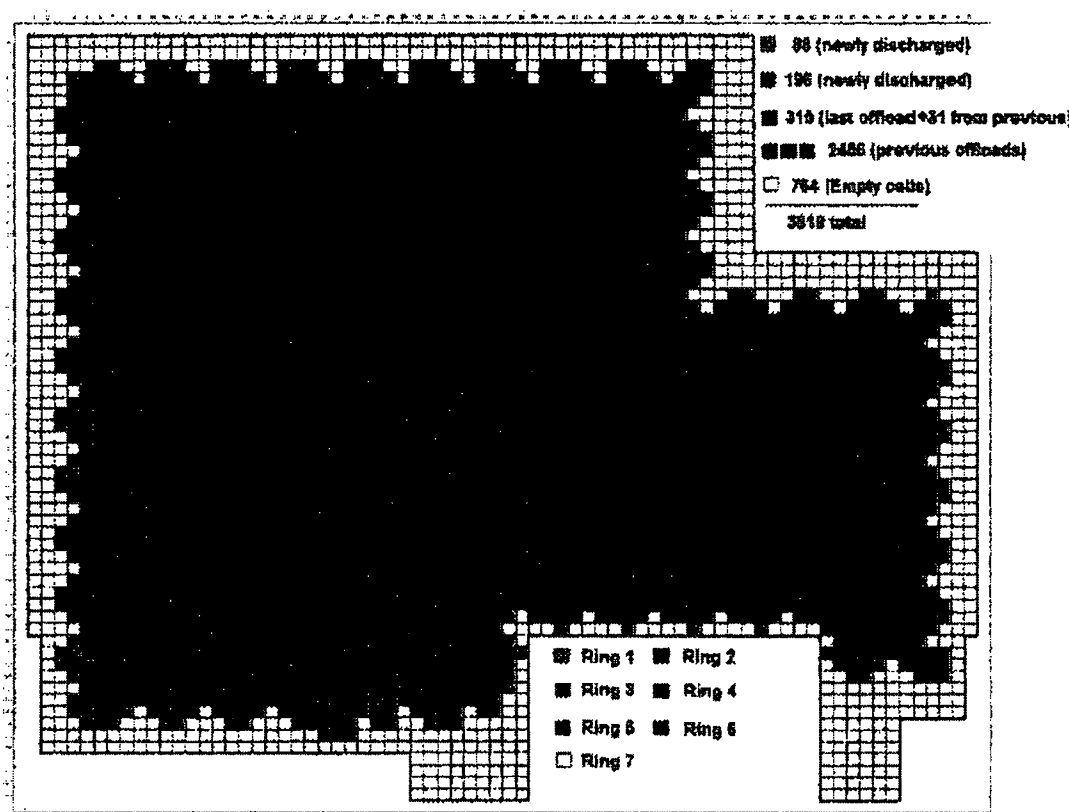

Figure 44: Layout of assemblies for OCP2 high density (1x4) model

The methodology described in Section 6.1.5 was used to estimate the decay heat power as a function of time for different OCPs. The results are shown in Table 24. The reactor power was based on the decay power for all assemblies residing in Peach Bottom reactor [NRC, 2012] by subtracting the power associated with assemblies that have already been moved to the SFP. For example, for OCP1, it is assumed that 88 assemblies are already in SFP.

Table 26: Distribution of decay heat in the reactor and SFP for high density loading

	Reactor (kW)	Spent Fuel Pool (kW)							
		Days	Ring 1 (88)[23]	Ring 3 (0)	Ring 5 (315)	Ring 2 (352)	Ring 4 (784)	Ring 6 (1320)	Total (2859)
[redacted]	10216	3.6	1927	0	465	80	179	301	2951
	9915	3.9	1867	0	452	80	179	301	2878
	9006	5.0	1690	0	417	80	178	300	2666
	7406	8.0	1403	0	358	80	178	300	2320
	6710	10.0	1282	0	334	80	178	300	2174
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (315)	Ring 2 (352)	Ring 4 (784)	Ring 6 (1320)	Total (3055)
[redacted]	4395	13.1	1144	1533	332	80	178	300	3567
	4117	15.0	1077	1444	330	80	178	299	3409
	3530	20.0	957	1294	318	79	176	296	3120
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (315)	Ring 2 (352)	Ring 4 (784)	Ring 6 (1320)	Total (3055)
[redacted]		37	720	973	324	79	177	298	2571
[redacted]		107	422	602	301	78	173	292	1868
[redacted]		383	191	315	230	73	162	273	1245

6.2.2. High-density Loading Post-Outage

The layout for the post-outage high-density loading is similar to OCP2 (see Figure 44). In post-outage, the assemblies are assumed to be in 1x4 pattern, which applies to OCP 3, 4, and 5. The assembly layout remained constant for these OCPs. However, the decay heat decreased from OCP 3 to OCP 5 as the aging time since reactor shutdown increased. The decay heat power in each ring is summarized in Table 26.

6.2.3. Low-density Loading During Outage

For the low density loading configuration, only the latest and the previous two offloads are considered. Therefore, for OCP2, the total number of assemblies in the pool is 852 (=284x3). For OCP1, the 196 assemblies from the current offload are still in the reactor and only 88 have been moved resulting in only 656 assemblies in the pool. Figure 45 shows the layout of assemblies in the SFP for OCP1, and the layout for OCP2 is shown in Figure 46. For both configurations, all the old fuel has been removed from the pool, and the current offload is in a 1x4 pattern with empties. Due to space limitations, the last two-offloads are placed in a

[23] The numbers in parentheses are the number of assemblies.

checkerboard pattern[24]. For the axial nodalization, Ring 1 contains both the channel (inside the canister) and the bypass (outside between canister and rack) control volumes while for Ring 2 both volumes are combined (see Figure 39). The basic radial thermal coupling from Figure 43 still applies, but the boundary area from Ring 6 to Ring 7 is 472 panels. For the modeling convenience, Rings 2, 4, and 6 from the high density layout are still present, but the cells contain only the rack component.

The distribution of decay heat in the pool is provided in Table 27. A comparison with the high density decay heat shows that the total decay heat in the pool for low density case is reduced by less than 20%. The total pool decay heat is dominated by the last offload, which is the same for the low and high density configurations. However, removing the old fuel also increases the available water volume (not occupied by the fuel and canister) while at the same time modifying the propagation characteristic of Zr fire due to reduced mass in the empty assemblies.

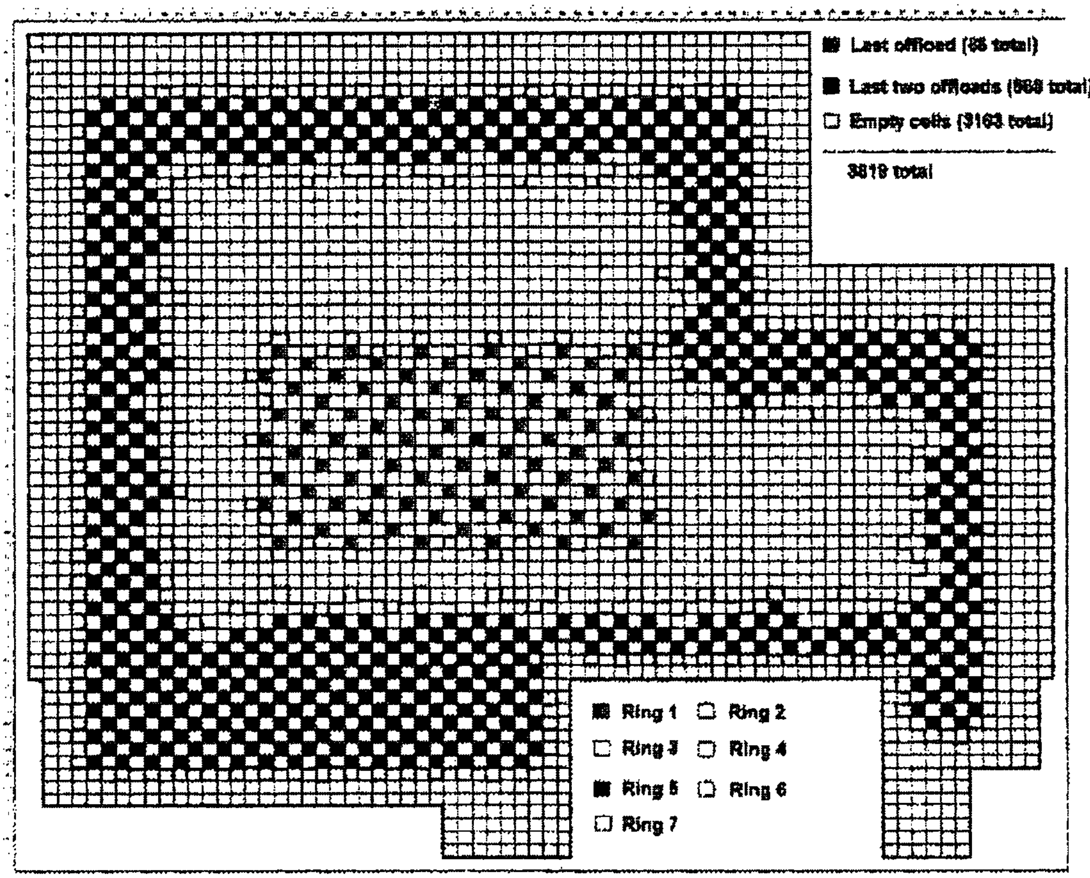

Figure 45: Layout of assemblies for OCP1 low density model

[24] There is not enough room to place all the fuel in 1x4. The current offload eventually requires 1420 cells (284 for assemblies and 284x4 for empties surrounding them) that would leave only 1635 cells. The 568 assemblies would require 2840 cells for storage in 1x4 pattern.

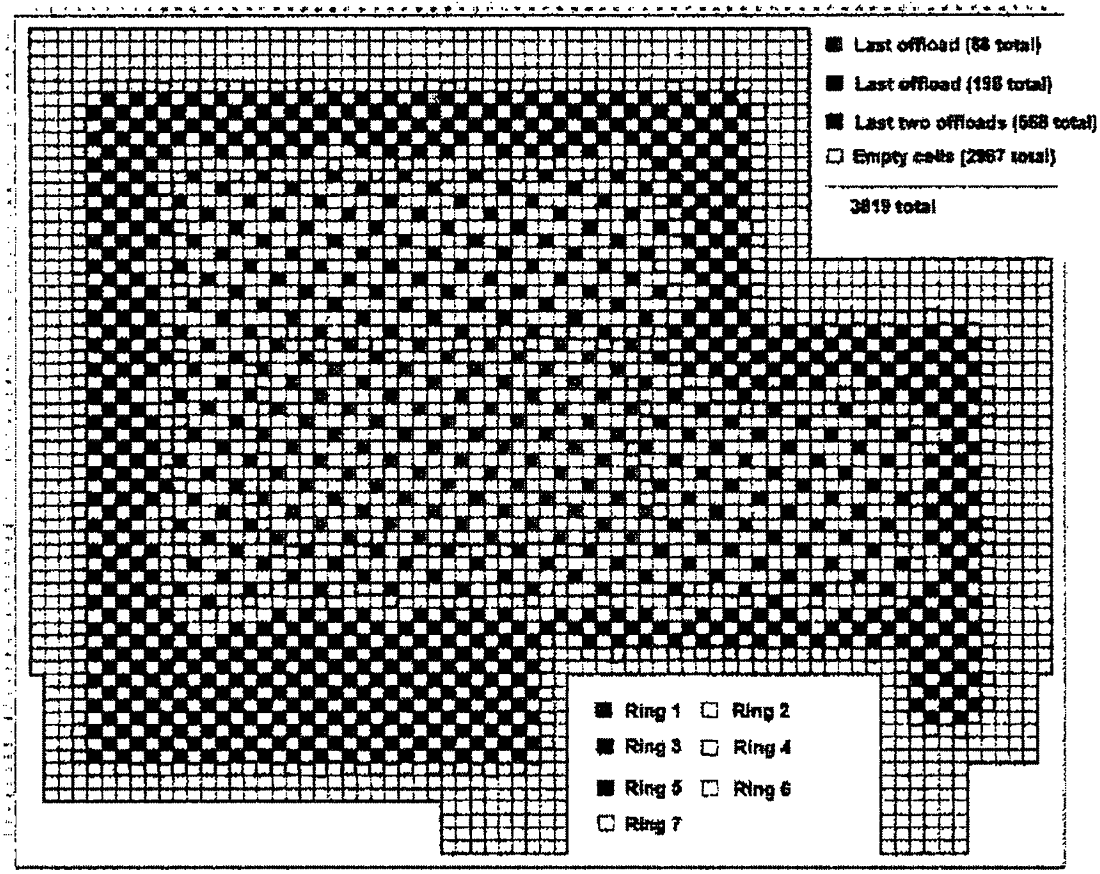

Figure 46: Layout of assemblies for OCP2 low density model

6.2.4. Low-density Loading Post-Outage

The post outage low density layout for OCP 3, 4, and 5 is identical to OCP 2 (see Figure 46), and the pool decay heat is provided in Table 27.

Table 27: Distribution of decay heat in the reactor and SFP for low density loading

	Reactor (kW)	Spent Fuel Pool (kW)							
		Days	Ring 1 (88)	Ring 3 (0)	Ring 5 (568)	Ring 2 (0)	Ring 4 (0)	Ring 6 (0)	Total (656)
[redacted]	10216	3.6	1927	0	599	0	0	0	2526
[redacted]	9915	3.9	1867	0	587	0	0	0	2454
[redacted]	9006	5.0	1690	0	551	0	0	0	2241
[redacted]	7406	8.0	1403	0	492	0	0	0	1895
[redacted]	6710	10.0	1282	0	468	0	0	0	1750
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (568)	Ring 2 (0)	Ring 4 (0)	Ring 6 (0)	Total (852)
[redacted]	4395	13.1	1144	1533	466	0	0	0	3143
[redacted]	4117	15.0	1077	1444	464	0	0	0	2985
[redacted]	3530	20.0	957	1294	448	0	0	0	2699
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (568)	Ring 2 (0)	Ring 4 (0)	Ring 6 (0)	Total (852)
[redacted]	[redacted]	37	720	973	455	0	0	0	2149
[redacted]	[redacted]	107	422	602	427	0	0	0	1451
[redacted]	[redacted]	383	191	315	339	0	0	0	845

6.3. MELCOR Analysis Results

6.3.1. Sequences That Do Not Lead to a Release

In general, four classes of scenarios do not result in a release from the fuel and include:

1. Boiloff scenarios with no SFP leaks
2. Mitigated scenarios for small leaks
3. Unmitigated scenarios in late phases (OCP 4/5)
4. Mitigated post-outage scenarios (OCP3 , 4, 5)

Boiloff

For the boiloff scenarios, a simplified model was used to estimate the pool heat up and water level drop. The model is shown in Figure 41, where all the assemblies are combined in two rings representing the fuel and empty cells. Only the thermal-hydraulic models in MELCOR are active, and the power for both the reactor well pool and SFP are provided as external sources to the water pool. The results are considered conservative since the heat capacities of the assemblies are not taken into account. The time-dependent power is taken from Table 26 for high density or Table 27 for low density cases. The top of the pool is connected to the reactor building (see Figure 40) in the same manner as in the detailed model. This simplified model is used as a screening tool to determine if more detailed analysis is needed. Figure 47 shows the

water level as a function of time for both high and low density cases for OCP 1, 2, 3, and 4[25]. Figure 47 also identified the time required to reach pool saturation. For cases in the same OCP, the high density cases become saturated sooner since there is more water volume. In the late OCPs following refueling, the difference in the timing directly correlates to the decay heat power. While there are differences in post saturation water level for OCP 3 and 4, the water level for OCP 1 and 2 are similar as a result of mixing assumed between the reactor well water and the SFP water (see Figure 41). For the OCP 4 low density and OCP 5 cases, the SFP never becomes saturated in 72 hours. The slight water level increase during the sensible heating period is due to change in pool density as the water heats up. The analysis shows that there is at 4.6 m (15 ft) of water above the top of racks in OCP 1 at 72 hours.

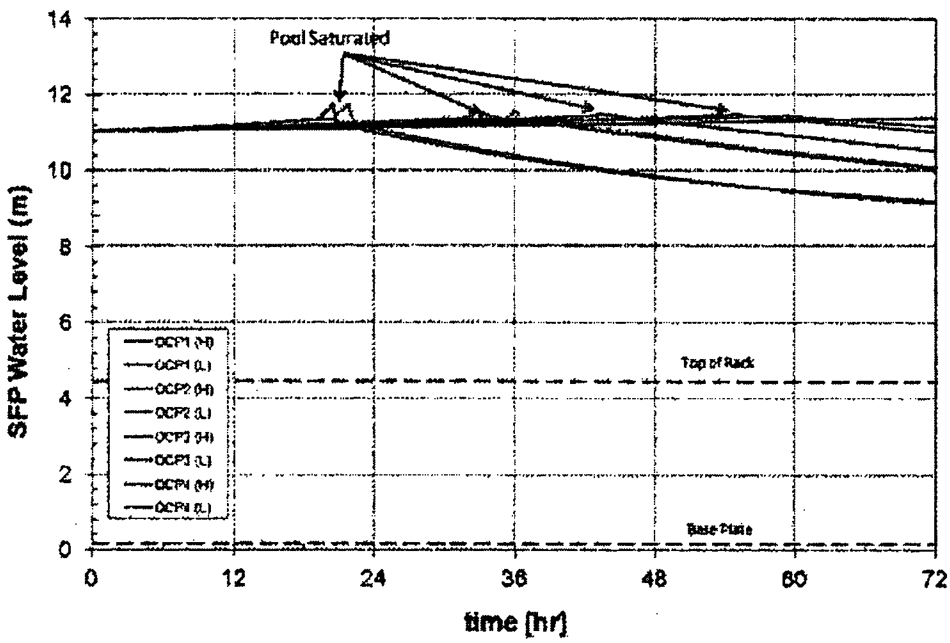

Figure 47: Water level for boiloff scenarios

<u>Mitigated Scenarios (Small Leaks)</u>

The small leak is modeled in MELCOR with a 1.75" diameter hole at the bottom of the pool based on the structural analysis and damage to the pool (effective size of cracks in the liner and the concrete). The water level and the injection and leak mass flow rates for the low density OCP1 case are shown in Figure 48 and Figure 49, respectively. Once the water level reaches 10 m at about 7 hours, the leak is detected and together with the deployment logic, the water injection begins at about 9.5 hours. In this cases, mitigation is direct makeup to the pool (injection) since the water level at the time of deployment is more than a meter above the top of

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[25] The initial water level is assumed to be 11 m. The initial water temperature is 28°C (82°F). Both these initial conditions are applied to all accident scenarios in this report. During a teleconference with the licensee (4/24/2012), this is the post-outage water temperature under steady state conditions where the heat exchangers are working. During outage (OCP 1 and 2), the water temperature could vary between ~80°F and ~100°F. The higher temperature affects the sensible heating of the pool and is not expected to change the overall conclusion of boiloff scenarios.

the racks. For this small leak, the initial water flow rate is about 250 gpm which is much lower than the makeup capacity and the water level is quickly restored. This calculation is only run for 24 hours to show the effectiveness of mitigation. Therefore, it is concluded that for all slow leaks scenarios, the fuel never becomes uncovered since the makeup capacity is twice the leak rate. The leak rate is only a function of the water level (hydrostatic head) and is independent of the SFP layout configuration as long as the water level remains above the top of the racks.

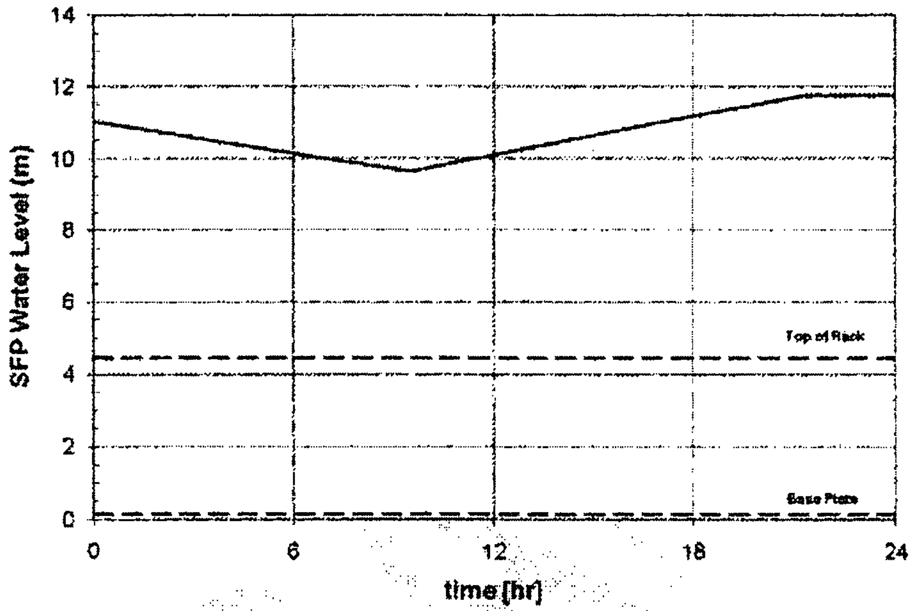

Figure 48: Water level for mitigated low density OCP1 (small leak) scenario

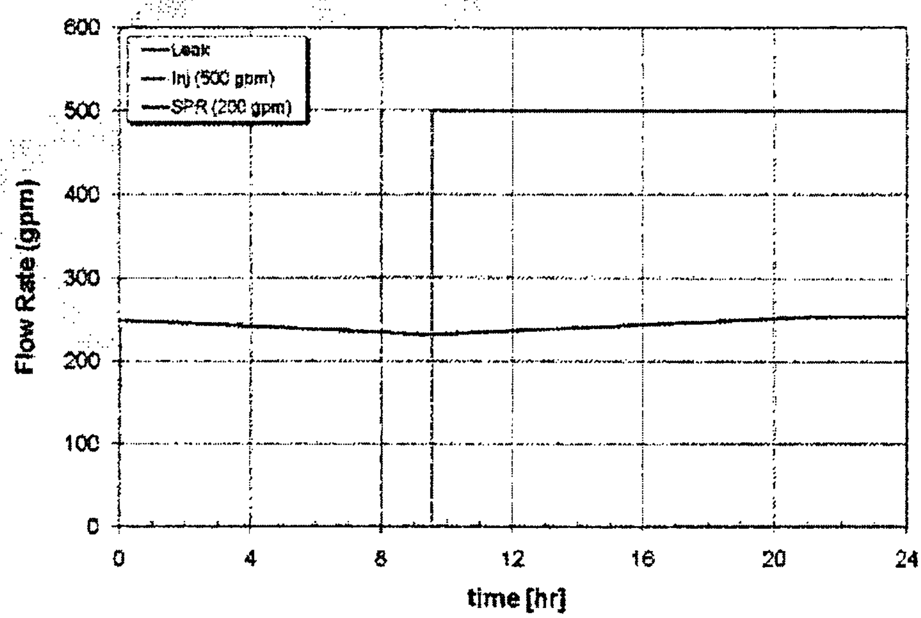

Figure 49: Flow rates for mitigated low density OCP1 (small leak) scenario

Unmitigated OCP 4/5 Scenarios

For OCP 4, the decay heat is between 37% to 48% lower than OCP 3. None of the unmitigated scenarios in OCP 4 or 5 lead to a release from the fuel[26]. The thermal hydraulic response of the high density pool to a small leak and a moderate leak are provided in Figure 50 through Figure 53. It takes less than 6 hours to clear the rack base plate and initiate air flow for the moderate leak, while for the small leak case, the rack base plate does not clear until about 39 hours. In both cases, there is a heat up of the fuel as the water level is reduced below approximately half the height of the fuel. For the small leak case, it takes longer and the heat up is slower as there is some steam cooling of the fuel.
The heatup rates for the low density cases are somewhat similar to the high density cases (see Figure 54 or Figure 55). The maximum clad temperature and the initial heatup rate in Ring 1 is actually higher for the low density cases because of reduced heat transfer from Ring 1 to Ring 2[27]. Even though for low density cases, the total decay heat in the pool is only 77% of the high density case, the decay heat in Ring 1 is identical in both cases.

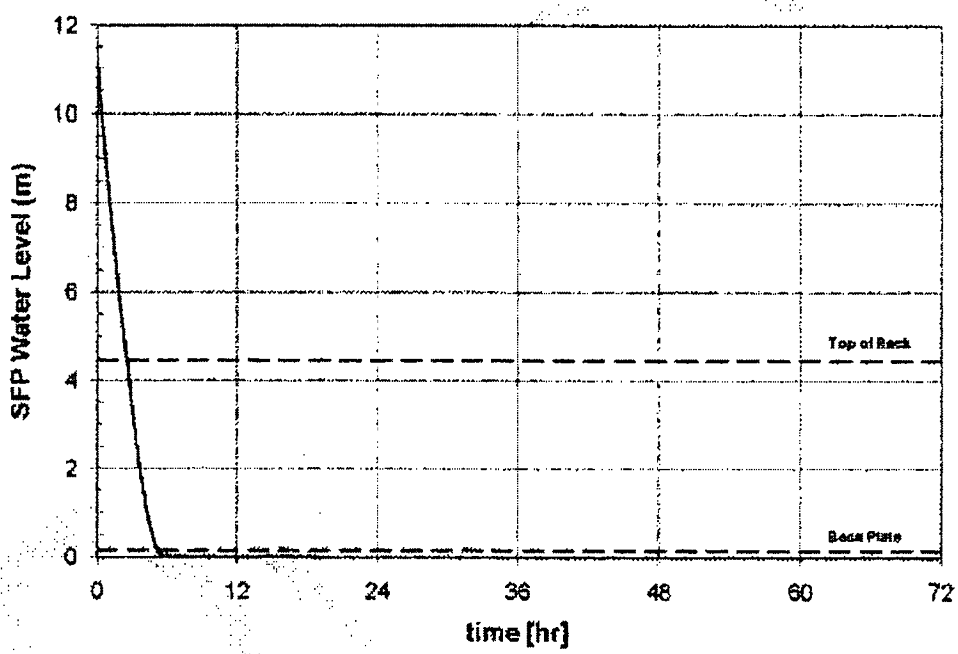

Figure 50: Water level for unmitigated high density moderate leak (OCP4)

[26] The start of the release of radionuclides from the fuel is modeled based on a temperature of 900 °C (1173 K). At this temperature, the cladding is assumed to fail and the gap inventory from the fuel is released. Further release from the fuel is based on the CORSOR-Booth model and is a function of fuel temperature [Gauntt, 2010].
[27] The reduced mass in Ring 2 (only racks) initially limits heat transfer from Ring 1 until a sustained natural circulation is established.

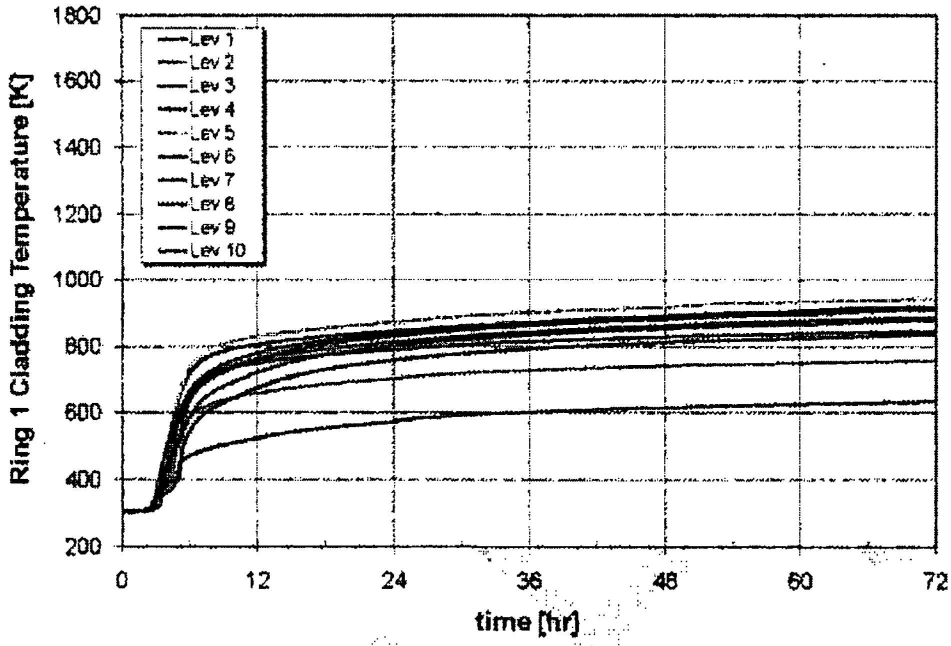

Figure 51: Ring 1 temperature for unmitigated high density moderate leak (OCP4)

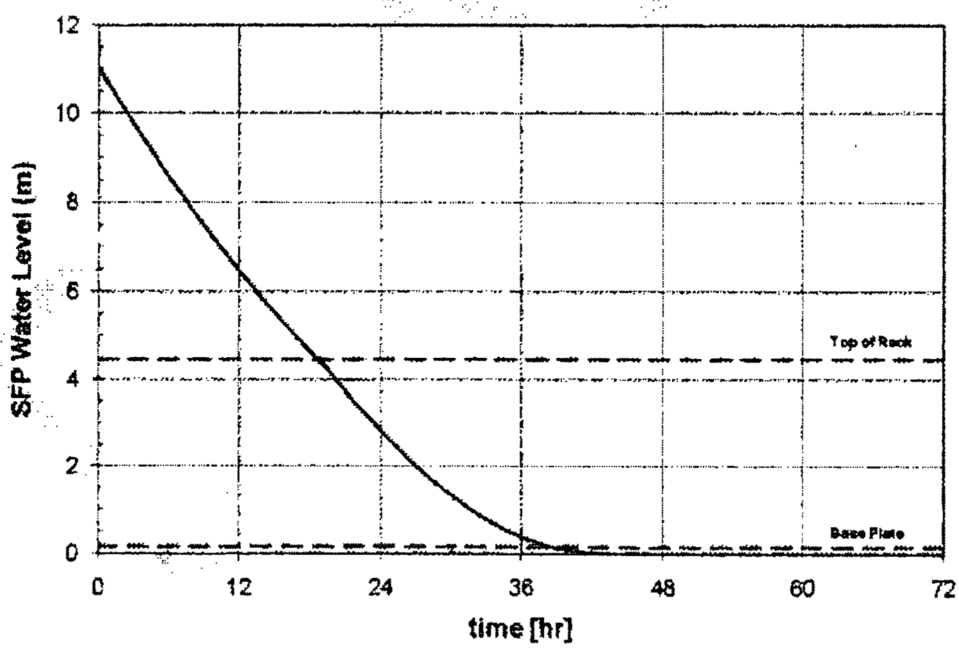

Figure 52: Water level for unmitigated high density small leak (OCP4)

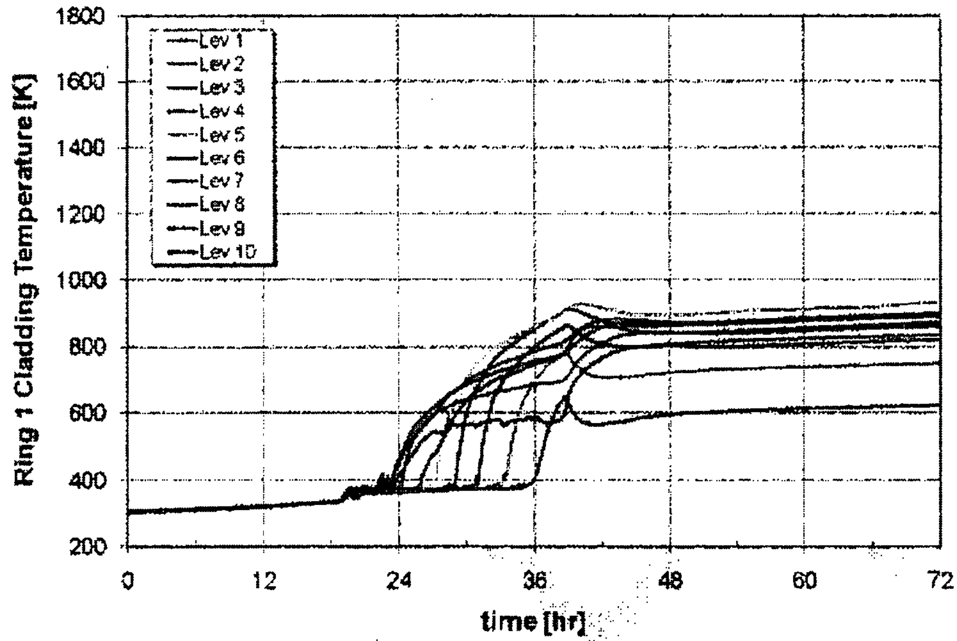

Figure 53: Ring 1 temperature for unmitigated high density small leak (OCP4)

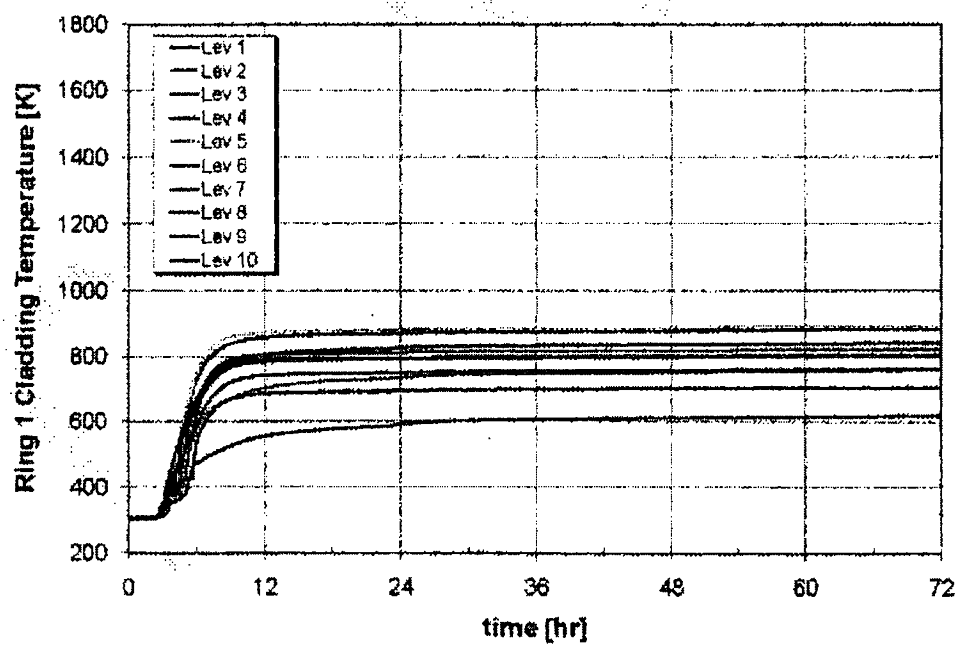

Figure 54: Ring 1 temperature for unmitigated low density moderate leak (OCP4)

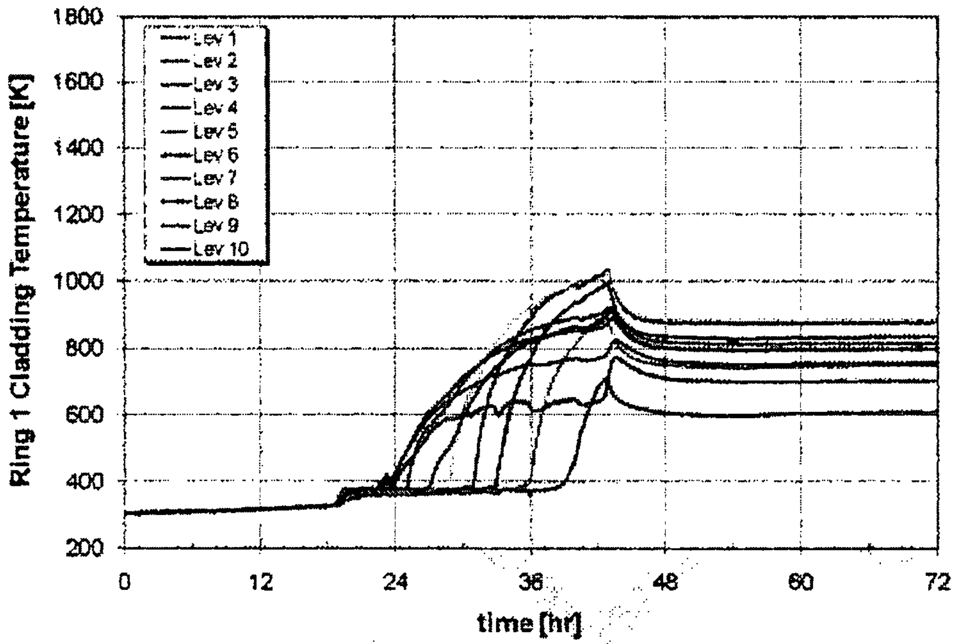

Figure 55: Ring 1 temperature for unmitigated low density small leak (OCP4)

<u>Mitigated post-outage moderate hole scenarios (OCP3, 4, 5)</u>

For the post-outage moderate hole scenarios, the mitigation involves actuation of the sprays. The moderate leak is modeled in MELCOR with a 4.5" diameter hole at the bottom of the pool based on the structural analysis and damage to the pool (effective size of cracks in the liner and the concrete). The MELCOR modeling of the sprays was discussed in Section 6.1.3, where two modeling options were presented (i.e., simple flow regime model on or off). Only high density OCP 3 results[28] are presented since the later phases unmitigated scenarios do not lead to release.

Figure 56 shows the water level for the moderate leak, high density OCP3 scenario. Because of the spray activation at 3 hours (see Figure 57), the bottom of the racks clears for natural circulation air flow more than an hour later compared to an unmitigated case (see Figure 50). Finally, the spray flow rate and the leak rate are equilibrated by about 8 hours as required by the hydrostatic head at the bottom of the pool. The actual spray water reaching the bottom of the pool is somewhat smaller than 200 gpm in Figure 57 because of heat transfer from spray droplets to the atmosphere and fuel rods[29]. The response of the clad in Ring 1 for the case where the simple flow regime model is active is shown in Figure 58. As expected, the top cells experience more cooling as there is more water coverage. The temperatures reach a quasi-

[28] Low density case is similar to high density and there is no release.

[29] It would take about 15 gpm of water to remove the entire decay heat in the pool. However, some of the decay heat is being removed by natural circulation through the assemblies and leaking out of the reactor building.

steady state by about 10 hours[30] and the maximum clad temperature is about 850 K. Figure 59 shows the clad temperatures for the case where the simple flow regime model is disabled. In this mode, the main cooling mechanism is by convection from the fuel rods to the atmosphere and none of axial segments experience quenching. The maximum clad temperature is about 840 K which is comparable to the previous case. Thus, even though the details of heat transfer and fuel heatup are different, the maximum clad temperatures are almost the same and well below the gap release criterion. This is partially due to the importance of the heat removal by air natural circulation through the racks. If there was not air natural circulation through the racks, the cooling of the fuel by the spray flow (i.e., modeled with the simple flow regime map), would be very important to the coolability of the fuel.

To further test the impact of the modeling assumptions, two additional calculations were performed by assuming a 3 hour delay in the actuation of the spray as shown in Figure 60.[31] Both Figure 61 and Figure 62 show that following the initial heat up of the fuel and reaching a maximum clad temperature (just below 900 K) at about 6 hours, the spray flow rate is sufficient to cool the fuel and avoid release.

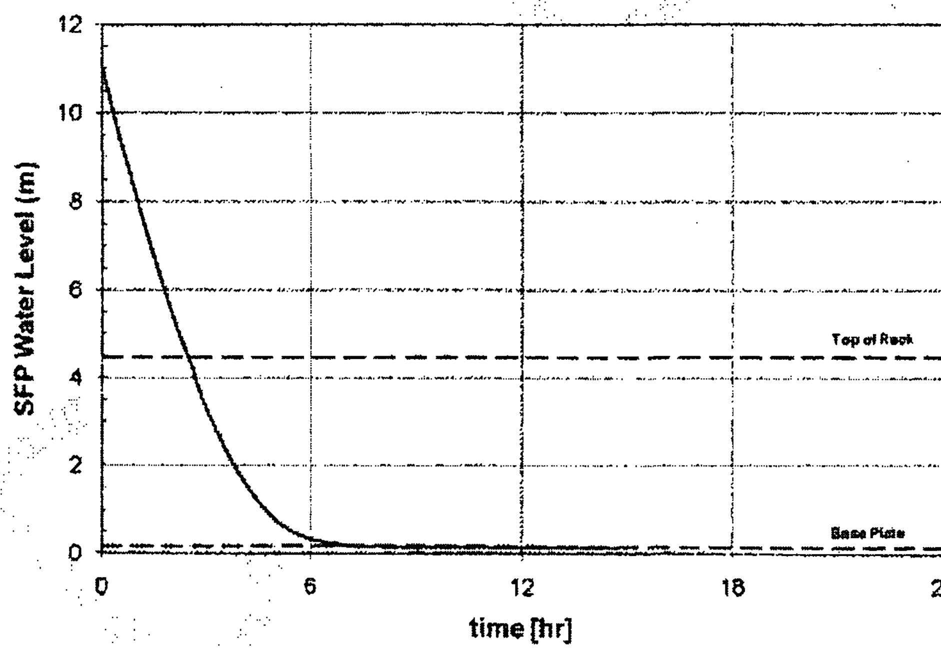

Figure 56: Water level for mitigated high density moderate leak (OCP3)

[30] The calculation fails shortly after 10 hours due to numerical problems.
[31] These cases were actually run based on an earlier logic for spray actuation that assumed a 3 hour delay at the end of deployment.

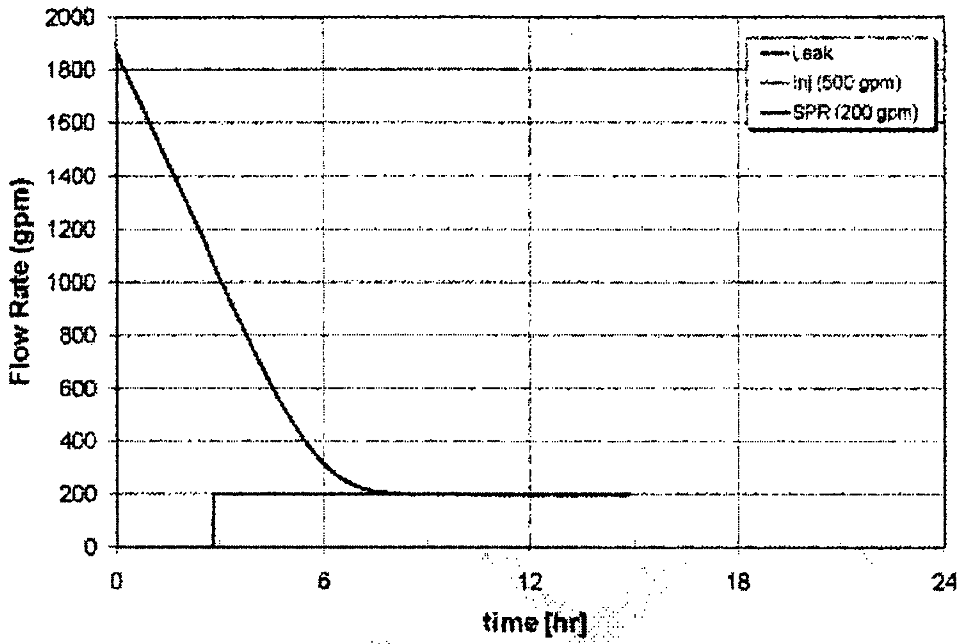

Figure 57: Water flow rates for mitigated high density moderate leak (OCP3)

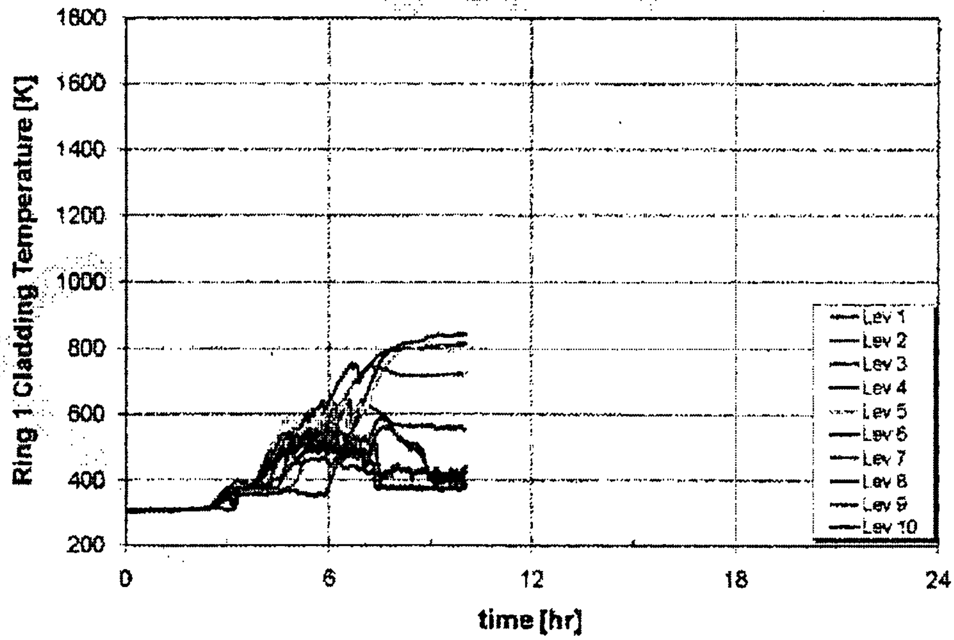

Figure 58: Ring 1 clad temperatures for mitigated [simple flow regime active] high density moderate leak (OCP3)

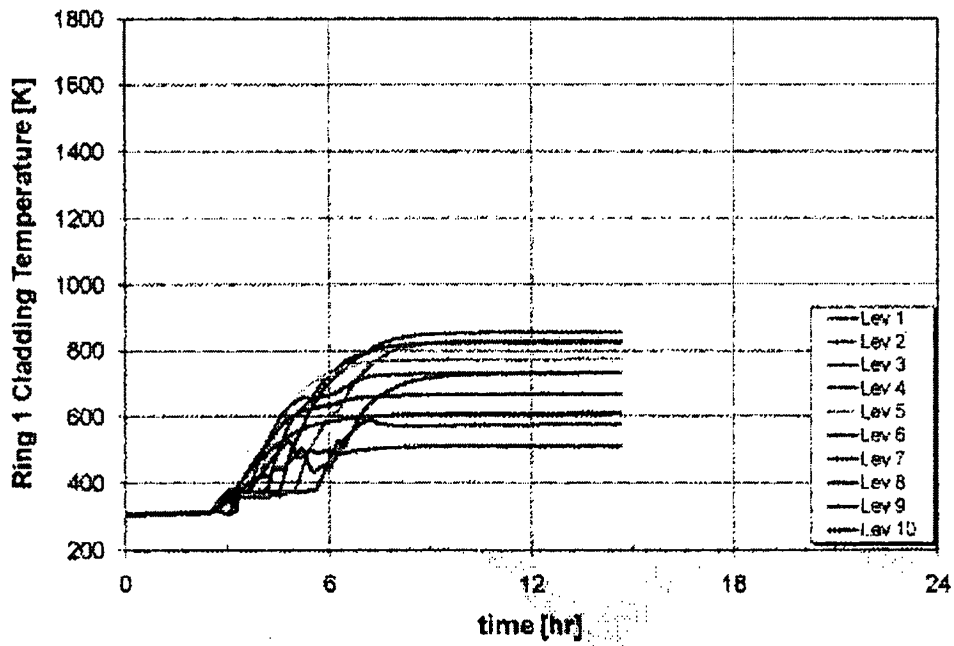

Figure 59: Ring 1 Clad temperatures for mitigated [simple flow regime inactive] high density moderate leak (OCP3)

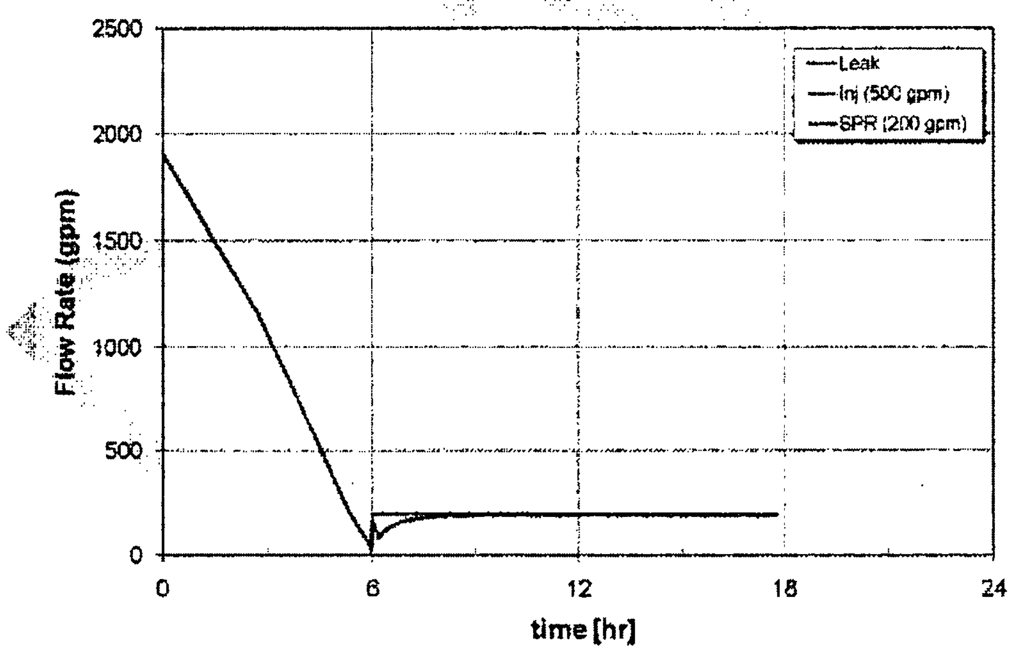

Figure 60: Flow rates for mitigated high density moderate leak (OCP3) with late actuation of sprays

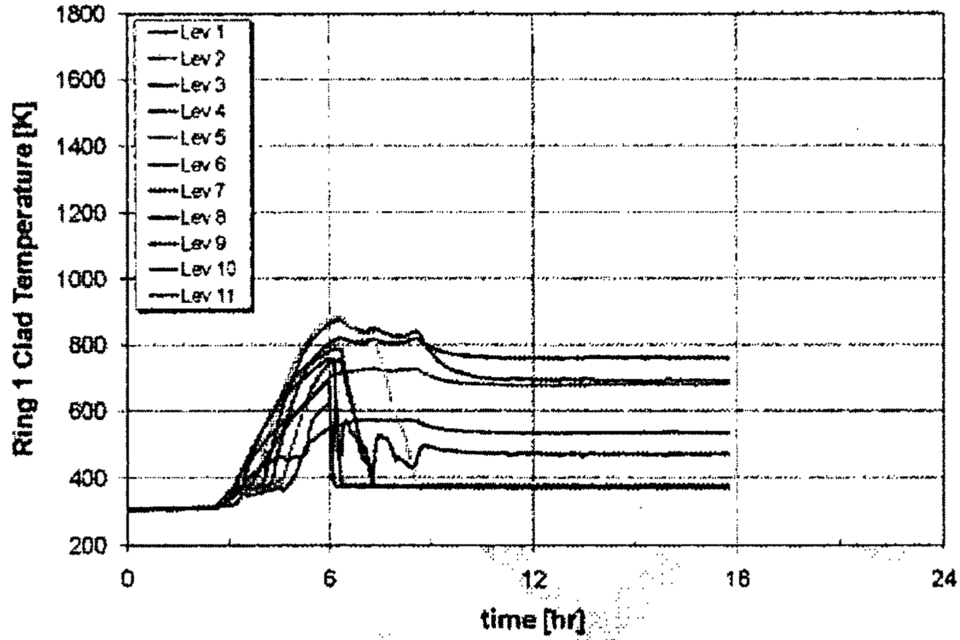

Figure 61: Ring 1 clad temperatures for mitigated [simple flow regime active] high density moderate leak (OCP3) with late actuation of sprays

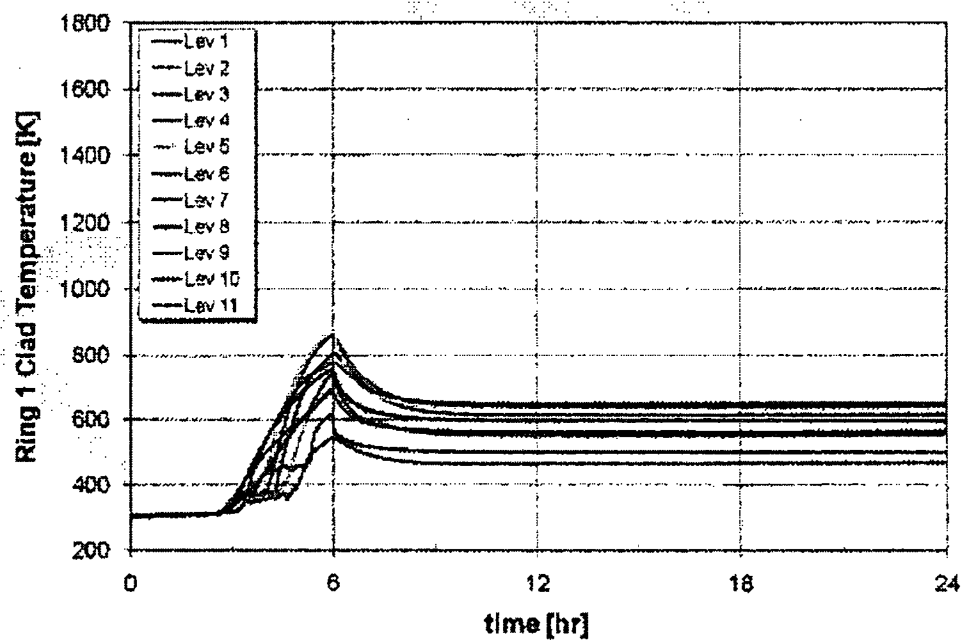

Figure 62: Ring 1 clad temperatures for mitigated [simple flow regime inactive] high density moderate leak (OCP3) with late actuation of sprays

6.3.2. Sequences That Do Lead to a Release

All the unmitigated sequences in OCP 1, 2 and 3 lead to release. In this section, only representative scenarios will be discussed to illustrate the accident progression phenomenology. One of the phenomena that has a significant impact on the overall release is the failure of the reactor building as a result of failure of the blowout panels or the roof. Failure of the reactor building introduces additional air that results in further oxidation of the hot fuel leading to enhanced release and fuel failure. The refueling room with the SFP at the top of the reactor building is modeled as a single volume (Figure 40) and hydrogen released from the SFP is assumed to mix with the entire volume. It is assumed that the hydrogen will combust at a 10% concentration if there is adequate oxygen ($X_{Oxygen} \geq 5\%$) and no steam inerting ($X_{steam} \leq 55\%$). The sensitivity of the ignition assumptions and potential for reactor building failure was considered on a case by case basis.

<u>Unmitigated Moderate Leak (OCP1) Scenario</u>

The water level for the high density scenario (Figure 63) shows that it takes about 8.5 hours to clear the rack baseplate and establish natural circulation in the pool. The timing is longer compared to post-outage scenarios (see Figure 50) because of the additional water in the reactor well connected to the SFP. The reactor power (Figure 64) is assumed to go to zero as the water level reaches the SFP gate and the pool is disconnected from the reactor well.

As the water level decreases, the clad temperatures (Figure 65) start to increase initially as a result of decay heat and then by clad oxidation as air is circulated through the assemblies. The heatup of the cladding in Ring 1 results in a Zr fire that starts near the top of full rod region (see Figure 39) and propagates downward. The heatup in Ring 1 (fuel, cladding, canister, and racks) propagates to Ring 2 assemblies (Figure 66) and leads to the failure of the racks in Rings 1 and 2. The failure of racks at about 12 hours results in formation of a debris bed in the bypass and relocation to the baseplate but the channel boxes are still intact at this time. Between 13.6-14.2 hours, the channel boxes in Rings 1 and 2 fail that allows additional cooling of the debris through flow diversion from the bypass region. As a result, the oxidation power is reduced and the heat transfer from the hot inner assemblies is propagated outward and starts to gradually heatup the SFP wall liner.[32] Natural circulation and radial heat transfer throughout the SFP keeps the temperatures relatively low following the initial heat up in Rings 1 and 2. However, the fuel continues to slowly heat until a second Zr fire initiated at the top of the fuel in Ring 4 at about 42 hours in the upper levels that propagate downward. The second heatup is more intense and involves the other rings as indicated by both the oxidation power (Figure 64), and the clad temperatures in the outer rings (Figure 66).

OCP1 had a relatively rapid draindown where an air natural circulation flow developed through the racks prior to significant oxidation of the fuel. As a result of a relatively short duration of the steam oxidation phase, there was relatively little hydrogen generation.[33] The peak

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[32] The initial heatup of the liner is due to heat transfer from the water. There is an initial cooldown as cooler air circulates before the heatup from the fuel caused the temperature to increase.

[33] Hydrogen generation only occurs by oxidation of the SFP Zircaloy and steel with steam. Hydrogen is disassociated from the steam and released into the building, which can lead to combustion. If oxygen is

concentration in the refueling floor was only 5%, which is well below the minimum threshold for combustion and below a quantity that would lead to a significant pressurization of the reactor building. Consequently, there was no potential for a burn inside the reactor building, which remains intact.

The fission product releases began at about 12 hours. Because the reactor building remained intact, all releases to the environment are limited by the nominal leakage (see Figure 40). The reactor building decontamination factor is shown in Figure 69[34]. Aerosols also begin to deposit inside the building and the decontamination factor for Cs and I aerosols remains between 3 to 4 for much of the accident. The decontamination factor is defined as the ratio of fractional release from the fuel to the fractional release to the environment. As discussed before, MELCOR keeps track of the fuel releases from individual rings. The fuel releases are divided by the overall decontamination factor to arrive at the environmental release for each ring. It should be noted that MELCOR mechanistically models all deposition mechanisms; however, because of the mixing within the reactor building only an overall decontamination factor can be defined for all rings.

The Cs environmental release fraction for individual rings is shown in Figure 70. The release starts at about 9 hours from ring 1 followed by the release from ring 2 at 12 hours. The release profiles are consistent with the heatup in Figure 66. The later releases are due to the second heatup and involve all the outer rings (ring 3 is empty for OCP1). The total release fraction is the input to the MACCS code for consequence analysis and is defined by Equation (11)[35]. The decontamination factor is a dynamic quantity as the outer rings start to release (see the fluctuations in Figure 69); therefore, care is taken now to allow the earlier releases from inner rings to oscillate (thus preserving their release history).

The results of the low density case are shown in Figure 71. Comparing the heatup with the high density case (see Figure 65), the ring 1 low density case clearly heats up more rapidly initially since there are a lot of empty cells surrounding it (with the exception of the rack component) and heat is not very efficiently transferred radially that results in slower heatup of the ring 5 as shown in Figure 72. Even though the racks fail even in this low density case, the canisters remain intact and the Zr fire then moves down initially and then upwards as shown in Figure 71. The Cs environmental release fraction for Ring 1 shown in Figure 73 is comparable to the high density case (Figure 70), but there is no release from the older assemblies that result in a reduced total release fraction for the low density case.

---

present, then only air oxidation occurs and there is no hydrogen generation. In a larger leak, the water level drops below the bottom of the racks and allows natural circulation of air, which will precludes steam oxidation.

[34] The integral decontamination factor is the ratio of the fission products released from the fuel to the amount that reaches the environment. Upon the start of fission product releases, the quantity is infinite until the release to the environment begins. Consequently, the initial peak is an artifact of the definition, whereas the long-term value is best characteristic of the reactor building performance.

[35] It is an activity weighted release and is a function of the inventories in each ring. Therefore, there is more contribution from the outer rings that have higher inventories even though the release from these rings are smaller compared to ring 1.

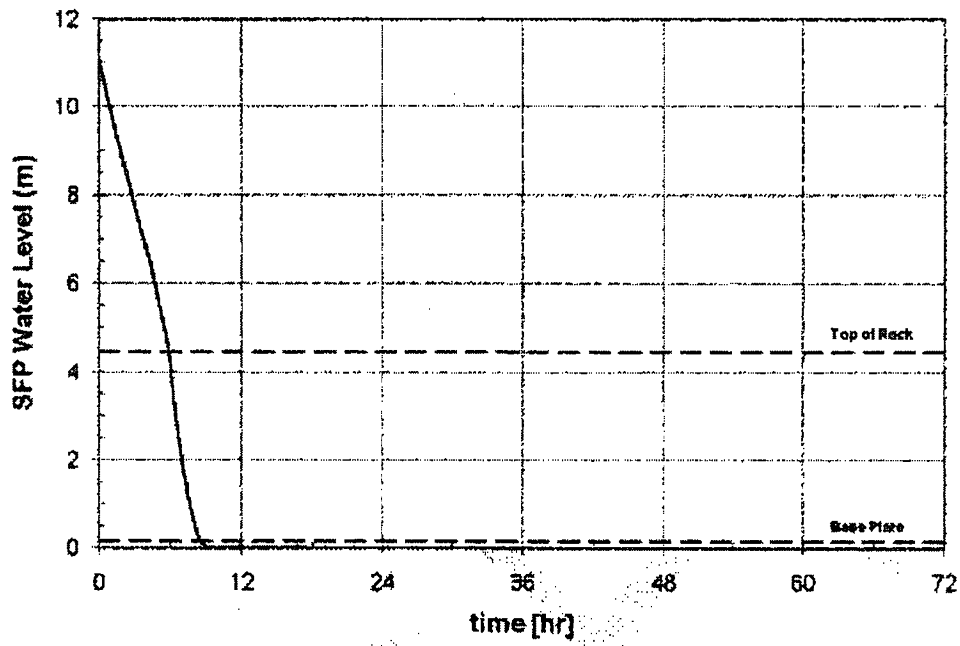

Figure 63: Water level for unmitigated high density moderate leak (OCP1)

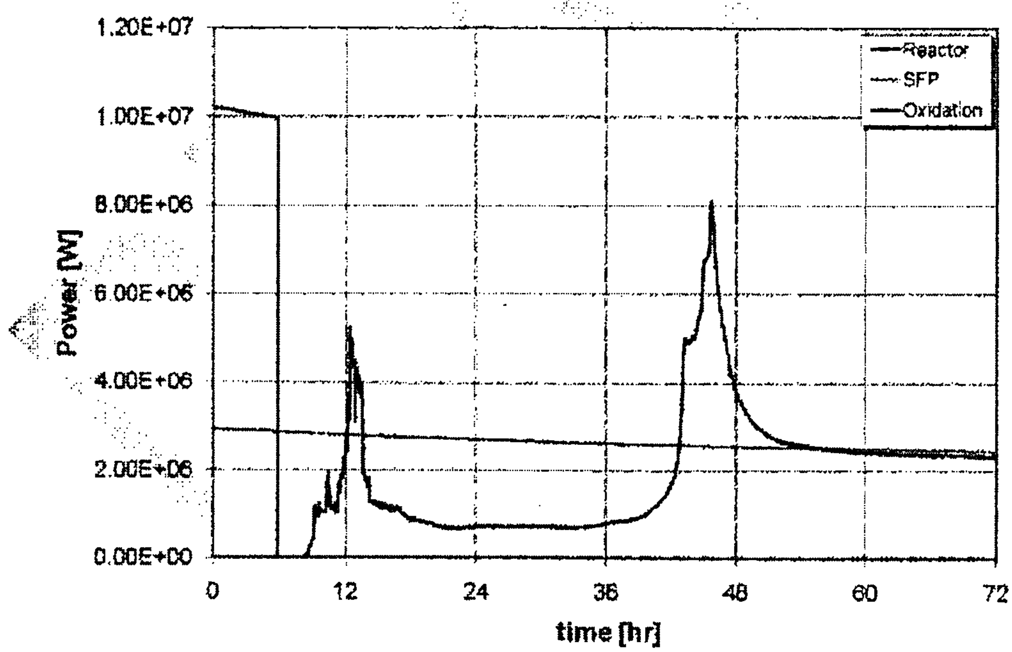

Figure 64: SFP power for unmitigated high density moderate leak (OCP1)

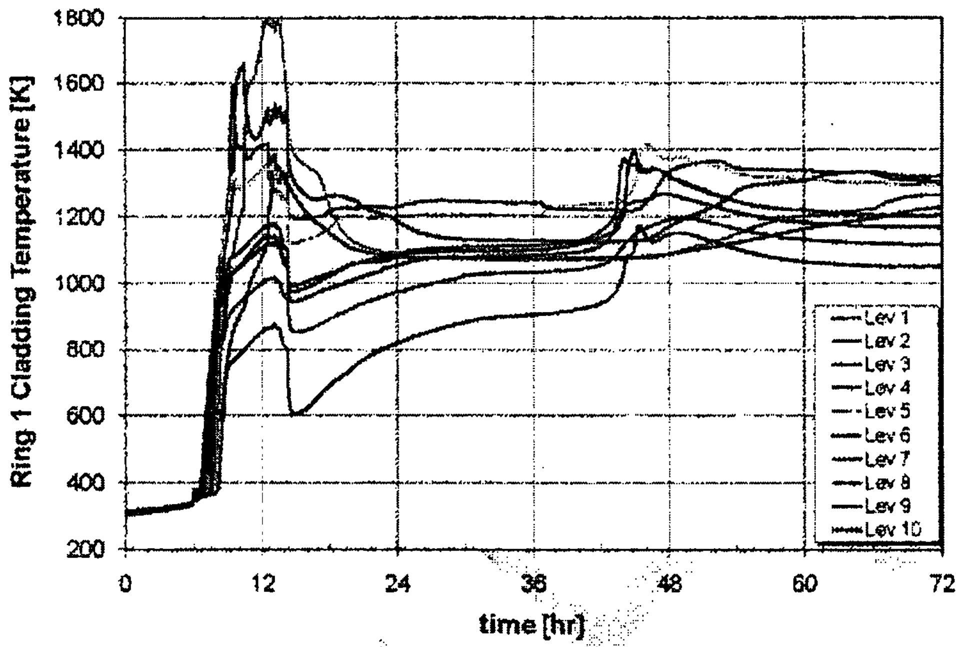

Figure 65: Ring 1 clad temperature for unmitigated high density moderate leak (OCP1)

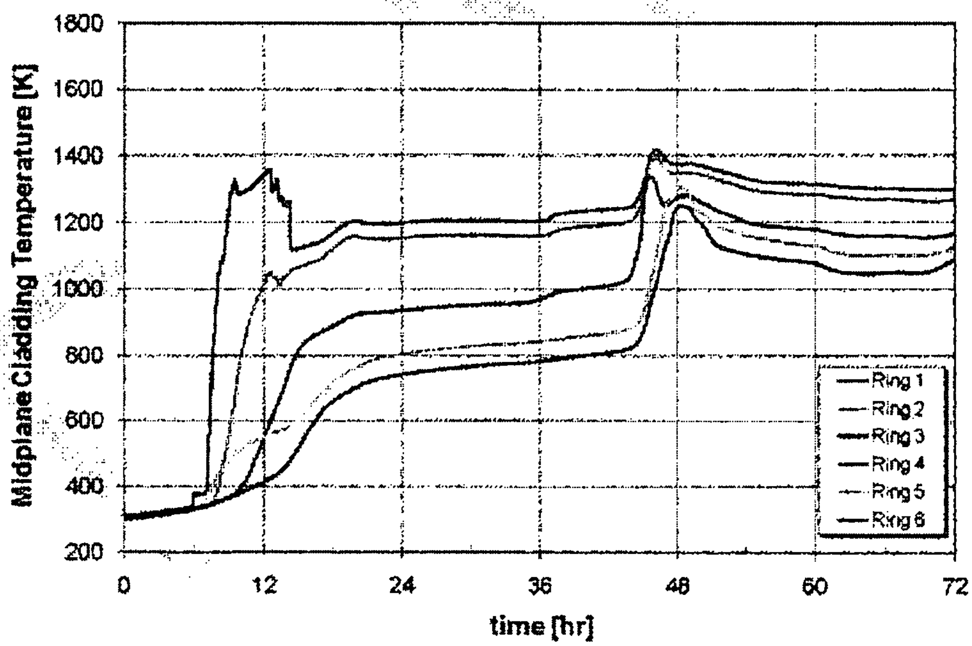

Figure 66: Midplane clad temperature for unmitigated high density moderate leak (OCP1)

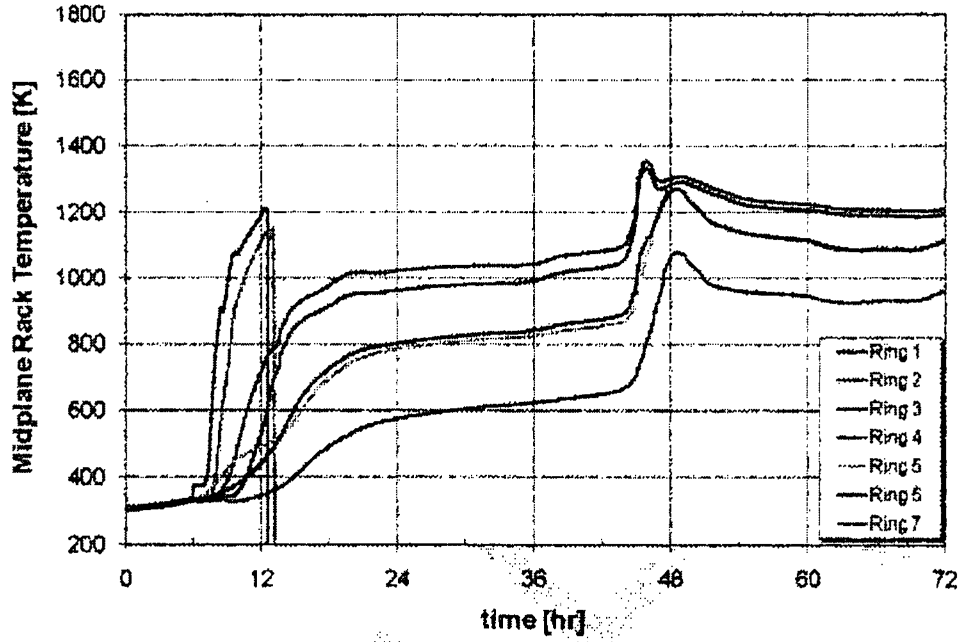

Figure 67: Midplane rack temperature for unmitigated high density moderate leak (OCP1)

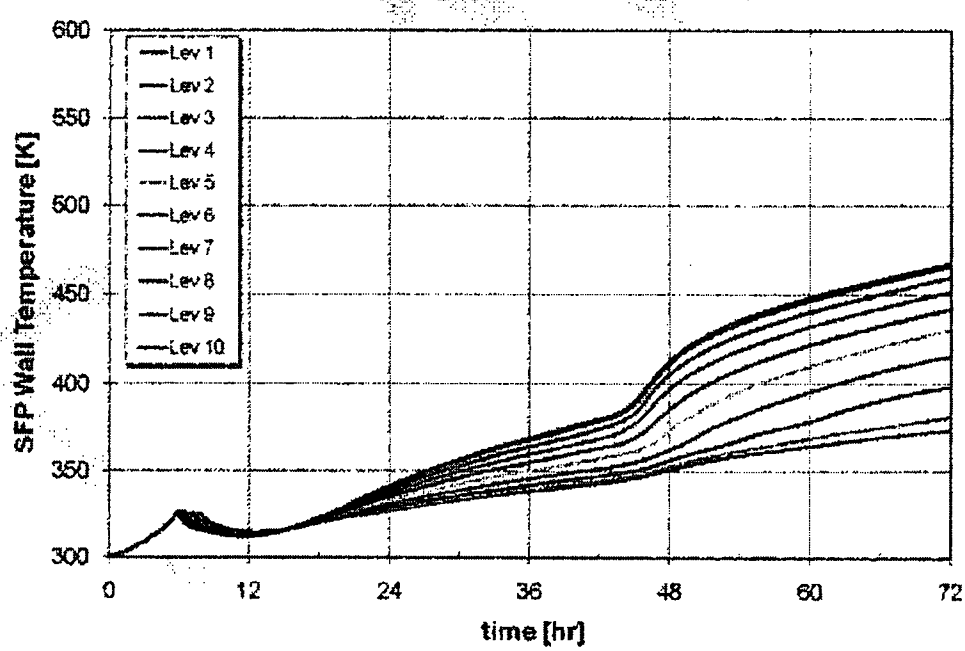

Figure 68: SFP wall liner temperature for unmitigated high density moderate leak (OCP1)

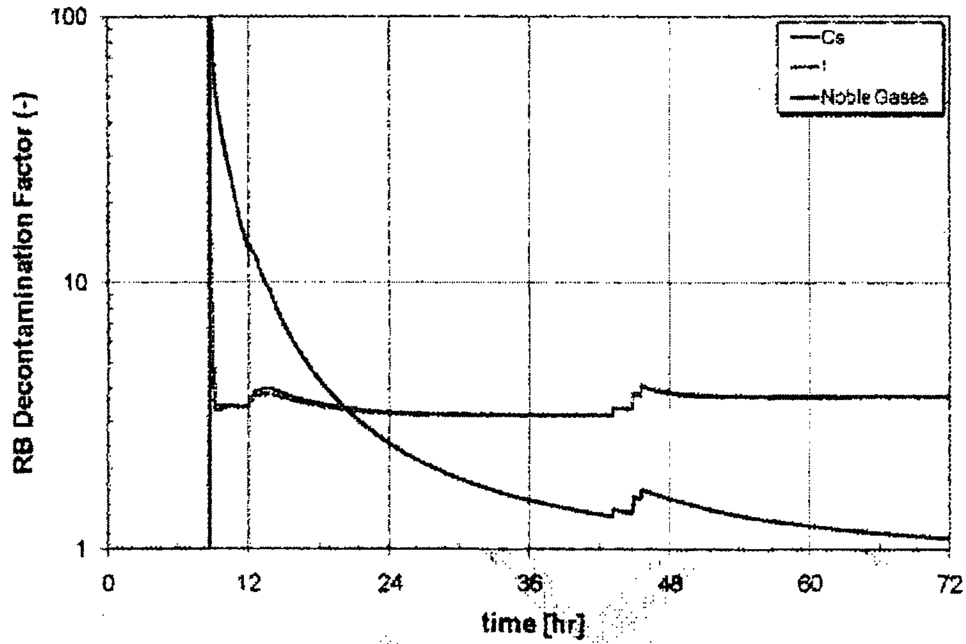

Figure 69: RB decontamination factor for unmitigated high density moderate leak (OCP1)

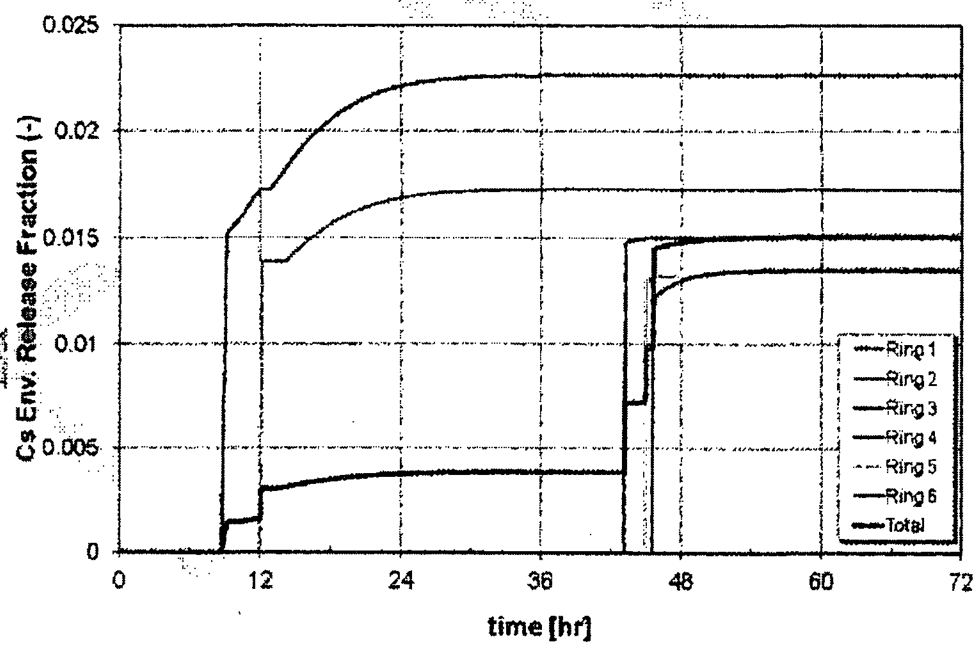

Figure 70: Cs environmental release fraction for unmitigated high density moderate leak (OCP1)

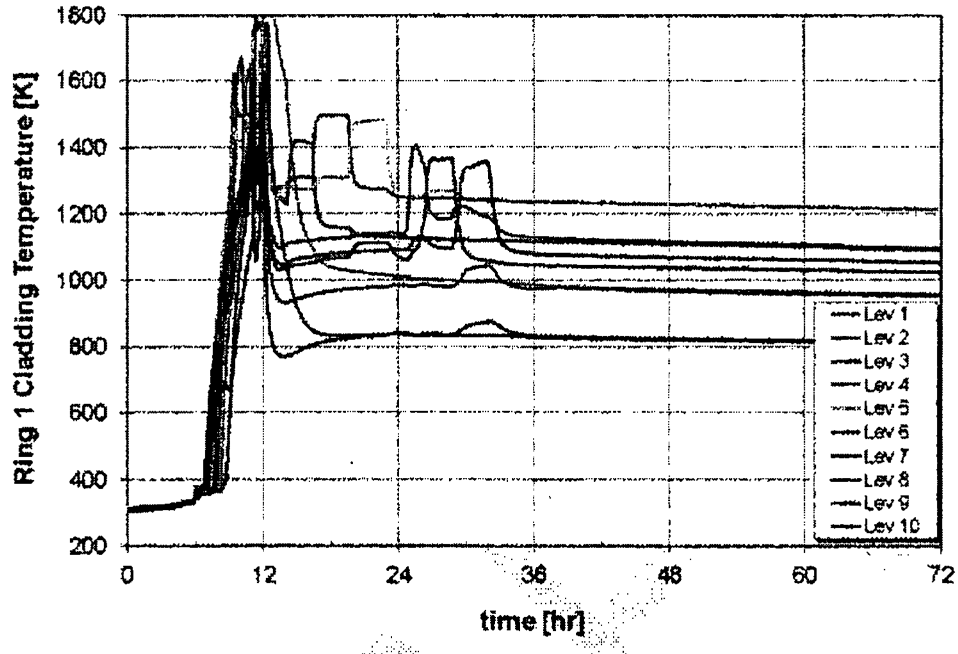

Figure 71: Ring 1 clad temperature for unmitigated low density moderate leak (OCP1)

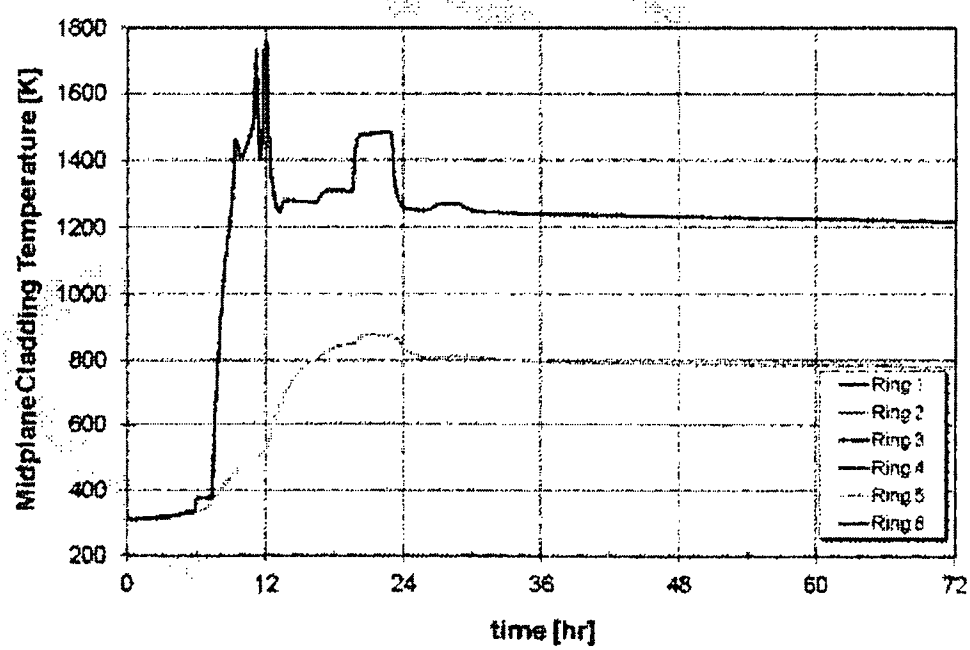

Figure 72: Midplane clad temperature for unmitigated low density moderate leak (OCP1)

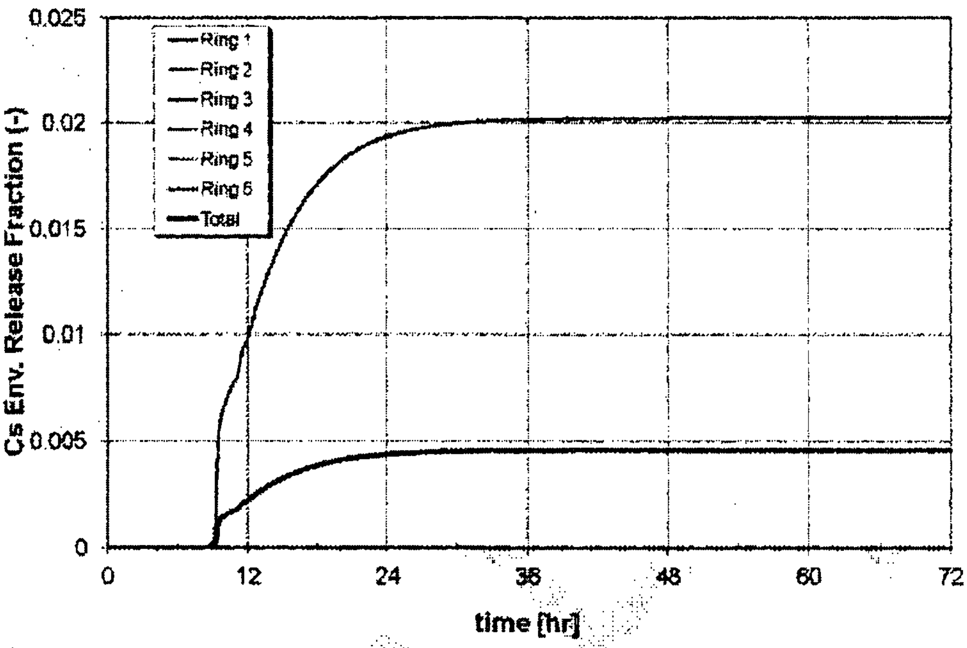

Figure 73: Cs environmental release fraction for unmitigated low density moderate leak (OCP1)

<u>Mitigated Moderate Leak (OCP1) Scenario</u>

The response of the pool to the mitigated scenario is shown in Figure 74. Because of the connectivity between the reactor and SFP and the additional volume of water, there is a relatively slow drain down and at the end of the mitigation deployment, the water level in the pool is more than 0.9 m above the top of the rack. [36] Therefore, instead of spray, mitigation is by direct injection into the pool. After about 12 hours, the water level remains relatively constant and the leak rate is balanced by the injection into the pool. The lower portions of the fuel remain cool and covered with water. Although there is heat up of the fuel (see Figure 75), there is no indication of a Zr fire and propagation through the pool. The peak fuel temperature reached 1200 K at 16 hours and remained near that value through 72 hours. A combination of radial heat transfer within the assembly, radial heat transfer from the recently discharge, high temperature fuel to adjacent fuel assemblies, and steam cooling due to boiling in the bottom of the assemblies between cells keep the fuel temperature near 1200K. Only Ring 1 had cladding failure and subsequent releases of the gap inventory as shown in Figure 76. All other fuel was below the threshold for cladding failure and fission product releases.

The clad temperature in Ring 1 for the low density case is shown in Figure 77. The heatup rate for the low density case is more extreme than the high density as was observed for the

[36] The level was close to 0.9 m above the top of the fuel of the fuel at the timing of the deployment of the sprays (i.e., 9.5 hours). If the spray system was used, cooling would be provided to uncovered portion of the fuel. The accident could have benefitted from natural circulation of air through the racks once the water level dropped below the rack baseplate and spray cooling from the top.

mitigated cases. Unlike the high density case, the low density case did not have low decay heat fuel assemblies adjacent to the recently discharged assemblies. Since an air natural circulation pattern through the racks was not established, the empty cells isolated the high decay heat assemblies and contributed to the higher heat up. The fuel in Ring 1 went through an oxidation transient, which led to peak fuel temperatures of 1800 K. However, once the steam in the assembly was consumed, the fuel temperatures dropped to 1200 K. The subsequent behavior was driven mainly by the decay heat which was very similar to the high density case. Because of higher fuel temperatures during the initial oxidation transient, there is slightly more release in the low density case.[37]

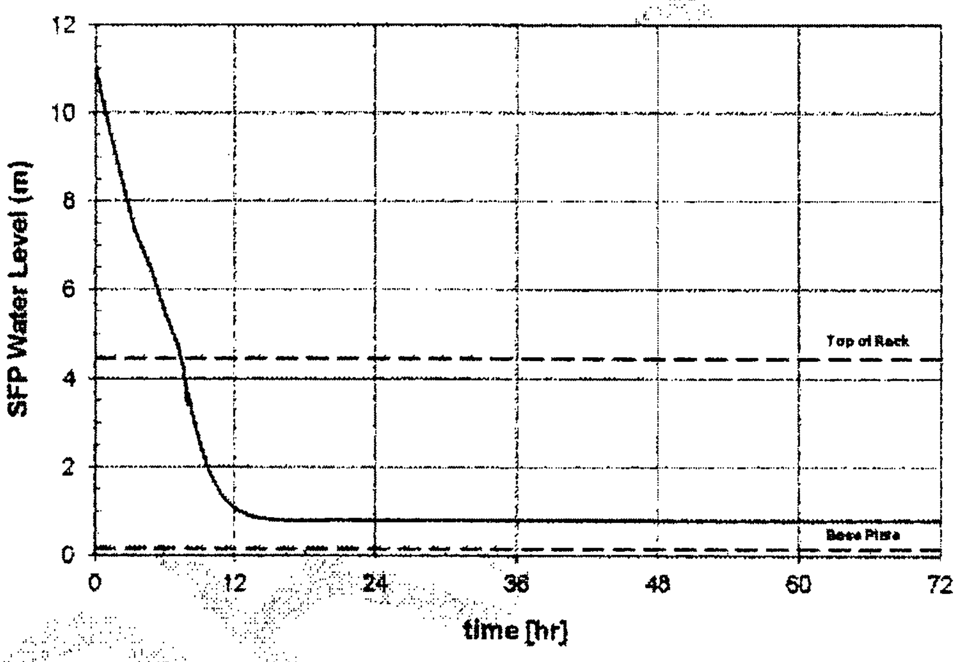

Figure 74: Water level for mitigated high density moderate leak (OCP1)

[37] As of this writing, this calculation was still running. However, the final results are not expected to be significantly different as the fuel temperatures continue to decrease and hydrogen concentration (<5%) does not support ignition.

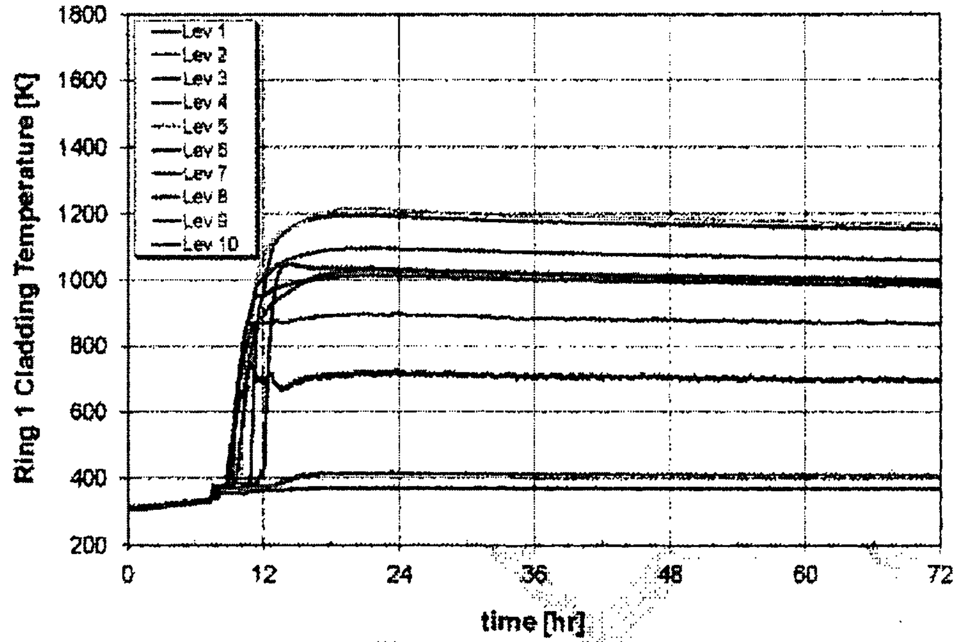

Figure 75: Ring 1 clad temperature for mitigated high density moderate leak (OCP1)

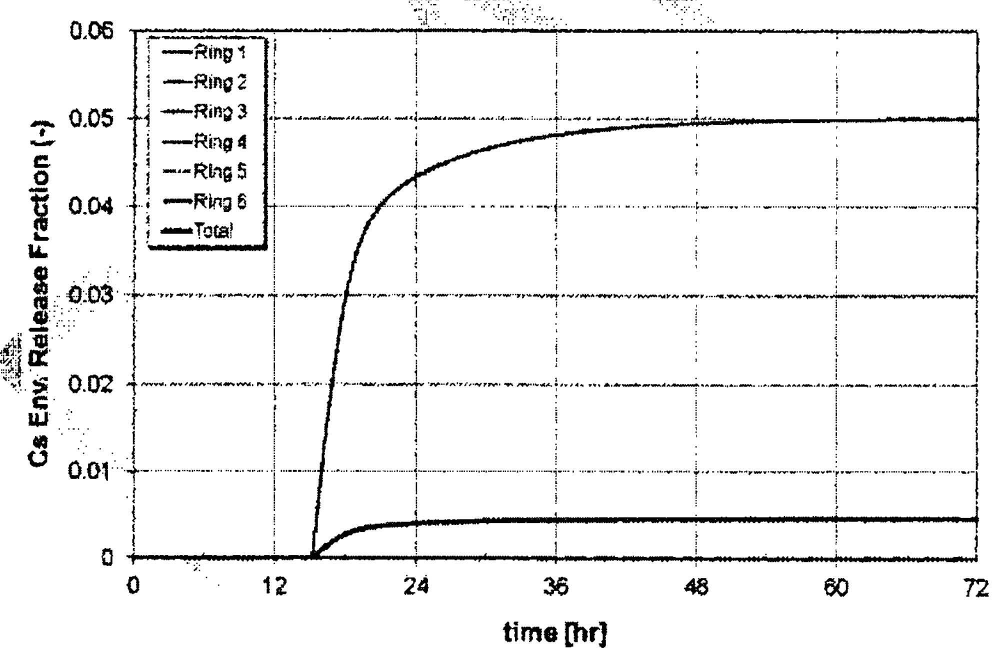

Figure 76: Cs environmental release fraction for mitigated high density moderate leak (OCP1)

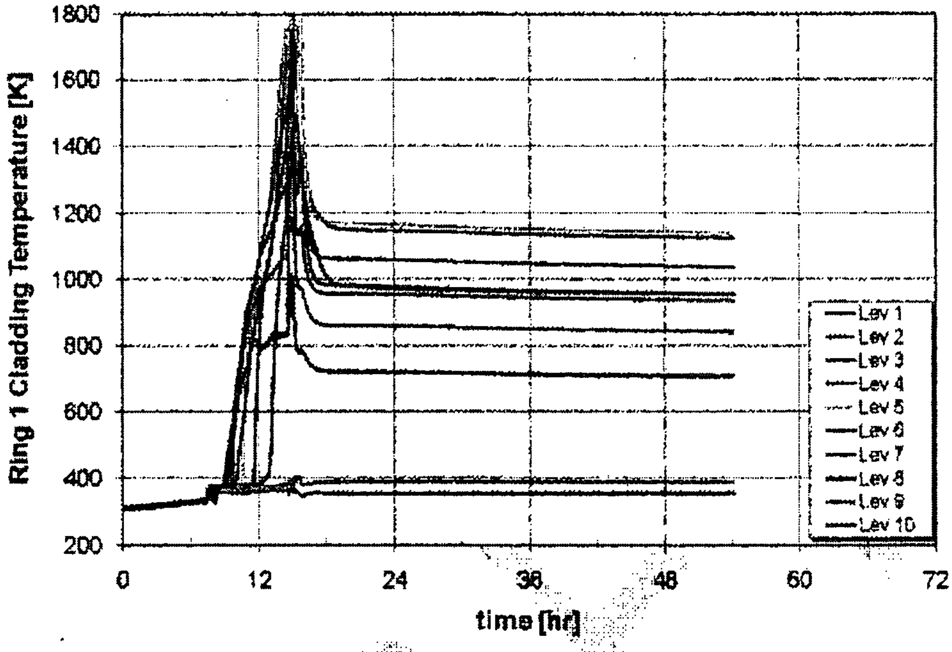

Figure 77: Ring 1 clad temperature for mitigated low density moderate leak (OCP1)

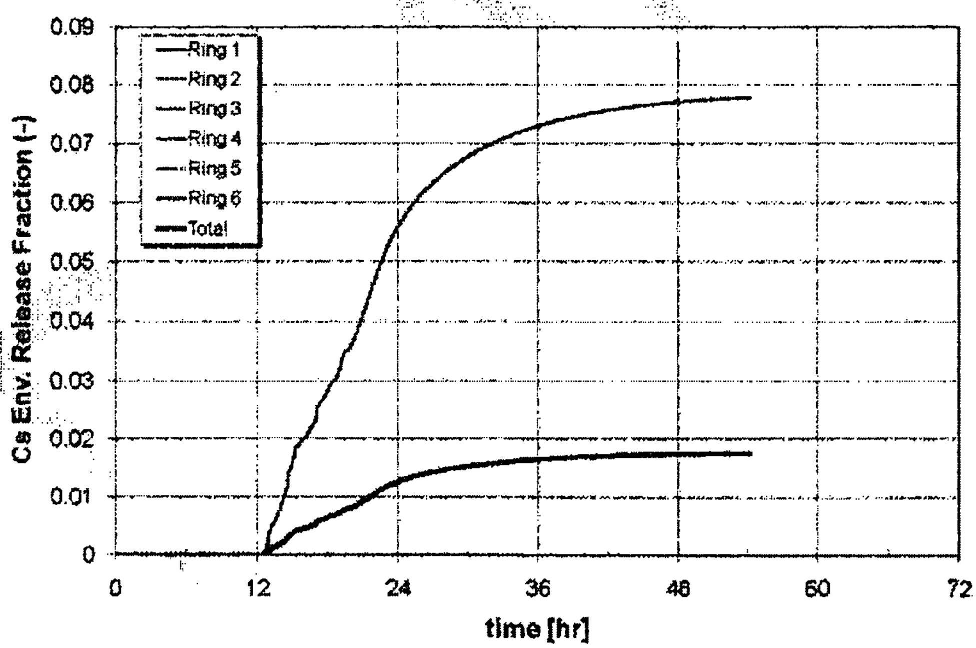

Figure 78: Cs environmental release fraction for mitigated low density moderate leak (OCP1)

Unmitigated Small Leak (OCP2) Scenario

This scenario involves a hydrogen combustion that caused a late rapid air oxidation. Figure 79 shows the slow drain down of the pool exposing the top of the racks at 42.6 hours compared to 18.7 hours for post-outage scenarios (see Figure 52). The decay power and the oxidation are shown in Figure 80. The air oxidation power reaches an order of magnitude higher than the decay heat during the oxidation transient after 60 hours. The fuel heat up begins after the water level reaches about fuel midplane, (see Ring 1 response in Figure 81). The high temperature fuel in Ring 1 heats the surrounding low decay heat fuel in Ring 2 as shown in Figure 82.

The evolution of reactor building steam and air shows that by the time the water level reached the SFP gate and the SFP is disconnected from the reactor, the building is filled with steam which continues to decrease as it is condensed on heat structures. The hydrogen concentration builds up until it reaches 10% at 65 hours and combusts. At the time of combustion, all the necessary conditions are satisfied; the hydrogen concentration is 10%, the oxygen is 10%, and the steam is less than the 55% threshold for inerting. The hydrogen combustion is sufficient to fail the blowout panels and the roof allowing fresh air to enter the refueling room. The fresh air circulates into the SFP, which leads to a rapid fuel heatup and failure in Rings 1 and then Ring 2. The RB decontamination approaches unity (Figure 84) resulting in about 17% Cs release to the environment (Figure 85).

The response for the low density case was similar but less severe than the high density case. The spacing of the fuel with empty rack cells reduced the propensity for propagation of the heat from the highest decay heat assemblies to the other assemblies in the SFP. The response of the highest decay heat assemblies in Ring 1 is shown in Figure 86. The peak fuel temperatures were less than 1400 K. As shown in Figure 87, the fuel in Ring 3 had a similar response but the fuel in Ring 5 was substantially lower. Due to the fewer fuel assemblies and lower peak temperatures, there was less oxidation and less hydrogen generation. The peak hydrogen concentration was well below the threshold for combustion. The overall cesium release is an order of magnitude lower (1.7%) than the high density case.

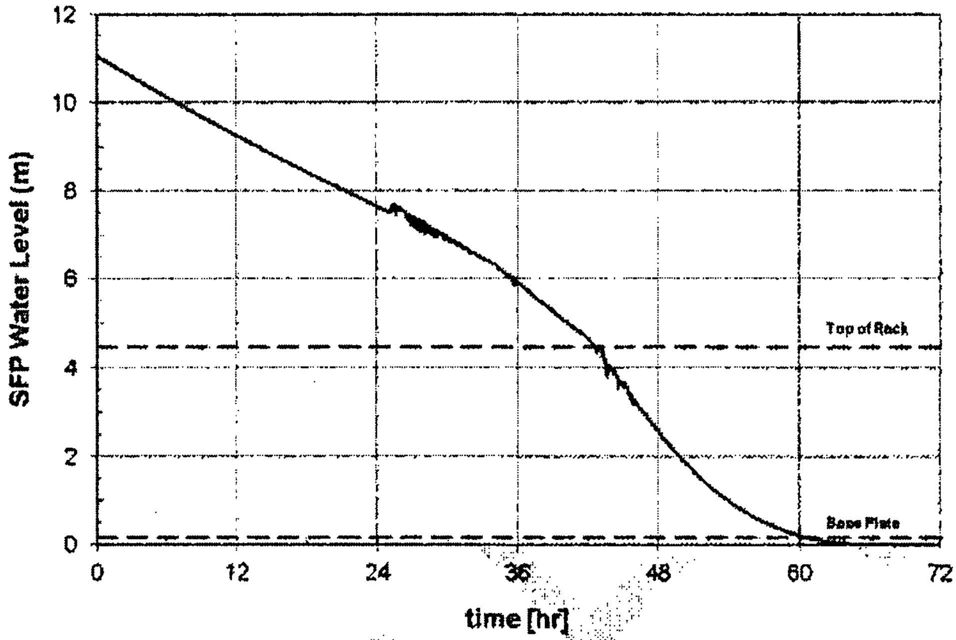

Figure 79: Water level for unmitigated high density small leak (OCP2)

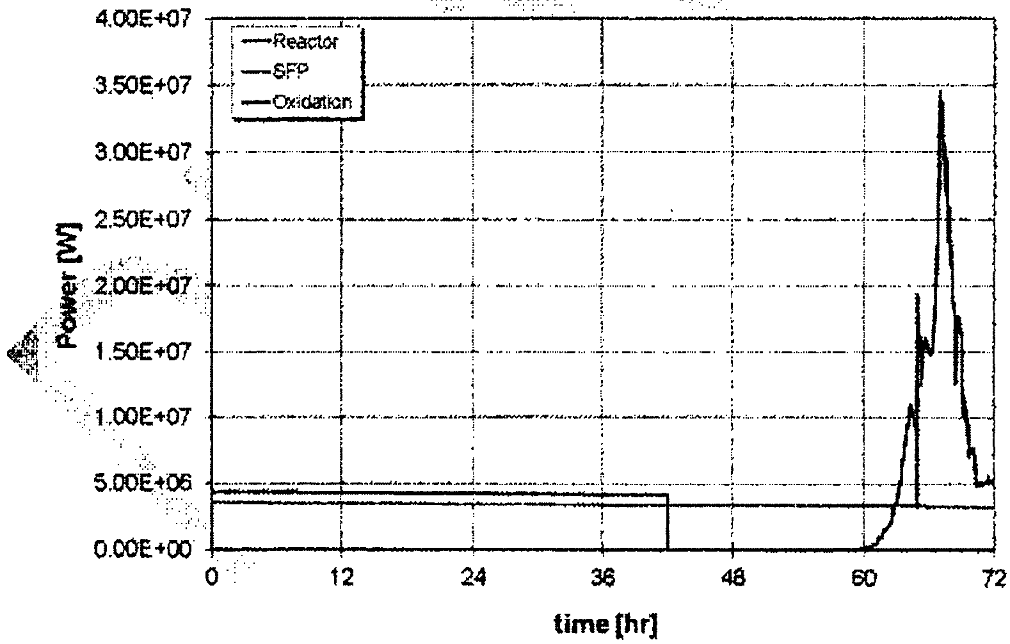

Figure 80: SFP power for unmitigated high density small leak (OCP2)

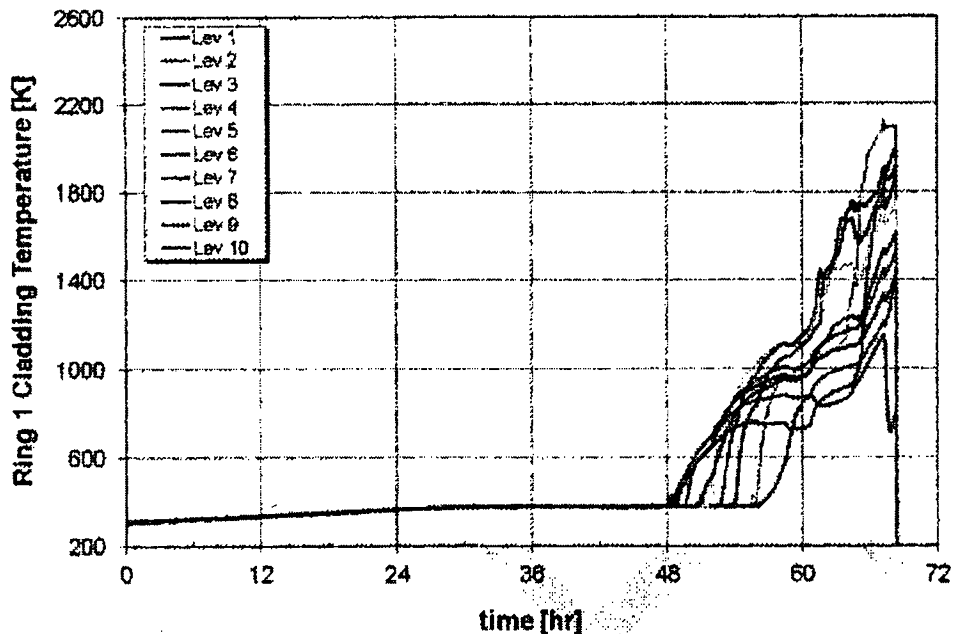

Figure 81: Ring 1 clad temperature for unmitigated high density small leak (OCP2)

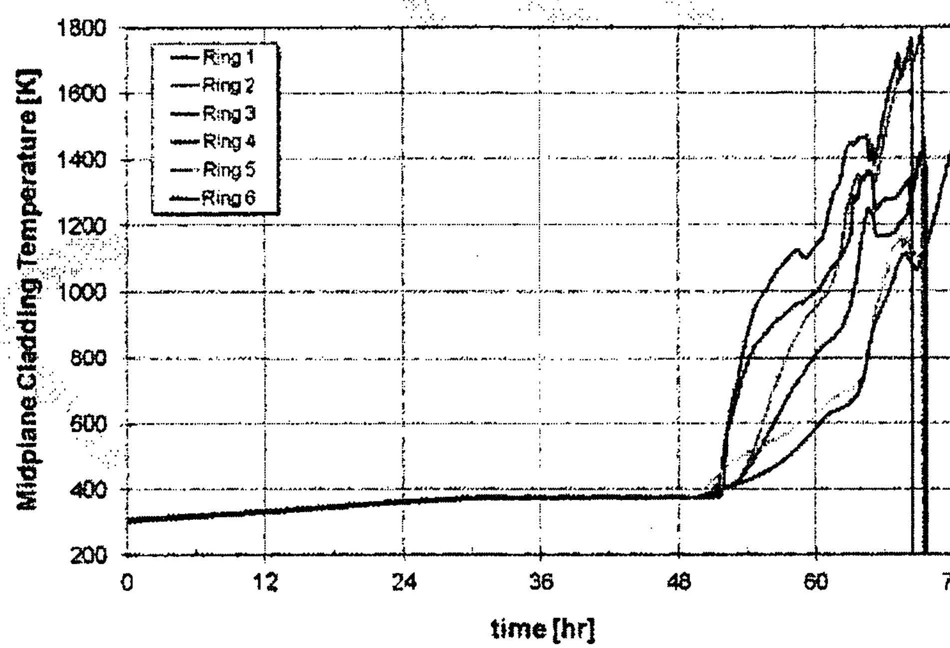

Figure 82: Midplane clad temperature for unmitigated high density small leak (OCP2)

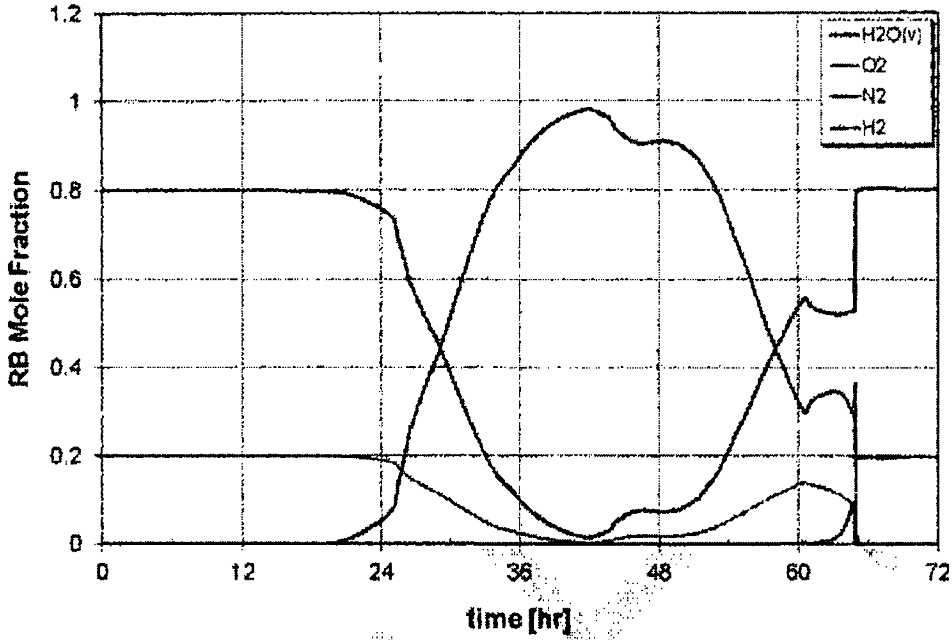

Figure 83: RB mole fraction for unmitigated high density small leak (OCP2)

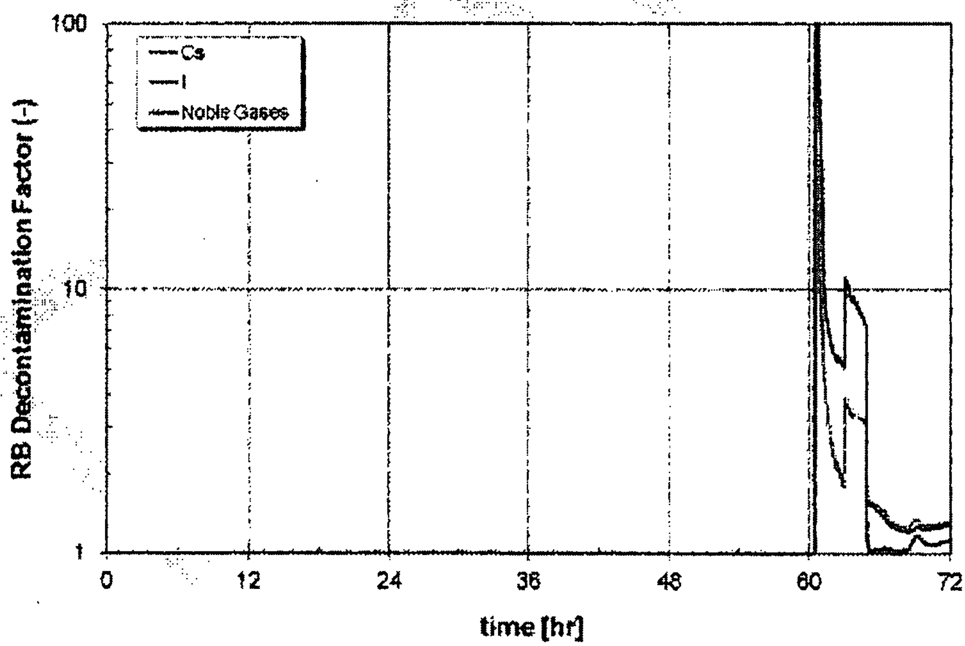

Figure 84: RB decontamination factor for unmitigated high density small leak (OCP2)

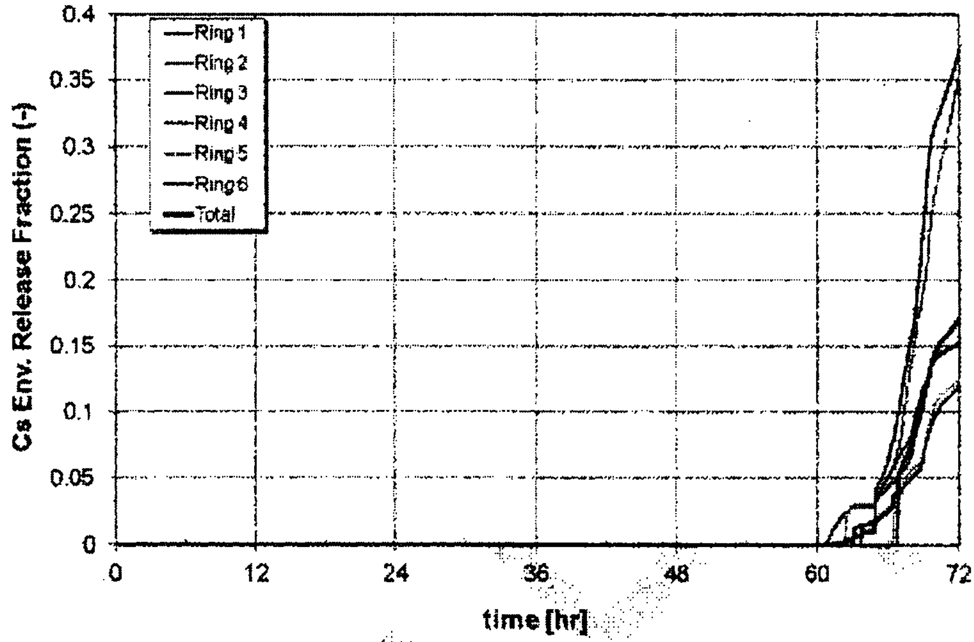

Figure 85: Cs environmental release fraction for unmitigated high density small leak (OCP2)

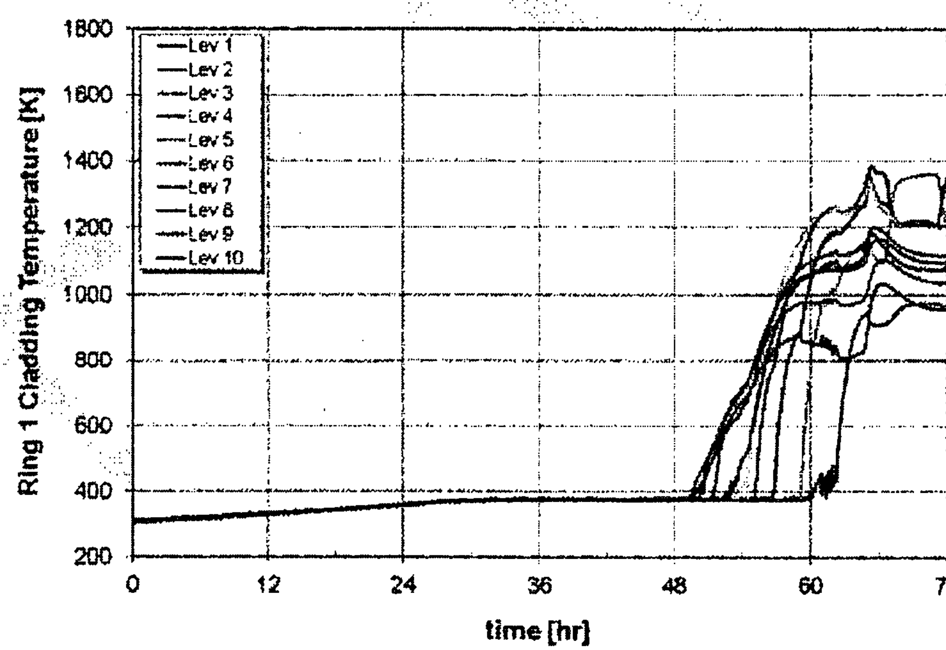

Figure 86: 1 Ring 1 clad temperature for unmitigated low density small leak (OCP2)

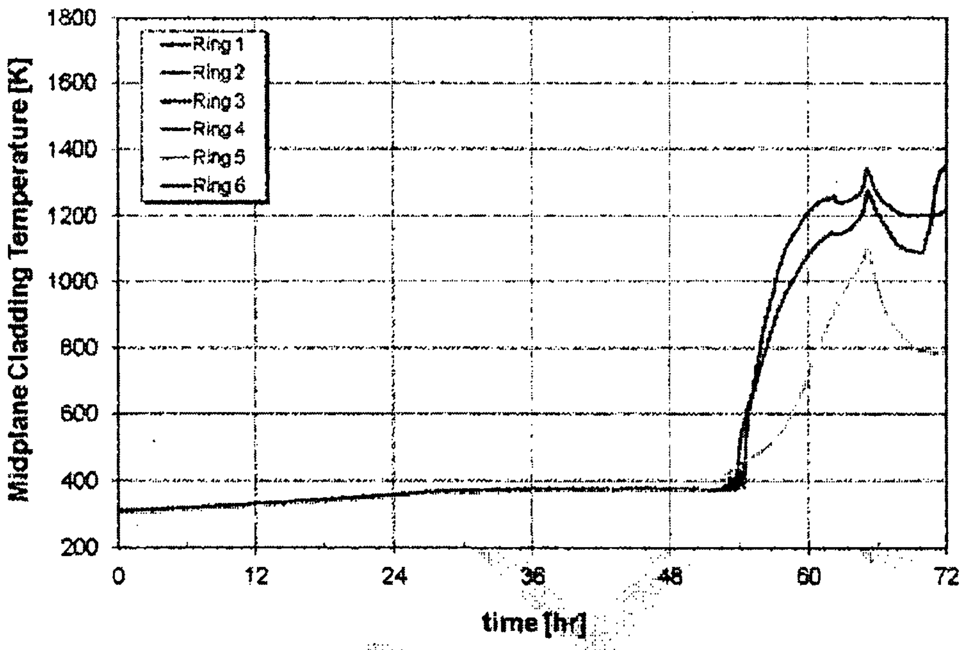

Figure 87: 1 Midplane clad temperature for unmitigated low density small leak (OCP2)

Unmitigated Moderate Leak (OCP3) Scenario

The response of the fuel temperature in Ring 1 for the unmitigated post-outage scenario OCP3 is shown in Figure 88 (compare to Figure 65 for OCP1). The heatup starts sooner because there is less water to drain and the approach to a Zr fire is more gradual because of lower decay heat (i.e., by a factor of 2.5, see Table 26) and the natural circulation of air through the assemblies. However, once the Zr fire is started, the maximum temperatures are comparable in both cases. As shown in Figure 88, the flame front starts at Level 5 but then moves slowly to Level 4, Level 3, and then to Level 2. After the peak temperature at Level 4, the peak temperature in the flame front decreases with each successive level. Radial heat transfer from the fuel racks to the SFP wall (Figure 89), the buildup of the oxide layer on the fuel, and the depletion of the oxygen in the reactor building (Figure 90). After 24 hours, the fuel temperatures in Ring 1 are relatively stable. There was no hydrogen combustion in this calculation. Although the peak hydrogen concentration reaches 8%, there was insufficient oxygen for combustion once the hydrogen concentration exceeded 7% (i.e., a lower threshold for ignition with an active ignition source, see Figure 90). When the hydrogen concentration peaks at 8%, the oxygen concentration is only 3% and well below an amount sufficient for combustion.

The temperature profiles for the low density case are shown in Figure 91. The low density temperatures are about 400 K lower than the high density case with the total Cs release being about 0.1% compared to 0.7% in the high density case. Similar to the previous OCP2 case, the low amount of fuel and the empty rack cells reduced the magnitude of hydrogen and the cesium release.

A sensitivity was performed to examine the effect of higher vapor pressure for the air oxidizing Ru releases. Figure 92 for the default Ru release model Figure 93 for the enhanced Ru release model used in the present study show that there is an order of magnitude difference in Ru release.[38] All the calculations under air oxidizing conditions in this work were based on the enhanced Ru release model.

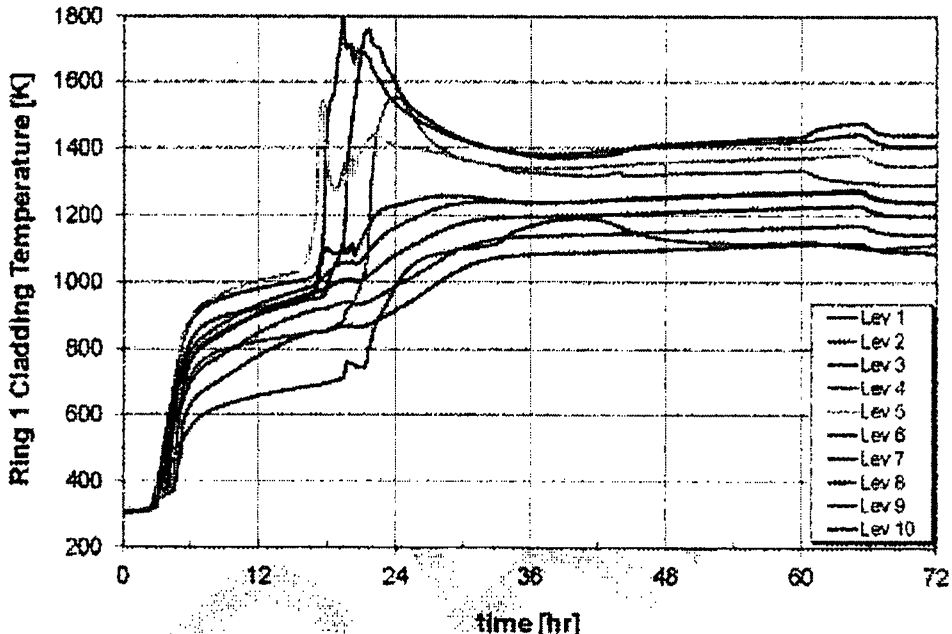

Figure 88: Ring 1 clad temperature for unmitigated high density moderate leak (OCP3)

[38] However, Ru release differences could be higher for scenarios in OCP 1 & 2.

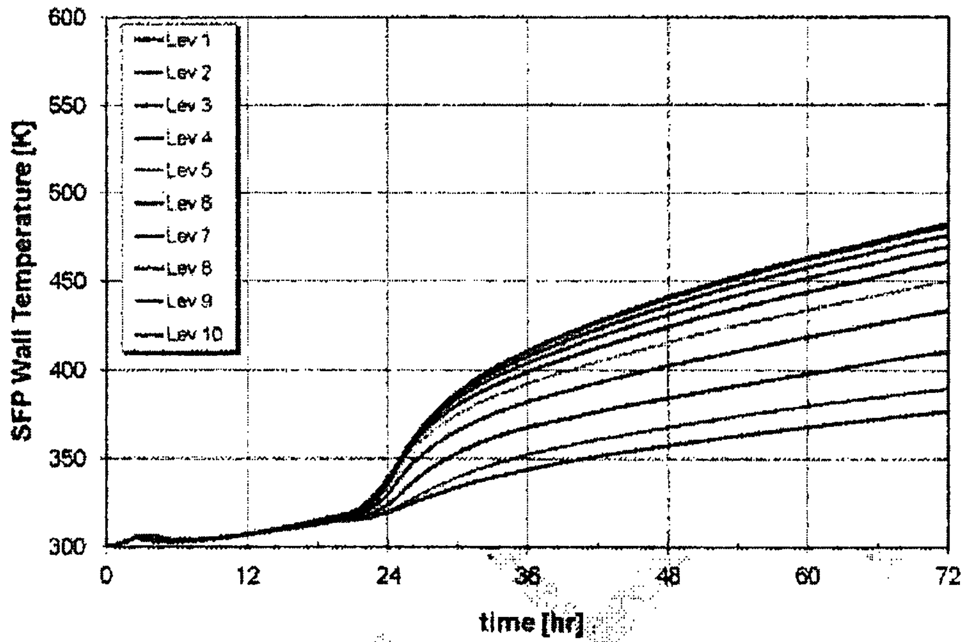

Figure 89: SFP wall liner temperature for unmitigated high density moderate leak (OCP3)

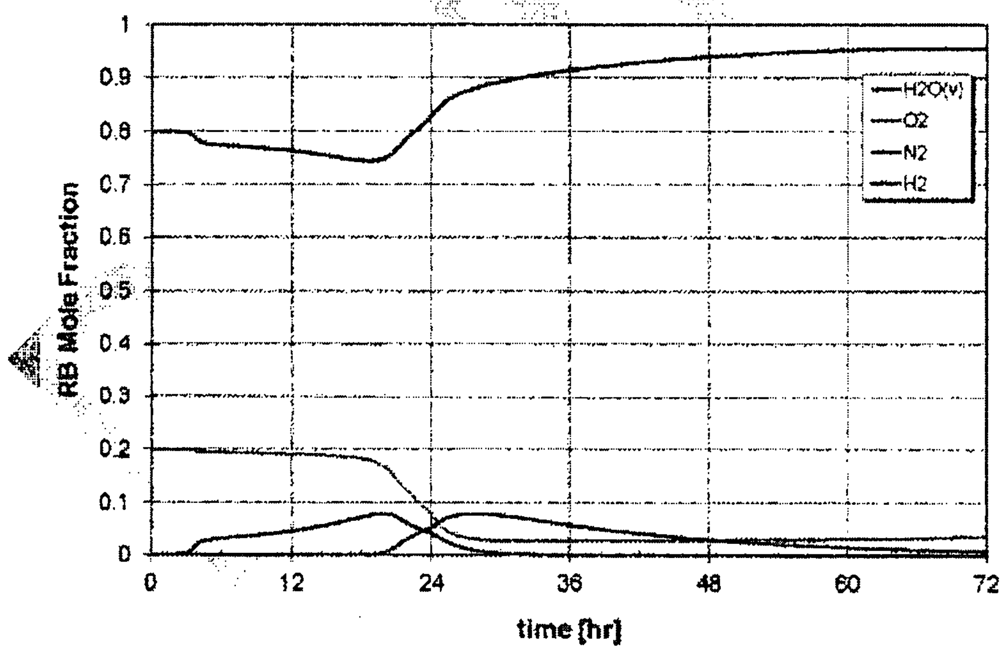

Figure 90: RB mole fractions for unmitigated high density moderate leak (OCP3)

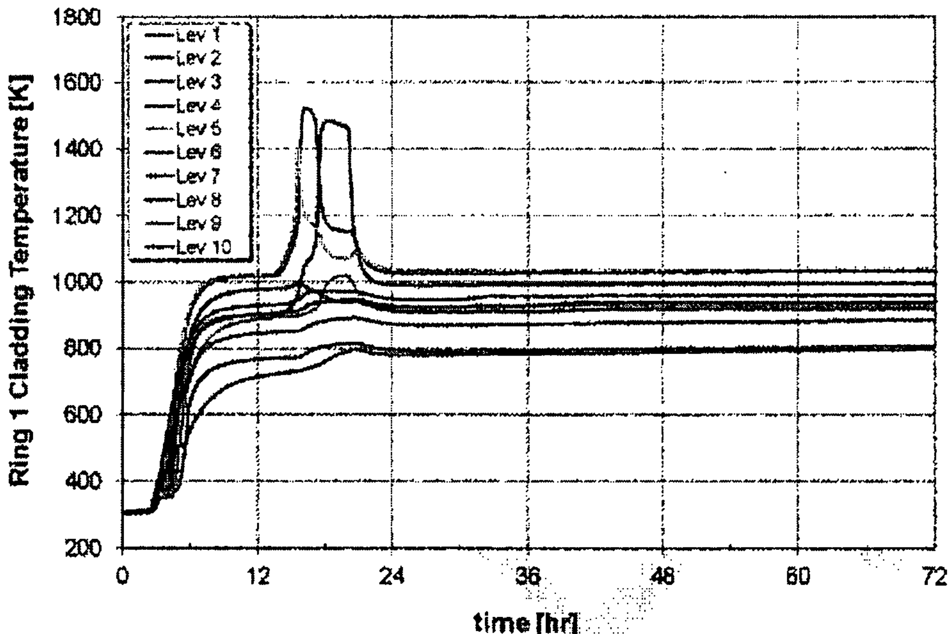

Figure 91: Ring 1 clad temperature for unmitigated low density moderate leak (OCP3)

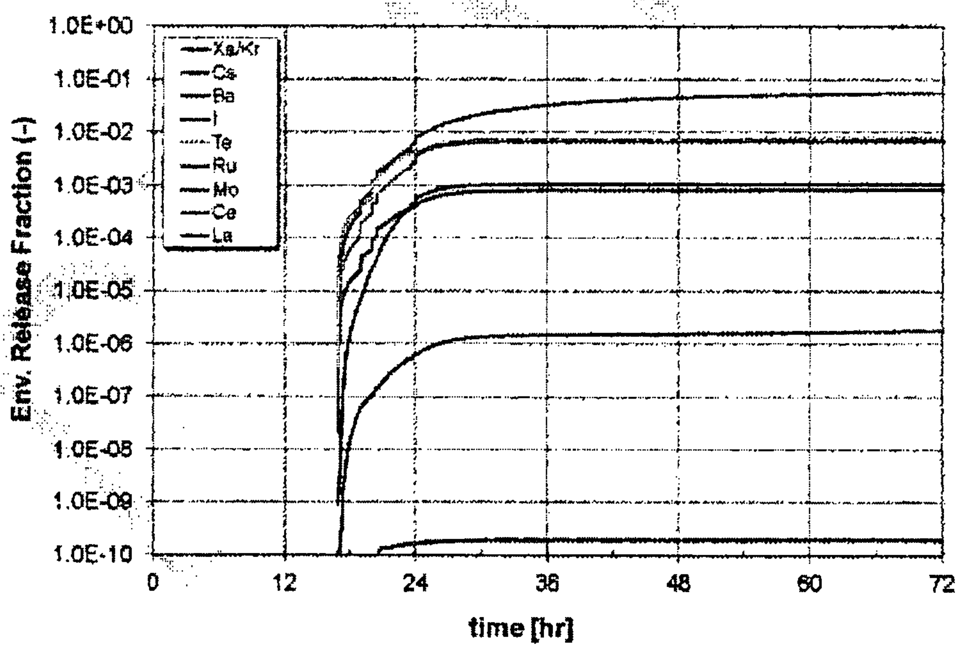

Figure 92: MELCOR default Ru release for unmitigated high density moderate leak (OCP3)

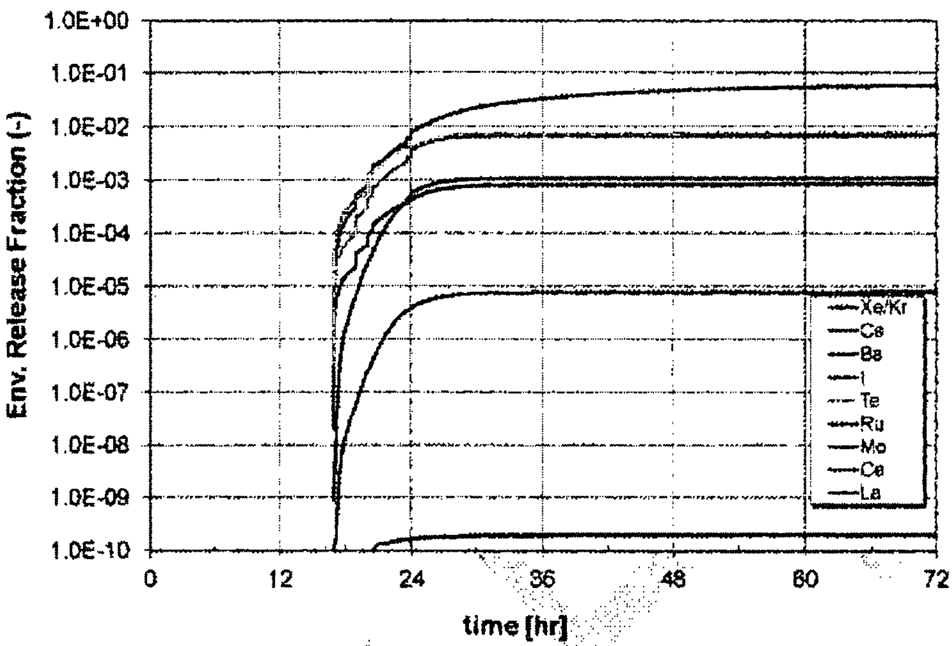

Figure 93: MELCOR enhanced Ru release under air oxidizing conditions for unmitigated high density moderate leak (OCP3)

6.3.3. Source Terms for Offsite Consequence Analysis

A summary of the release characteristics and key events is provided in Table 28 for the high density scenarios and Table 29 for the low density scenarios. More detailed discussion of key phenomena for selected sequences was provided in previous sections. The releases are binned for offsite consequence analysis to be described in Section 7.

For the high density loading, all the mitigated scenarios have no release, either because the makeup exceeds the leak rate as in the small leak cases or the mitigation is successful to limit the fuel heat up and avoid gap release. All the scenarios that do not involve a hydrogen deflagration have relatively low releases since the depletion of the oxygen limits clad oxidation and fuel heatup. A building failure results in air ingress into the assemblies and late phase rapid oxidation.

None of the scenarios in the low density cases had hydrogen combustion and the releases were relatively small. In the absence of hydrogen deflagration, the release fractions for both high density and low density cases are generally comparable. One exception is the low density OCP 1 cases that had higher release fractions than the high density cases in some instances. This was due to more rapid heatup of the fuel in Ring 1 due to less efficient heat transfer to the outer assemblies. Clearly the inventories in the low density configuration are lower and for the same release fractions, the released activity would be lower. Overall, for the moderate leaks, the low density cases lead to earlier gap release due to larger inventory of water (assemblies removed)

resulting in longer times for clearing the baseplate. The gap release first occurs in Ring 1 (hot assemblies) that has the same decay heat in both high density and low density configurations.

6.3.4. Accumulation of Water Elsewhere in the Reactor Building

The reactor building MELCOR model is simplified (see Figure 40). Therefore, all water leakages corresponding to the SFP damage and drain down and overflow from water accumulation due to condensation are directed directly to the environment. The model does not track the flow of the water and accumulation in other parts of the reactor building.

Table 28: Summary of release characteristics for high density scenarios

High Density Case #	Scenario Characteristics					Release Characteristics			
	SFP Leakage?	50.54(hh)(2) Equipment?	Fuel Uncovery (hr)	Gap Release (hr)	Hydrogen Deflagration (hr)	Cs released at 72 hours?	Cs-137 (MCi) Released	I release at 72 hours	I-131 (MCi) Released
OCP # 1	None	Yes							
	None	No							
	Small	Yes							
	Small	No	39.7	54.2	No	0.6%	0.33	3.5%	0.27
	Moderate	Yes	7.4	15.1	No	0.5%	0.26	5.0%	0.39
	Moderate	No	5.9	8.7	No	1.5%	0.80	2.1%	0.16
OCP # 2	None	Yes							
	None	No							
	Small	Yes							
	Small	No	42.6	60.5	64.8	17.1%	7.90	17.1%	1.91
	Moderate	Yes	[redacted]						
	Moderate	No	5.9	11.6	No	1.6%	0.73	2.0%	0.22
OCP # 3	None	Yes							
	None	No							
	Small	Yes							
	Small	No	18.7	40.6	[illegible]	42.0%	24.20	51.2%	0.73
	Moderate	Yes	[redacted]						
	Moderate	No	2.5	16.9	No	0.7%	0.39	0.7%	0.01

Table 29: Summary of release characteristics for low density scenarios

Low Density Case #	Scenario Characteristics					Release Characteristics			
	SFP Leakage?	50.54(hh)(2) Equipment?	Fuel Uncovery (hr)	Gap Release (hr)	Hydrogen Deflagration (hr)	Cs release at 72 hours	Cs-137 (MCi) Released	I release at 72 hours	I-131 (MCi) Released
OCP # 1	None	Yes	[redacted]						
	None	No							
	Small	Yes							
	Small	No	40.3	54.7	No	3.1%	0.33	4.6%	0.36
	Moderate	Yes	7.4	12.6	No	1.8%	0.19	7.0%	0.55
	Moderate	No	5.9	8.7	No	0.6%	0.05	1.7%	0.13
OCP # 2	None	Yes	[redacted]						
	None	No							
	Small	Yes							
	Small	No	43.1	59.2	No	17%	0.28	3.3%	0.37
	Moderate	Yes	[redacted]						
	Moderate	No	5.9	10.5	No	0.4%	0.07	0.7%	0.08
OCP # 3	None	Yes	[redacted]						
	None	No							
	Small	Yes							
	Small	No	18.8	41.6	No	0.6%	0.10	1.2%	0.02
	Moderate	Yes	[redacted]						
	Moderate	No	2.5	15.2	No	0.1%	0.02	0.2%	0.00

7. OFFSITE CONSEQUENCE ANALYSIS

In the unlikely event of a severe accident that might damage the spent fuel pool (as detailed in the previous sections), a release of radioactive material from the nuclear power plant site into the atmosphere might occur. Such a release of radioactive material is expected disperse from the site through the atmosphere and to the surrounding population, by expanding and moving downwind. After modeling the onsite accident progression and potential mitigation measures, the MACCS2 code (MELCOR Accident Consequence Code System, version 2) is used to model offsite release and consequence of radioactive material.

MACCS2 [1] has been developed by Sandia National Laboratories (SNL) over the past two decades. It has the ability to evaluate the impacts of atmospheric releases of radioactive aerosols and vapors on human health and on the environment. The MACCS2 code has the ability to use site-specific weather conditions, population data, and evacuation plans to calculate and model the radiation exposure of the population through all of the relevant dose pathways: cloudshine, inhalation, groundshine, and ingestion.

Over the past decade, Sandia National Laboratories has also developed WinMACCS for the NRC. WinMACCS is a user-friendly front-end to MACCS2 that facilitates selection of input parameters and sampling of uncertain input parameters and performs post-processing of results after the calculation.

7.1. MACCS2 Modeling Overview

For the purposes of the SFP Scoping Study, many of the input values for offsite release and consequence modeling are based on NUREG-1935. This modeling for NUREG-1935 was in turn were based on previous studies such as NUREG-1150, an expert elicitation of the NRC/Commissions of the European Communities (CEC) to update certain at transport and dose parameters [9] and an update of the dose response modeling to be consistent with the latest federal guidance report at the time (FGR-13)[10].

Similarly, the scope and modeling decisions for this portion of the study are similar to NUREG-1935, except in instances where a severe accident from a spent fuel pool is expected to differ from a reactor and where the objectives of the SFPSS are different (e.g. reporting of land contamination).

NUREG-1935 documents and justifies the modeling parameters used in that previous study. This study gives a general, high level understanding of the offsite consequences modeling. But because the offsite consequence modeling for the SFP Scoping Study closely resembles NUREG-1935, this report will instead focus on the specific areas that are different from this previous research study.

7.2. Offsite Consequence Scope

7.2.1. Atmospheric Transport

The project focuses on atmospheric environmental releases. Atmospheric releases are most likely to have an immediate impact on the public, and therefore this study treats these releases as having primary significance. Release of contaminated water, on the other hand, has a more delayed impact that can be mitigated much more easily than atmospheric releases. This study follows the current state-of-practice for consequence analyses, which focuses on the more immediate, more difficult to mitigate, atmospheric release path. It is important to note that MACCS2, as used in this study, also accounts for deposition onto surface water and runoff of contaminants into surface water. These contribute to potential doses from drinking water.

Other types of releases, with longer-term consequences, are clearly possible. For example, accidental release of contaminated water took place during the Fukushima accident. Much of this water immediately flowed into the ocean and presumably some entered the groundwater. For a similar release at an inland site, freshwater resources, such as rivers or lakes, could be impacted. Future studies would benefit from more comprehensive analyses that consider releases of contaminated water and longer-term migration through groundwater.

7.2.2. Dose Truncation (in the Dose Response Model)

Experts generally agree that it is difficult to characterize cancer risk because of the low statistical precision associated with relatively small numbers of excess cases at low doses. This limits the ability to estimate trends in risk. From an epidemiological standpoint, the number of LCFs attributable to radiation exposure from accidental releases from a severe accident would not be statistically detectable above the normal rate of cancer fatalities in the exposed population (i.e., the excess cancer fatalities predicted are too few to allow the detection of a statistically significant difference in the cancer fatalities expected from other causes among the same population). From an epidemiological standpoint, in most, if not all, cases the number of latent cancer fatalities (LCFs) attributable to radiation exposure from accidental releases from a severe accident would not be statistically detectable above the normal rate of cancer fatalities in the exposed population (i.e., the excess cancer fatalities predicted are too few to allow the detection of a statistically significant difference in the cancer fatalities expected from other causes among the same population).

In the absence of additional information, the International Commission on Radiological Protection (ICRP), National Council on Radiation Protection and Measurements (NCRP), the U.S. National Academy of Sciences, and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) have each indicated that the current scientific evidence is consistent with the hypothesis that a linear, no threshold (LNT) dose response relationship exists between exposure to ionizing radiation and the development of cancer in humans. Although most scientific organizations do not rule out the possibility of LCFs from very low doses, some organizations such as the Health Physics Society (HPS), French National

Academy of Medicine, and the ICRP, consider the use of an LNT dose response model to calculate LCFs from very low doses below a certain threshold inappropriate. While some scientific organizations take this position, few organizations have endorsed a definitive dose threshold, other than LNT, to calculate LCFs. The HPS concludes that quantitative estimates of risk should be limited to individuals receiving a whole body dose greater than 0.05 Sv (5 rem) in 1 year or a lifetime dose greater than 0.1 Sv (10 rem) in addition to natural background radiation. While the NCRP supports the LNT model, it also recommends binning exposures into ranges and considering those ranges separately.

Given the uncertainty of low dose health effects, this study suggests using a range of annual dose truncation values, ranging from LNT to the HPS recommendation (5 rem/yr and 10 rem lifetime). One intermediate annual dose-truncation level is also to be analyzed, that being the 620 mrem/yr (which is the other is the U.S.-average annual dose to the public from medical and background radiation exposures). Depending on the time available, additional truncation values may also be analyzed.

This approach is similar to that used in NUREG-1935. NUREG-1935 used the same dose truncation levels; however the approach also analyzed an annual dose truncation of 10 mrem. In that study, this very small dose truncation has shown to produce up to a factor of two difference in health effects from LNT for relatively very small releases. However, more typical results from this truncation tend not to deviate from LNT and therefore the truncation is not expected to provide very meaningful results for this new study either.

This approach differs from many past analyses from before NUREG-1935 that have assumed an LNT dose response model. An LNT dose response model is consistent with the current NRC regulatory approach, and the NRC is neither changing nor contemplating changing radiation protection standards or policy as a result of the approach to be taken in this SFP Scoping Study. Given the uncertainty of low dose health effects, results for these four dose-truncation levels (which include LNT) are to be reported without bias for each of the accident scenarios.

7.2.3. Distance Truncation (From Point of Release)

The reported distances are based on the presentation of the results, which in turn are based on the priority of objectives for the study. The project is careful not to use a distance truncation in a manner that artificially reduces the consequence numbers, if and when such a consideration is applicable.

Reported land contamination includes the entire site region and is not truncated due to distance. Health effect risk, which is higher near the point of release, is primarily reported for 0 to 10 miles. Both health effect risk and land contamination as a function of distance is reported up to the ingestion pathway emergency preparedness zone (50-miles), or a distance that which no more protective actions are predicted, whichever is further.

Due to the possibility of truncating consequences for certain dose response models, concerns over the effect of low doses (such as low doses at large distances) are not captured in the form of a distance truncation. Instead, the effect of low doses is explicitly captured in the different dose response modeling.

NUREG-1935 reported LCFs out to 50 miles. Most previous analyses (e.g.. NUREG-1150) calculated LCFs to 1,000 miles with forced deposition (i.e. boundary weather) to account for all non-noble gas radionuclides in the dose calculation.

7.2.4. Time Truncation of Release

The time truncation used for this research project is 72 hours (or 48 hours, if at this point in time, the spent fuel pool is still covered with water).

In the event of a prolonged severe accident, radiation and other hazards could make any truncation of an ongoing SFP release challenging. On the other hand, many resources are available at the State, regional, and national level that could be available to mitigate an accident. Considering both viewpoints on this issue, project staff judge 72 hours to be a reasonable time truncation. The use of a time truncation is a point of uncertainty that can significantly affect the results. See Section 5 for additional discussion on mitigation assumptions in this study, and Section 8 for proposed future sensitivity analyses.

The use of offsite response resources could be comparable, in some respects, to the use of 50.54(hh)(2) equipment. For instance, offsite response resources could be used in similar ways to truncate the accident, such as water makeup or water spray onto the SFP. However, unlike the onsite 50.54(hh)(2) actions, offsite response actions could be more ad hoc in implementation.

The reliability of deployment and operation of the 50.54(hh)(2) equipment and offsite response actions is not being quantitatively measured in this project. These issues will receive additional attention in the upcoming Site Level 3 PRA research study (SECY-11-0089), in the context of human reliability analysis.

7.3. Offsite Consequence Modeling Inputs

The parameters found in this list are of select, high-level data and modeling inputs to the MACCS2 models. The list is not meant to be comprehensive. In order to organize the list, the following broad categories have been created: emergency response, dose, economic, emergency and long-term phases. However, the parameters found in the emergency and long-term phase could be considered the "remaining" parameters, as emergency response, dose, and economic parameters would also technically fall into one of these two phases of the accident. (Some are applicable to both.)

7.3.1. Atmospheric Transport & Dispersion Modeling

Source Term:

A source term definition for MACCS2 is created for each accident consequence calculation using MELMACCS [6]. MELMACCS reads a MELCOR plot file and extracts information useful for MACCS2.

In addition, a number of user options have to be input when using MELMACCS. This includes the related radionuclide chemical groups or classes to be included in the analysis (i.e., the Xe, Cs, Ba, I, Te, Ru, Mo, Ce, and La groups), as well as the related radionuclide inventory (e.g. plant-specific burnup) in order to make the analysis consistent with the MELCOR calculation.

Weather Considerations:

Because MACCS2 is primarily a tool for probabilistic risk assessment (PRA), it accounts for the weather uncertainties that are inherent to accidents that might occur at any time in the future. Thus, the modeled results represent the average consequences attributable to the variability in the weather.

Weather Sampling:

The weather-sampling strategy for this study uses a non-uniform weather-binning approach in MACCS2. Weather binning is an approach used in MACCS2 to categorize similar sets of weather data based on wind speed, stability class, and the occurrence of precipitation. This approach, which allows the user to specify a different number of random samples to be chosen from each bin, has been available since MACCS2 was first released [1] but was not commonly used in the past. This sampling strategy was chosen as a means of improving the statistical representation of the weather. Further discussion on this can be found in NUREG-1935, for which the use of this strategy is based.

Weather Data:

Meteorological data used for this project consisted of 1 year of hourly meteorological data (8,760 data points per site for each meteorological parameter). This was primarily accomplished via a cooperative effort with the licensee using onsite meteorological tower observations for NUREG-1935. Peach Bottom provided 2 years of weather data, including directly measured hourly precipitation data. Stability class data were derived from temperature measurements at two elevations on the site meteorological towers. The specific year of data chosen for each reactor was 2005, as was based on data recovery (greater than 99 percent being desirable) done for NUREG-1935. Different trends (e.g., wind-rose pattern and hours of precipitation) between the years were estimated to have a relatively minor (<25 percent) effect on the final NUREG-1935 results. More specific details of the weather data can be found in NUREG-1935.

Aerosol Deposition Velocities:

Dry deposition velocities have been updated to account for a more typical surface roughness of 60cm for the Peach Bottom site. (A surface roughness of 20cm was used for NUREG-1935.) The relative aerosol deposition velocities, as well as much of the non-site-specific data for acute health effects, is developed from a set of reports that document a joint NRC/Commission of the European Communities (CEC) expert elicitation study [7].

Aerosol deposition velocities, combined with the different aerosol sizes from MELCOR, determine the rate aerosols are depleted from the plume of radionuclides. Generally, the larger aerosols deposit more quickly and so are depleted more rapidly from the plume. The peak in the aerosol size distribution is usually a few micrometers, which corresponds to a deposition velocity of a few millimeters per second.

Plume Segments:

Because MACCS2 allows plume segments to travel in only one compass direction, as determined by weather data, atmospheric releases of radionuclides are discretized into (at longest) 1-hour plume segments. The MELCOR analyses provide the amount each radionuclide contributes to each aerosol bin for each plume segment. Consideration of each radionuclide, instead of just each chemical group, is a significant advancement for offsite consequence modeling developed for this project.

Many plume segments can better represent plume transport and dispersion due to possible changes in the weather (such as the wind direction) during the release. Finer resolution of these releases was not necessary to maintain the fidelity of the calculation.

7.3.2. Exposure and Dose Modeling

Exposure Pathways:

The project considers groundshine, cloudshine, inhalation, and ingestion dose exposure pathways. The principal exposure pathway to members of the public occupying land contaminated by deposits of radioactive materials from reactor incidents is expected to be exposure of the whole body to external gamma radiation. Although it is normally expected to be of only minor importance, the inhalation pathway would contribute additional doses to internal organs. [14]

Shielding Factors:

NUREG-1935 reviewed the shielding factors applied to evacuation, normal activity, and sheltering for each dose pathway (e.g. groundshine) used in NUREG-1150 [2] and NUREG/CR 6953, Vol. 1 [9]. The review updated just one factor, that being the fraction of time the average

person spends indoors for normal activity (81 percent). This study uses the same shielding factors as NUREG-1935.

Census Data:

Site population data has been updated to 2011. This was done with the latest version of the code SECPOP2000 [8]. SECPOP2000 uses a multiplier to account for population growth and the 2000 Census in order to create site data for MACCS2. A multiplier value of 1.1051 from the United States Census Bureau was used to account for the average population growth in the United States from 2000 to 2011.

The new site file was created for 16 compass sectors, which is the only angular resolution supported by SECPOP2000. WinMACCS (the preprocessor for MACCS2) can interpolate these site files onto a 64 compass-sector grid for better special resolution during the MACCS2 offsite analysis.

Dose Coefficients (DCFs):

The DCFs to be used in this study are based on Federal Guidance Report (FGR)-13 [10]. With the DCFs, organ-specific doses can be calculated from exposure to radiation. In conjunction with organ-specific risk factors, total health effects can be calculated. This study plans to follow NUREG-1935 as the basis of calculating health effects.

NUREG-1935 recommends using dose to the pancreas as a surrogate for dose to soft tissue to estimate residual cancers. The reason for the choice of the pancreas dose coefficient for the 'residual' cancer sites is because it serves as a reasonable surrogate for the residual group for both external radiation fields and the intake of radionuclides. Because MACCS2 does not currently read the data for the pancreas from the dose conversion factor file, a workaround was created. Values of the dose coefficients for the pancreas were copied into the organ called bladder wall. Thus, residual cancers are associated with the organ called bladder wall, which actually contains data for the pancreas. The inhalation factors in FGR-13 were processed to account for a distribution of particle sizes. An activity median aerodynamic diameter (AMAD) of 1 micron was assumed with a log-normal form for the distribution and with a geometric standard deviation of about 2.5. MACCS2 does not currently allow the dose coefficient to change as the particle size distribution changes during the MACCS2 calculation. The particle size distribution might change because of changes in the emitted particle sizes for different plumes or changes as the plumes progress downwind and deposit material.

Risk Factors for Latent Cancer Fatality:

In estimating health effects from a severe accident, NUREG-1935 calculated the radiation exposure to the population and then applied dose-response models to analyze early fatality and latent cancer fatality risks. NUREG-1935 used latent cancer expression coefficients for the US population as detailed in U.S. Environmental Protection Agency's publication "*Estimating*

Radiogenic Cancer Risks"[11] and implemented in the EPA's Federal Guidance Report 13, "*Cancer Risk Coefficients for Environmental Exposure to Radionuclides*" (FGR-13)[10]. These risk factors include seven organ-specific cancers plus residual cancers not accounted for directly. In 2009, the National Research Council released the BEIR VII report, an additional study of the biological effects of ionizing radiation. No one-to-one correspondence exists between the cancers reported in BEIR VII compared with the earlier BEIR V report. Therefore, the dose coefficients of tissues of the body in FRG-13 may or may not be consistent with the BEIR VII cancer sites. Thus, this study is following the basis used in NUREG-1935 in deciding to await EPA's review of BEIR VII and subsequent update of FGR-13 before implementing BEIR VII risk coefficients.

Biological Effectiveness Factors (BEFs):

FGR-13 also recommended changes to the BEFs for alpha radiation for two of the organs used to estimate latent cancer health effects to be consistent with the way the risk factors for cancers associated with those organs were evaluated. The two organs are bone marrow and breast; for these organs, the BEFs for alpha radiation were changed from the standard value of 20 to 1 and 10, respectively. Doses to these organs are used to evaluate occurrences of leukemia and breast cancer, respectively. The choice of BEFs for these tissues is dictated by the EPA [11].

Dose and Dose Rate Effectiveness Factor (DDREF):

A DDREF is to be applied to all doses in the late phase modeling of the offsite consequence calculation and to those doses in the early phase modeling (i.e. the first week) that were less than 20 rem to the whole body. This factor accounts for the fact that protracted, low doses are perceived to be less effective in causing cancer than acute doses. The DDREF for all cancers except for breast was 2.0 and for the breast was 1.0, as done in NUREG-1935.

Acute Health Effect Parameters:

Parameters that relate to acute health effect parameters to be used for this study, as well as much of the non-site-specific data used for consequence is taken from a set of reports that document a joint NRC/Commission of the European Communities (CEC) expert elicitation study [7]. (All of the input parameters extracted from the expert elicitation are median values.)

7.3.3. Emergency Phase Modeling

The MACCS2 models are setup to calculate exposure in two distinct phases: the emergency phase and the late phase. The emergency phase models calculate the emergency phase dose and associated health effects to the public, as well as the emergency preparedness protective measures that protect the public. The chosen time period for the emergency phase begins with the initiating event and continues for 1 week. The time length of this phase has been chosen to

ensure that the effects of the plumes have been captured, and in order to capture all the calculated acute exposures (without exclusion to smaller exposures).

As required by 10 CFR Part 50, offsite response organizations, (OROs) develop emergency response plans for implementation to protect the public health and safety in the unlikely event of an accident at a nuclear power plant. These response plans are developed for the plume exposure pathway and ingestion pathway emergency planning zones (EPZs) which are described in NUREG 0654/FEMA REP 1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," [NRC, 1980]. The plume exposure pathway EPZ is the area of about 10 miles around a nuclear power plant. Within the EPZ, detailed emergency plans are in place and this planning provides a substantial basis for expansion of response efforts if necessary [NRC, 1980].

The analysis of the SFP accidents for the Peach Bottom site allows the use of the advanced consequence modeling techniques developed and implemented in SOARCA [NRC, 2012a; NRC, 2012b]. Specifically, the offsite response to the accident may be realistically modeled in the MACCS2 consequence model. This includes modeling the protective action decisions of OROs and the implementation of these protective actions by individual population segments. The project used a normal weather winter, weekday scenario that includes schools in session.

The SFP project identified many potential accident sequences and performed preliminary consequence modeling to identify the boundary conditions for more detailed modeling. These preliminary results were used to identify areas that may be potentially affected. Some of the results showed elevated doses at distances well beyond the plume exposure pathway EPZ requiring review of the state emergency response plans to assist in determining the types of protective actions that may be implemented in these areas. The results of the preliminary modeling were binned to support an efficient use of detailed consequence modeling to determine the potential effects of such accidents. The binning tally, presented in Table 30 was based on the Environmental Protection Agency (EPA) protective action guides (PAGs) [14]. For this project, if the 4 day projected dose (early phase) is expected to exceed 1 rem for a member of the public, the PAG is exceeded. In NUREG-1935, a 7 day period was used for the early phase.

Table 30: Binning of Expected Dose Projection

PAG	Sequence Tally
PAG not exceeded beyond 10 miles	8
PAG exceeded beyond 10 miles but not beyond 20 miles	5
PAG exceeded beyond 20 miles	7

The distance to which the PAG may be exceeded assisted in determining the extent of offsite protective actions and the type of protective actions that may be implemented such as sheltering in lieu of evacuation. For each of the accident sequences, it was determined that a General Emergency (GE) would be declared promptly, within 15 minutes, based on the emergency action levels for the operating reactor. The timing of significant radiological release varied among the accident sequences and was an important factor in the response modeling.

The timing of significant radiological release occurs well after evacuation of the EPZ would be accomplished according to the site evacuation time estimate. Sensitivity calculations for fuel configurations not used at Peach Bottom predict earlier release, but even these do not begin until evacuation is well underway or completed within the EPZ.

Using the dose projection binning, three consequence models were considered for analysis as shown in Table 31. Time constraints limited the current analysis to two of these including consequence model 1 and 2.

Table 31: Modeling Sequences

Consequence Model	Release to Environment	EPZ	Area beyond EPZ
1	Small: Does not exceed PAG beyond EPZ.	Immediate evacuation	Shadow evacuation of 30% of the public from this area. Hotspot and normal relocation if required. Habitability is applied.
2	Large Late: exceeds PAG beyond EPZ.	Immediate evacuation	Shadow evacuation of 30% of the public from this area. Delayed evacuation to a distance of 30 miles. Shelter in Place of 30 to 40 mile area. Shadow evacuation of 30% of the public from this area Shelter-in-place (SIP). Hotspot and normal relocation if required. Habitability is applied.
3 Not Complete	Large Early: exceeds PAG beyond EPZ.	Immediate evacuation	TBD

The protective action terms described in Table 31 are defined below:

<u>Immediate Evacuation:</u> Residents evacuate the affected area. The residents are modeled as cohort groups to represent the difference in response timing associated with these cohort groups.

<u>Shadow Evacuation:</u> A shadow evacuation occurs when members of the public evacuate from areas that are not under official evacuation orders and typically begin when a large scale evacuation is ordered [NRC, 2005]. Although shadow evacuations are often reported and observed there is little quantitative data available. In a national telephone survey of residents of EPZs, about 20 percent of people that had been asked to evacuate had also evacuated for situations in which they were asked not to evacuate [NRC, 2008]. These situations were typically related to hurricane hazards. In the SFP project, the initiating event is an earthquake that will be felt by residents of the EPZ. The event will be followed with media information related to an accident at the nuclear power plant, wide spread loss of power and damage to some buildings. It was assumed that these factors would increase the shadow evacuation to 30% of the public in the environs of the plant.

<u>Hotspot and Normal Relocation:</u> Models are included in the MACCS2 code to reflect OROs relocating people from areas that were not included in the evacuation order where the dose

exceeds PAGs. Within the MACCS2 calculation, individuals who would be relocated because their projected total committed dose from these pathways is projected to exceed the protective action criteria are prevented from receiving any additional dose during the emergency phase. The emergency phase is the 4 day period after the start of the release. This relocation dose criterion is applied at a specified time after plume arrival at the affected area and is applied to the entire population within the analysis area, including the non-evacuating cohort within the EPZ. The dose and time values were developed specific to each sequence and are provided below.

- Hotspot: 5 rem at 8 hour
- Normal: 1 rem at 12 hour.

The delay to relocate affected populations models the projected availability of ORO radiological resources to locate and survey areas of potential PAG exposure. This timing would vary depending upon the status of the EPZ evacuation and significant radiological release. The timing could be shorter for releases that occur long after the General Emergency declaration due to the increased availability of response resources from corporate, local, state and national organizations.

<u>Shelter-in-place:</u> For those areas where dose may be reduced below the PAG through SIP, rather than evacuation, SIP is modeled as an expected protective action. The emergency plans for both states allow for this protective action.

For those sequences in which the dose exceeded the PAGs beyond the EPZ, analysis of greater populations than were considered in SOARCA was required. The population was divided into multiple cohorts to better represent the response of the public. A cohort is a population group that mobilizes or moves differently from other population groups.

The site specific evacuation time estimate (ETE) was used to inform the development mobilization times for the public and travel speeds. To model evacuation in WinMACCS, each cohort was loaded onto the roadway network at a specified time, and a single speed was used. However, evacuations occur as a distribution in which the percent of public evacuating the area increases over time until all members of the public have evacuated. Evacuations are typically represented as a curve that is relatively steep at the beginning and tends to flatten as the last members of the public exit the area. The point at which the curve tends to flatten occurs when approximately 90 percent of the population has evacuated. The last 10 percent of the population is called the evacuation tail (Wolshon, 2010) which was modeled as a separate cohort.

The site specific ETE provides information on population characteristics, mobilization of the public, special facilities, transportation infrastructure and other information used to estimate the time to evacuate the EPZ. A distance over time ratio was generally used to develop the speed of the general public cohorts. A distance of 13 miles was assumed as a maximum travel distance to provide for the fact that roadways are not directly radial away from the plant. Adjustment factors within the consequence model were then used to increase or decrease

speeds for each cohort at the grid level in areas where greater congestion may be expected or in rural areas where less congestion may occur. Because MACCS2 does not currently have the capability to move populations over time, cohorts are modeled to begin moving together at a specific time after notification. The analysis includes the State of Pennsylvania position that, if an evacuation is ordered, it will include the entire EPZ. This position differs from other states, where evacuation of downwind areas would be implemented rather than the full EPZ. For this project, a full evacuation was modeled assuming that the State of Maryland OROs would agree with the Pennsylvania protective action decisions.

Assumptions:

The following general assumptions were applied in this analysis:

- Protective actions will be implemented within the EPZ.
- Protective actions will be expanded beyond the EPZ.
- Dose projections will be developed and available to support protective action decisions.
- Residents will expect they cannot return and will take more belongings with them, increasing mobilization times.
- Residents are generally aware of an impending emergency through media broadcasts.
- For the delayed release sequences where a plume does not start for greater than 24 hours, it is assumed that schools beyond the EPZ would be closed rather than evacuated.
- It is assumed that evacuees from the EPZ are transported to safe distances. There are no modeling provisions for relocating evacuees from congregate care centers which may be located within 10 to 20 miles from the plant for sequences where doses exceed PAGs in these areas.
- There is no loss of power beyond 20 miles. Communications, traffic signals, and EAS messaging are not impacted in this area.

The chosen time period for the emergency phase begins with the initiating event and continues for 4 days. The time length of this phase has been chosen consistent with the EPA use of the early phase in developing PAGs. At this time, the effect of the plume is complete, and all the calculated acute exposures may be captured (without exclusion to smaller exposures).

Network Evacuation Model

The evacuation area was mapped onto the WinMACCS radial sector grid network. The roadway network within the EPZ was reviewed against the site-specific evacuation plan to determine the likely evacuation direction in each grid element. Travel directions were input at the grid level to approximate travel along evacuation routes and primary roadways. For evacuations beyond 20 miles, travel directions were set radially outward to implement evacuation in these areas. This method was necessary to meet the schedule and resource constraints. Speed adjustment factors were then applied at the grid element level to speed up vehicles in the rural uncongested areas and to slow vehicles in more urban settings where the modeling indicates that speeds are lower than the average values used in the analyses.

Potassium Iodide

The State of Pennsylvania and Maryland potassium iodide (KI) programs distribute KI tablets through several different means. The WinMACCS KI model only allows KI to be assigned by population fraction, not by location. Only residents within the EPZ would have KI and only about half of these would typically be assumed to have access to their KI and take it within the specified timeframe. The KI model was turned on for the sequences where the PAGs are not expected to be exceeded beyond the EPZ and was turned off for those sequences where PAGs would be exceeded beyond the EPZ due to computer model limitations.

Adverse Weather

Adverse weather is typically defined as rain, ice, or snow that affects the response of the public during an emergency. Adverse weather was addressed in the movement of cohorts within the analysis using the evacuation speed multiplier (ESPMUL) parameter in WinMACCS to reduce travel speed when precipitation is occurring (indicated from the meteorological weather file). The ESPMUL factor was set at 0.7, which effectively slows down the evacuating public to 70 percent of the established travel speed when precipitation exists.

Infrastructure Analysis

The limited seismic evaluation of the potential failure of roadway infrastructure conducted for SOARCA identified 12 bridges and roadway segments that could fail under the postulated conditions. The EPZ evacuation routes identified in the emergency plan indicate that evacuees west of the river would typically evacuate in a westerly or southerly direction, and evacuees east of the river would evacuate in a northerly or easterly direction. Thus, the loss of bridges and the Conowingo Dam Road crossing the Susquehanna River has a minimal effect on the evacuation time. The other bridges and roadways that fail in the earthquake serve sparsely populated areas where alternative roads are available. Alternate routes out of the EPZ have more than sufficient capacity to support the evacuating population.

The seismic event causes the loss of all onsite and offsite power, which can affect the response timing and actions of the public. Sirens would be sounded following the GE declaration, and because Peach Bottom will have a fully backed up siren system by about 2013, it was assumed sirens sound for this analysis. The loss of power affects the number of residents receiving instructions via EAS messaging. The residents within the EPZ will have felt the earthquake which will effectively serve as the initial warning. It may be expected that the residents will use multiple methods of communication, such as cell phones, telephones, websites, and direct interface to communicate the emergency message.

A review of the roadway network within the EPZ indicates that there are only a few traffic signals within the EPZ and that most intersections are controlled with stop signs. The loss of power will cause traffic signals to default to a four-way stop mode, which is less efficient than normal signalization. It is expected emergency response personnel would respond to these

intersections and direct traffic as indicated in the site ETE. Therefore, the loss of signalization will have a limited effect on the evacuation. It is assumed that at distances beyond 20 miles, there is no loss of power and traffic signals and EAS messaging are not impacted.

Using the information and approach described above, the evacuation timing and speeds for each cohort were developed and are presented in Table 32.

Table 32: WinMACCS response parameters for sequences where PAGs are not exceeded beyond the EPZ

Population			Response Delays (hours)				Phase Duration (hr)		Evacuation Travel Speeds (mph)		
	Cohort	Fraction	Siren (Oalarm)	Delay to Shelter	Delay to Evac	Total (Depart time)	Early (durbeg)	Middle (durmid)	Espeed (early)	Espeed (mid)	Espeed (late)
1	0 to 10 miles Shadow	0.3	1	0	0	1	1	0.5	20	15	5
	10 to 20 miles Shadow			2	1	4					
2	0 to 10 miles General Public	0.417	1	1	1	3	0.25	3	5	2	20
3	0 to 10 miles Special Facilities	0.008	1	0	4	5	0.5	0.5	2	15	20
4	0 to 10 miles Evacuation Tail	0.1	1	2	3	6	0.5	0.5	2	15	20
5	0 to 10 miles Schools	0.172	1	0	0.5	1.5	1	0.5	2	15	20
6	0 to 10 miles Non-evacuating Public	0.005	1	-	-	-	-	-	-	-	-

Selected input parameters for WinMACCS are described below:

- Delay to shelter (DLTSHL) represents a delay from the time of the start of the accident until cohorts enter the shelter.
- Delay to evacuation (DLTEVA) represents the length of the sheltering period from the time a cohort enters the shelter until the point at which it begins to evacuate.
- The speed (ESPEED) is assigned for each of the three phases used in WinMACCS including Early, Middle, and Late. Average evacuation speeds were derived from the Peach Bottom ETE report. Speed adjustment factors were then used in the WinMACCS application to represent free flow in rural areas and congested flow in urban areas.

- Duration of beginning phase (DURBEG) is the duration assigned to the beginning phase of the evacuation and may be assigned uniquely for each cohort.
- Duration of middle phase (DURMID) is the duration assigned to the middle phase of the evacuation and may also be assigned uniquely for each cohort.

WinMACCS response parameters for sequences where PAGs are not exceeded beyond EPZ.

The following cohorts were established for SFP project:

0 to 10 Shadow: A shadow evacuation may be expected from within the EPZ. Focus group work conducted with NUREG/CR 6953, Volume 2 (NRC, 2008) suggested that some residents are prepared and ready to evacuate at the first indication of an accident at the nuclear power plant. Results of the telephone survey conducted with NUREG/CR 6953 showed that on a national level, 20% of residents of EPZs have packed a 'go-bag' and are ready to leave. Although this may be more appropriately called an early or spontaneous evacuation because the population begins to evacuate prior to receiving an order to evacuate, it is defined here as a shadow evacuation. Because the accident is initiated by a severe earthquake, it is assumed 30% of the public evacuate.

10 to 20 Shadow: A shadow evacuation may be expected in the area beyond the EPZ. The distribution of the shadow evacuation would likely include a larger percentage of the public near the boundary of the EPZ and the percent would decrease proportional to the distance away from the EPZ. For this analysis, it is assumed that 30 percent of the general public from the 10 to 20 mile area shadow evacuate. This cohort will begin evacuating as they observe EPZ evacuees traveling through the area.

0 to 10 Public: This population group would evacuate as a distribution with some residents leaving promptly and others leaving later. For this analysis, the cohort is modeled as a single group.

0 to 10 Special Facilities: This is a small but unique population group within this EPZ. There is no delay to shelter because these residents are assumed to be in a robust facility when the accident begins. Specialized vehicles to evacuate these facilities take time to mobilize.

0 to 10 Tail: The tail represents the last 10% of the EPZ population who typically take a longer time to begin to evacuate.

0 to 10 Schools: This cohort includes elementary, middle, and high school student populations within the EPZ. Schools receive early and direct warning from OROs and have response plans in place to support busing of students out of the EPZ. While the movement of school children are specifically considered in the evacuation model, the dose models do not treat them differently than the average person.

Non-evacuating Public: A portion of the public does not follow protective action orders. It is assumed that 0.5% of the general public within the EPZ refuse to evacuate.

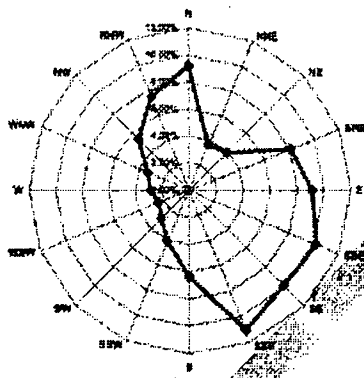

Figure 94: Peach Bottom Windrose

WinMACCS response parameters for late release sequences where PAGs are exceeded beyond the EPZ.

Preliminary data suggests that emergency phase doses may reach 1.15 rem in 30 to 40 mile ring. Under these conditions, the EPA PAGs would suggest evacuation to these distances. In this analysis, it is assumed that evacuation to 30 miles is completed and SIP is implemented in the 30 to 40 mile area which reduces the dose to the public below the PAG.

The population within the 30 mile radius is approximately 1.4 million, based on SECPOP. The population of the 40 mile radius is approximately 3.4 million. As these distances expand and the populations increase, it is important to better understand the potential direction of the plume. The Peach Bottom windrose in Figure 94 would suggest that the dominant wind direction is to the south and east generally toward lower population areas. Thus it is more likely that a much smaller population would be affected than this analysis considers.

For this sequence, a 30 mile radius is evacuated after the EPZ has evacuated. This would be implemented in a staged evacuation manner which is common for plume related emergency response. In addition, an SIP is assumed to be ordered for the 30 to 40 mile radius area.
It is assumed that 24 hours after the start of the accident, OROs begin to order evacuations beyond the EPZ. This is based on the preliminary dose timeline which indicates a large release beginning at 48 hours.

To develop the evacuation time estimate and corresponding speeds for the areas beyond the EPZ, it was assumed 90% of general public can be evacuated 24 hours after order to evacuate. This is consistent with the lengthy travel times observed in hurricane evacuations of similar populations. The last 10% (evacuation tail) takes an additional 12 hours longer.

Because of the lengthy time for this release to the environment, this sequence is effectively modeled as two separate evacuations including the EPZ first followed later by the 10 to 30 mile area.

The following cohorts were established for SFP project:

0 to 10 Schools: This cohort includes elementary, middle, and high school student populations within the EPZ. Schools receive early and direct warning from OROs and have response plans in place to support busing of students out of the EPZ.

0 to 10 Shadow: A shadow evacuation may be expected from within the EPZ. Although this may be more appropriately called an early or spontaneous evacuation because the population begins to evacuate prior to receiving an order to evacuate, it is defined here as a shadow evacuation.

0 to 10 Public: This population group would evacuate as a distribution with some residents leaving promptly and others leaving later. For this analysis, the cohort is modeled as a single group.

10 to 20 Shadow: A shadow evacuation may be expected in the area beyond the EPZ. The distribution of the shadow evacuation would likely include a larger percentage of the public near the boundary of the EPZ and the percent would decrease proportional to the distance away from the EPZ. For this analysis, it is assumed that 30 percent of the general public from the 10 to 20 mile area shadow evacuate. This cohort will begin evacuating as they observe EPZ evacuees traveling through the area.

0 to 10 Special Facilities: This is a small but unique population group within this EPZ. There is no delay to shelter because these residents are assumed to be in a robust facility when the accident begins. Specialized vehicles to evacuate these facilities take time to mobilize.

0 to 10 Tail: The tail represents the last 10% of the EPZ population who typically take a longer time to begin to evacuate.

10 to 30 Public: This very large population group would evacuate as a distribution with some residents leaving promptly and others leaving later. For this analysis, the cohort is modeled as a single group.

30 to 40 Shadow: A shadow evacuation may be expected in the area beyond the evacuation area. For this analysis, it is assumed that 30 percent of the general public from the 30 to 40 mile area shadow evacuate. This cohort will begin evacuating as they observe evacuees traveling through the area.

10 to 30 Tail: The tail represents the last 10% of the population of this area who typically take a longer time to begin to evacuate.

30 to 40 SIP: For this sequence, it is assumed that 80% of the public remaining after the shadow evacuation comply with an SIP order.

Non-evacuating Public: A portion of the public does not follow protective action orders. It is assumed that 0.5% of the general public within the 0 to 40 mile area refuse to evacuate.

Table 33: WinMACCS response parameters for PAGs exceeded beyond the EPZ

Cohort		Population Fraction	**Delay to Shelter DLTSHL (hr)	Delay to Evacuation DLTEVA (hr)	Depart	DURBEG (hr)	DURMID (hr)	ESPEED (early) (mph)	ESPEED (mid) (mph)
0 to 10 Schools	1	Same as SOARCA baseline	0.25	0.75	1	0.25	2	20	10
0 to 10 Shadow	2	20% of 0-10 mile population	0.5	0.5	1	0.25	2	20	10
0 to 10 Public	3	Same as SOARCA	1.00	1.00	2	0.25	6	5	3
10 to 20 Shadow	4	30% of the public in this area	2	2	4	0.25	6	5	3
0 to 10 Special Facilities	5	Same as SOARCA baseline	0	5	5	.5	1	3	10
0 to 10 Evacuation Tail	6	10% of EPZ population	1.00	5	6	.5	1	3	10
10 to 30 Public		SECPOP then subtract other cohorts	24	4	28	2	18	2	1
30 to 40 Shadow		20% of public in this area	24	4	28	1	8	5	5
10 to 30 Special Facilities		Use same pop fractions SOARCA sensitivity #2	0[illegible]	30	30	1	20	10	2
30 to 40 Shadow		10% of public in this area	28	4	32	1	15	1	1
10 to 30 Evacuation Tail		10% of 10 to 40 mile area population	36	4	40	18	2	1	10
30 to 40 Shelter in Place		80% of public remaining after the Shadow in this area comply	24	All of Emergency Phase	NA	NA	NA	NA	NA
0-40 Non-Evac		0.5%	NA	NA	NA	NA	NA	NA	NA

**Delay to shelter is from the start of the accident. Set O-alarm accordingly.

WinMACCS response parameters for early release sequences where PAGs are exceeded beyond the EPZ.

(This section TBD)

7.3.4. Long-term Phase Modeling

This phase is the period following the emergency phase and continues for 50 years. Exposure during this phase includes internal doses from inhalation of resuspended radionuclides and

ingestion of food/water with trace contaminants. However, exposure is mainly from external doses from trace contaminants that remain after the land is decontaminated, or in lightly contaminated areas where people never had to evacuate or relocate. Depending on the relevant protective action guides and the level of radiation, food/water below a certain limit could be considered adequately safe for ingestion, and lightly contaminated areas could be considered habitable.

<u>Protective Actions against Contaminated Land</u>

Three protective actions are modeled to occur for contaminated land during the long-term phase: decontamination, interdiction, and condemnation. In modeling, MACCS2 will consider these actions when land is not habitable. The actual determination of when land will be decontaminated after a nuclear accident will be based on a long-term cleanup strategy, which is currently being developed by the NRC, EPA, and other federal agencies. Furthermore, a cleanup standard with a set dose level may purposefully not be predetermined, due to the accident-specific and regional-dependent characteristics of such an event.

In MACCS2, the models will consider land decontamination if decontamination can make the land habitable and if the value of the land is greater than the cost to decontaminate. If the level of contamination is too high, or if the cost of decontamination were assessed to be higher than the land value, the land is assumed to be condemned. Because both the land values and the level of possible decontamination affect decisions on whether contaminated areas can be restored to habitability, they also affect predicted long-term doses, health effects, and economic costs.
During this decontamination period, the land is interdicted (e.g. the land is temporarily uninhabitable), and furthermore, potentially interdicted for an additional period if needed to make the land habitable. If land cannot be restored to habitability in 30 years, the land is modeled as condemned and residents will not return during the long-term phase. The length of the late phase (i.e. 50 years) has been chosen to provide a reasonable time period for calculating consequences from exposure of the average person. For some small children the time period is too short, but for other older residents it is too long.

<u>Habitability Criterion:</u>

Site-specific values are used to determine long-term habitability. Most States adhere to EPA guidelines that allow a dose of 2 rem in the first year and 500 mrem each year thereafter. However, Pennsylvania has a more strict habitability criterion of 0.5 rem beginning in the first year and each following year), which is the value that is used for this study. This value was used in the Peach Bottom analysis even though areas outside of Pennsylvania could also potentially be affected.

<u>Land Values:</u>

Land values are divided into two types in MACCS2: farmland and non-farmland. In NUREG-1935, these values also are scaled from NUREG-1150 values using the Consumer Price Index (CPI) as the basis for price escalation. A scaling factor of 1.250 was used to account for inflation between the years 2002 and 2011. Land values affect the decision to decontaminate, interdict, or condemn land.

<u>Decontamination Factors:</u>

MACCS2 considers two levels of decontamination: a decontamination factor of 3 and 15. This is consistent with NUREG-1935, as well as done previously in NUREG-1150. The time periods associated with these decontamination levels depend on the size of the contaminated area. However, this study models both of these decontamination factors to take one year to achieve, which is the current maximum time period allowed in MACCS2. This study uses the values in NUREG-1935 for the cost of decontamination.

7.4. Rationale for Reporting Metrics of Land Contamination

A major consideration to reporting land contamination is: At what level is land considered contaminated? At the current time, neither the EPA nor the NRC has a general standard for long-term, unrestricted public use of land. That being said, the EPA and the NRC are currently considering such a long-term cleanup strategy. In addition NRC, EPA, and others do have standards related to this question, as seen below:

Table 34: Radiation Dose Limits to Members of the Public

Radiation Dose Limits to Members of the Public				
Annual Dose (mrem)	Standard/Regulation/Guide	Timeframe	Applicability	Notes
???	EPA/NRC long-term cleanup strategy	long-term	severe accidents	currently in rulemaking
25	10 CFR 20 subpart E	long-term	decommissioning	unrestricted use of land
100	10 CFR 20 subpart D	operation	licensed operation	
100	ICRP recommendation	long-term	severe accidents	with "optimization"
500	Pennsylvania Code Title 25 § 219.51	after emergency	severe accidents	
2000	EPA Protective Action Guide	after emergency	severe accidents	for first year (500 mrem thereafter)

<u>10 CFR 20 subpart E</u>: This section relates to decommissioning of nuclear facilities. At 25mrem/yr plus ALARA, land that was previously used for nuclear facilities can be used for unrestricted use by the public. In addition, this section of the law, under certain circumstances,

could allow a relaxation of the standard to 100 mrem/yr if there are institutional controls placed on the land (e.g. no fishing).

While CFR 20 subpart E is specifically for "Radiological Criteria for License Termination", this section creates a precedent for what the NRC may consider safe for long-term public use of land.

<u>10 CFR 20 subpart D</u>: This section relates to restrictions on the licensee to limit the annual dose to the members of the public to 100mrem from licensed operation. This section of the law allows for certain exceptions (e.g. a visitor that demonstrates a need and has NRC permission can have a 500mrem limit), as well as stricter limits for certain release paths as determined by ALARA principles. However, similar to subpart E, subpart D is specific to licensed operation, and therefore does not apply to beyond design basis accidents, which exist outside of the licensing basis.

<u>ICRP recommendation</u>: The ICRP standard for long-term, unrestricted public use of land is 100mrem/yr + optimization. "Optimization" allows the jurisdiction of the local area to adjust the ultimate dose level criterion (upwards or downwards), in order to consider the needs of their locale in response to a specific reactor accident. For instance, the local jurisdiction may wish to use a lower dose standard for a playground.

<u>Pennsylvania dose limits for the public</u>: Similar to the EPA PAGs, Pennsylvania has a state-specific value for relocation of 500 mrem in the first year. This can be found in the Pennsylvania Code, Title 25 Chapter 219 – § 219.51 - Dose Limits for Individual Members of the Public.

<u>Environmental Protective Agency intermediate phase protective action guides</u>: The EPA[14] states that for nuclear incidents, relocation is warranted when the projected sum of the dose equivalent from external gamma radiation and the committed effective dose equivalent from inhalation of resuspended radionuclides exceeds 2 rem in the first year.
This value was established based on an objective to limit the total dose to 5 rem in 50 years. 50 years is the usual length of time given for the average remaining lifetime of exposed individuals, and 5 rem is the corresponding 50-year dose of the ICRP recommended 100 mrem/yr upper bound for all sources combined for chronic exposure.
The 5 rem lifetime dose corresponds to an upper bound of 2 rem from exposure during the first year and 0.5 rem from exposure during the second year when considering radioactive decay and weathering from a nuclear reactor accident. A spent fuel pool accident is expected to have a significantly longer average radioactive decay, which would affect this calculation. However, the EPA [14] has stated that:

> Although these Protective Action Guides (PAGs) were developed based on expected releases of radioactive materials characteristic of reactor incidents, they may be applied to any type of incident that can result in long-term exposure of the public to deposited radioactivity.

Another consideration in reporting land contamination is: "At what point in time is the land to be evaluated?" As time passes, radionuclides will transport and decay. Also, depending on society's capability and effort, decontamination will also affect the amount of contamination.
In addition, different dose standards are applicable to different points in time. A dose level on a piece of property could be low enough for people to return, or habitable in the short-term with certain institutional controls; however, a similar dose level may not be considered acceptable for long-term habitation, and institutional controls for the land may be different.

Because our study is site-specific, and our SFP is in Pennsylvania, this research project reports the area of land that exceeds the dose limits in the Pennsylvania dose limits for habitation (500mrem), for the first year after the accident.

In addition to using the 500mrem annual dose to evaluate land contamination and for select scenario(s), the project also currently plans to report the area of land that exceeds a range of dose levels:

- 25mrem (10 CFR 20 subpart E)
- 100mrem (10 CFR 20 subpart D)
- 100mrem + optimization (ICRP recommendation)
- 500mrem (Pennsylvania dose limits to the public)
- 2rem (EPA intermediate phase Protective Action Guide criterion for relocation)

Modeling land contamination in the immediate year after the accident is more straightforward because as stated before, neither the EPA nor the NRC currently has a general standard for long-term, unrestricted public use of land. Ultimately, the strategy may be some form of optimization for local jurisdictions as opposed to a set dose value. In addition, modeling potential land contamination in the immediate year after the severe accident carries less uncertainty than long-term modeling. This is because economic factors such as the cost of decontamination and property values, as well as ecological factors such as groundwater transport and plant-soil interactions, will play a role in long-term contamination.

7.5. Offsite Consequence Binning

In order to incorporate large numbers of sequences, and to avoid conveying a more significant level of accuracy for any particular sequence, the sequences were binned by their Cs-137 and I-131 release activities (see Table 35).

The first criterion used to bin the sequences was Cs-137 release, because Cs-137 is the most significant contributor to consequence. I-131 was also chosen to bin the sequences, because I-131 is a good indicator for short-lived radionuclides that may be released from recently discharged spent nuclear fuel. These short-lived radionuclides are the largest potential contributors to early doses and acute health effects. The tally into each of these bins can be seen in Table 36.

Table 35: Release category types

Release Category Binning		Cesium-137 Release Activity (MCi)		
		0 to 0.5	0.5 to 5	5+
Iodine-131 Release Activity (MCi)	0 to 0.25	RC11	RC21	RC31
	0.25 to 0.55	RC12	RC22	RC32
	0.55+	RC13	RC23	RC33

Table 36: Release category tally

Release Category	RC11	RC12	RC13	RC21	RC22	RC23	RC31	RC32	RC33	Total
Sequence Tally	6	6	0	2	0	3	0	0	3	20

A representative sequence was chosen from each bin. The decision of which sequence to represent the bin considered a number of different factors, including the release frequency, the relative Cs-137 and I-131 release for the bin, the start of release, loading configuration, and the availability of the source term (certain accident progression calculations were still ongoing at the time). In addition, due to the significant differences in Release Category 33 relative to the other bins, all 3 of these sequences were analyzed.

In order to incorporate large numbers of sequences, and to avoid conveying a more significant level of accuracy for any particular sequence, the sequences were binned by their Cs-137 and I-131 release activities.

The different sequences were then aggregated into their respective scenarios, those being high and low density loading, and with and without deployed 50.54(hh)(2) equipment. For simplicity, sequences with no release are not included, as they do not have offsite consequences. Sequences with no release include all OCP4 and OCP5 sequences, and all the sequences with no hole in the SFP.

For all sequences, deployment of 50.54(hh)(2) prevents release of radioactive material, except for a moderate size hole during OCP1, which is when newly discharged fuel is first loaded from the reactor. Without successful deployment of 50.54(hh)(2), the predicted scenario-specific release frequency is 1E-7.

Table 37: Listing of specific sequences

High Density (1x4) Fuel Loading								Low Density Fuel Loading							
Unsuccessful mitigation				Deployed 50.54(hh)(2)				Unsuccessful mitigation				Deployed 50.54(hh)(2)			
Sequence		Release Frequency (/yr)	Release Category	Sequence		Release Frequency (/yr)	Release Category	Sequence		Release Frequency (/yr)	Release Category	Sequence		Release Frequency (/yr)	Release Category
OCP1	small hole	6E-09	RC12	OCP1	mod hole	6E-09	RC12	OCP1	small hole	6E-09	RC12	OCP1	mod hole	6E-09	RC12
	mod hole	6E-09	RC21						mod hole	6E-09	RC11				
OCP2	small hole	2E-08	RC33 (2.4)					OCP2	small hole	2E-08	RC12				
	mod hole	2E-08	RC21						mod hole	2E-08	RC11				
OCP3	small hole	4E-08	RC33 (3.4)					OCP3	small hole	4E-08	RC11				
	mod hole	4E-08	RC11						mod hole	4E-08	RC11				
	Total	1E-07			Total	6E-09			Total	1E-07			Total	6E-09	

7.6. MACCS2 Results

Note: These are preliminary results calculated with a 10-mile evacuation. While a 10 mile evacuation is applicable to some of the smaller releases, they are not realistic for Release Category 33. Therefore, while longer-term consequences (such as uninhabitable land) will not be affected, latent cancer fatalities are subject to change. That being said, it is in project staff opinion that latent cancer fatalities is also largely a longer-term consequence, and therefore these results should not change significantly.

Table 38: Overall consequence results

Scenario-Specific Consequences					
SFP Fuel Loading	Density (# of assemblies)	High Density (OCP1 = 2859) (OCP2 through OCP5 = 3055)		Low Density (OCP1 = 856) (OCP2 through OCP5 = 852)	
	Pattern	1x4		Mixture of 1x4 and checkerboard	
Operator Actions		Unsuccessful	Deployed 50.54(hh)(2)	Unsuccessful	Deployed 50.54(hh)(2)
Type of Consequence					
Release Frequency (/yr)		1.18E-07	6.14E-09	1.18E-07	6.14E-09
Conditional* Probability of Release		0.69%	0.036%	0.69%	0.036%
Earliest Time to Release (hr)		8.7	15	8.7	13
Hydrogen Combustion Event		Possible	Not Predicted	Not Predicted	Not Predicted
Cumulative Cs-137 Release @ 72 hours for Limiting Case (MCi)		24	0.26	0.33	0.19
		Conditional** Consequences			
Individual Latent Cancer Fatality Risk*** for 0-10 Miles		4.3E-04	3.3E-04	2.0E-04	3.3E-04
Individual Early Fatality Risk		0	0	0	0
		Release Frequency-Factored Consequences (/yr)			
Individual Latent Cancer Fatality Risk*** for 0-10 Miles (/yr)		5.1E-11	2.0E-12	2.4E-11	2.0E-12
Individual Early Fatality Risk (/yr)		0	0	0	0
Uninhabitable Land**** (Hectares/yr)		2.9E-01	3.6E-04	4.9E-03	3.6E-04
Displaced Individuals**** (Persons/yr)		4.9E-01	7.1E-04	9.3E-03	7.1E-04

* Given Specified Seismic-Event Occurs

** Given Atmospheric Release Occurs

*** Linear-No Threshold, Weather-Averaged, Release Frequency-Weighted, and Population-Weighted

**** 1st Year Post-Accident, Weather-Averaged, Release Frequency-Weighted; Calculation uses a dose limit of 500mrem/yr, according to Pennsylvania Code, Title 25 § 219.51 - Dose Limits for Individual Members of the Public

Table 39: Dose Response Model Results

Dose Response Model Comparison (Scenario Specific, Weather-Averaged, Release Frequency-Weighted)					
Dose Response Model	Fuel Loading Pattern	High Density (1x4)		Low Density	
	Operator Actions	Unsuccessful	Deployed 50.54(hh)(2)	Unsuccessful	Deployed 50.54(hh)(2)
Linear, No Threshold	Individual Latent Cancer Fatality Risk** for 0-10 Miles (/yr)	5.1E-11	2.0E-12	2.4E-11	2.0E-12
	Conditional* Individual Latent Cancer Fatality Risk** for 0-10 Miles	4.3E-04	3.3E-04	2.0E-04	3.3E-04
	Latent Cancer Fatalities (/yr)	2.2E-03	1.5E-05	1.7E-04	1.5E-05
	Latent Cancer Incidence (/yr)	4.9E-03	3.5E-05	3.9E-04	3.5E-05
620mrem/yr truncation	Individual Latent Cancer Fatality Risk** for 0-10 Miles (/yr)	7.3E-15	3.1E-16	3.1E-15	3.1E-16
	Conditional* Individual Latent Cancer Fatality Risk** for 0-10 Miles	6.2E-08	5.1E-08	2.7E-08	5.1E-08
	Latent Cancer Fatalities (/yr)	1.7E-04	8.4E-07	1.0E-05	8.4E-07
	Latent Cancer Incidence (/yr)	4.1E-04	2.0E-06	2.4E-05	2.0E-06
5rem/yr or 10rem lifetime truncation	Individual Latent Cancer Fatality Risk** for 0-10 Miles (/yr)	1.6E-14	1.1E-16	6.6E-16	1.1E-16
	Conditional* Individual Latent Cancer Fatality Risk** for 0-10 Miles	1.3E-07	1.8E-08	5.6E-09	1.8E-08
	Latent Cancer Fatalities (/yr)	3.4E-05	2.1E-07	2.4E-06	2.1E-07
	Latent Cancer Incidence (/yr)	7.7E-05	4.9E-07	5.5E-06	4.9E-07

* Given Atmospheric Release Occurs

** Population-Weighted

7.7. Tabulation of Delta Consequences

Table 40: Consequence comparison # 1: Low/High density (1x4) loading

Benefit of Low Density vs. High Density (1x4) Fuel Loading (Scenario Specific, Weather-Averaged, Release Frequency-Weighted, Unsuccessful Deployment of 50.54(hh)(2))		
Type of Consequence	Release Frequency-Factored Consequences (/yr)	Conditional** Consequences
	Reduction Factor (dimensionless)	
Release Frequency	1.0	-
Individual Latent Cancer Fatality Risk*** for 0-10 Miles	2.1	2.1
Latent Cancer Incidence	13	13
Latent Cancer Fatalities	13	13
Uninhabitable Land (Hectares)	58	58
Displaced Individuals (Persons)	53	53

** Given Atmospheric Release Occurs

*** Linear-No Threshold, Population-Weighted

Table 41: Consequence Comparison #2: Low/High Density (uniform) loading

Benefit of Low Density vs. High Density (uniform) Fuel Loading (Scenario Specific, Weather-Averaged, Release Frequency-Weighted, Unsuccessful Deployment of 50.54(hh)(2))		
Type of Consequence	Release Frequency-Factored Consequences (/yr)	Conditional* Consequences
	Reduction Factor (dimensionless)	
Release Frequency	1.0	-
Individual Latent Cancer Fatality Risk** for 0-10 Miles	3.3	3.3
Latent Cancer Incidence	18	18
Latent Cancer Fatalities	18	18
Uninhabitable Land (Hectares)	81	81
Displaced Individuals (Persons)	73	73

** Given Atmospheric Release Occurs

*** Linear-No Threshold, Population-Weighted

Table 42: Consequence comparison # 3: High density (1x4 and uniform) loading

Benefit of High Density (1x4) vs. High Density (uniform) Fuel Loading (Scenario Specific, Weather-Averaged, Release Frequency-Weighted, Unsuccessful Deployment of 50.54(hh)(2))		
Type of Consequence	Release Frequency-Factored Consequences (/yr)	Conditional* Consequences
	Reduction Factor (dimensionless)	

Release Frequency	1.0	-
Individual Latent Cancer Fatality Risk*** for 0-10 Miles	1.56	1.56
Latent Cancer Incidence	1.40	1.40
Latent Cancer Fatalities	1.40	1.40
Uninhabitable Land (Hectares)	1.38	1.38
Displaced Individuals (Persons)	1.38	1.38

** Given Atmospheric Release Occurs

*** Linear-No Threshold, Population-Weighted

Table 43: Consequence comparison # 4: Successful/Unsuccessful deployment of 50.54(hh)(2)

Benefit of Successful Deployment of 50.54(hh)(2) Equipment (Scenario Specific, Weather-Averaged, Release Frequency-Weighted)		
Fuel Loading Density	High Density (1x4)	Low Density
Release Frequency (/yr)	19	19
	Release Frequency-Factored Consequences (/yr)	
Type of Consequence	Reduction Factor (dimensionless)	
Individual Latent Cancer Fatality Risk** for 0-10 Miles (/yr)	25	12
Latent Cancer Incidence (/yr)	142	11
Latent Cancer Fatalities (/yr)	142	11
Uninhabitable Land (Hectares/yr)	792	14
Displaced Individuals (Persons/yr)	698	13
	Conditional* Consequences	
Type of Consequence	Reduction Factor (dimensionless)	
Individual Latent Cancer Fatality Risk*** for 0-10 Miles	1.3	0.60
Latent Cancer Incidence	7.4	0.58
Latent Cancer Fatalities	7.4	0.58
Uninhabitable Land (Hectares)	41	0.71
Displaced Individuals (Persons)	36	0.68

* Given Atmospheric Release Occurs

** Linear-No Threshold, Population-Weighted

References:

[1] Chanin, D. and M.L. Young, *NUREG/CR-6613, SAND97-0594: Code Manual forMACCS2 User's Guide*. 1997, Sandia National Laboratories: Albuquerque, NM.

[2] NUREG-1150, V., *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*. 1990, U.S. Nuclear Regulatory Commission: Washington, D.C.

[3] Laur, M.N., "Meeting with Sandia National Laboratories and an Expert Panel on MELCOR/MACCS Codes in Support of the State of the Art Reactor Consequence Analysis Project," Memo to J.T. Yerokun, Agency Document Access and Management System Accession Number ML062500078, U.S. Nuclear Regulatory Commission, Washington, D.C., September, 2006.

[4] NUREG-0654, *NUREG-0654/FEMA-REP-1, Rev. 1: Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*. 1979, U.S. Nuclear Regulatory Commission: Washington, DC.

[5] NUREG/CR-6864, Vol. 1: "Identification and Analysis of Factors Affecting Emergency Evacuations: Main Report." January 2005

[6] McFadden, K.L., N.E. Bixler, and R.O. Gauntt, *MELMACCS System Documentation (MELCOR to MACCS2 interface definition)*. 2005, Sandia National Laboratories: Albuquerque, NM.

[7] Bixler, N.E., E. Clause, C. W. Morrow, and J. A. Mitchell, *Evaluation of Distributions Representing Important Non-Site Specific Parameters in Off-Site Consequence Analyses*, to be published as a NUREG Report, Sandia National Laboratories, Albuquerque, NM, 2010.

[8] Bixler, N.E., et al., *NUREG/ER-6525, Rev. 1, SAND2003-1648P: SECPOP2000: SectorPopulation, Land Fraction, and Economic Estimation Program*. 2003, Sandia National Laboratories: Albuquerque, NM.

[9] NUREG/CR-6953, "Review of NUREG-0654, Supplement 3, 'Criteria for Protective Action Recommendations for Severe Accidents' - Focus Group and Telephone Survey," U.S. Nuclear Regulatory Commission, Washington, D.C., 2008

[10] EPA (2202) Federal Guidance Report No. 13 CD Supplement, E.-C.-.-., Rev. 1. 2002, Prepared by Oak Ridge National Laboratory, Oak Ridge, TN for Office of Air and Radiation, U. S. Environmental Protection Agency: Washington, DC.

[11] EPA (1994). *Estimating Radiogenic Cancer Risk*. EPA 402-R-93-076 (U.S. Environmental Protection Agency, Washington, D.C.)

[12] Eckerman, K. F., Review of Risk Coefficients for SOARCA Project, , memo, Oak Ridge National Laboratory, February 2009.

[13] NAS, *Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V*. 1990, National Academy of Sciences, National Research Council, National Academy Press: Washington, DC.

[14]Environmental Protection Agency (EPA). EPA-400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents." Washington D.C.: EPA. May 1992. (EPA, 1992). http://www.epa.gov/radiation/docs/er/400-r-92-001.pdf

Nuclear Regulatory Commission, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." NUREG

0654/FEMA REP 1, Rev. 1, Nuclear Regulatory Commission: Washington DC. November 1980.

Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-6864. "Identification and Analysis of Factors Affecting Emergency Evacuations." Washington D.C.: NRC, January 2005. (NRC, 2005).

Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-6953, Vol. 2. "Review of NUREG-0654, Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents, Focus Groups and Telephone Survey." Washington D.C.: NRC. October 2008. (NRC, 2008).

Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-7110, Volume 1. " State of the Art Reactor Consequence Analyses (SOARCA) Project: Peach Bottom Integrated Analysis." Washington D.C. NRC. January 2012. (NRC, 2012a).

Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-7110, Volume 2. " State of the Art Reactor Consequence Analyses (SOARCA) Project: Surry Integrated Analysis." Washington D.C. NRC. January 2012. (NRC, 2012b).

Wolshon, Brian, J. Jones, and F. Walton. "The Evacuation Tail and Its Effect on Evacuation Decision Making." Journal of Emergency Management. January/February 2010, Volume 8, Number 1. (Wolshon, 2010).

8. CONSIDERATION OF UNCERTAINTY

There is a desire to perform sensitivity studies to assess the effects of various uncertainties on the results presented in this report. For the time being, focus is being placed on completing the production analysis. However, this section catalogues key uncertainties in the study assumptions or computer modeling that will be investigated if resources permit.

Overall Approach:

- Regarding multi-unit events, see proposed sensitivity listed below under "Consequence Analysis"

Seismic Assessment:

Structural and Related Initial Damage State Characterization:

- Failure of the pneumatic seal surrounding the fuel transfer canal gate during non-outage configurations...

Scenario Delineation and Probabilistic Treatment:

- There are numerous assumptions made when assigning the specific layout of spent fuel during the various OCPs, particularly for the high-density loading situation. Most of these assumptions are driven by simplifications needed to make the MELCOR modeling tractable when considering so many different scenarios. The effects of the more important decisions will be assessed qualitatively.
- The assumptions on diagnosis will have an effect on some scenarios. For this reason, a specific scenario that represents a case that was near the threshold between success and failure (in terms of a release) will be chosen. This scenario will be re-run using alternative assumptions on diagnosis. The first case will investigate earlier diagnosis (e.g., no 30 minute delay) while the second case will investigate later diagnosis (e.g., 1 hour delay).
- The assumptions used limit the mode (makeup versus spray) of the 50.54(hh) equipment to whichever mode is initially deployed. For a scenario of interest, a more detailed timeline will be developed using the decision process guidance from Figure 2-1 of NEI-06-12, Revision 2, if the results of any of the cases analyzed suggest that a switch in mode might be prompted.
- If feasible with the given information, a simplified analysis may be performed to look at the effect of a single assembly in the lifted position at the time of the event.
- For OPC #1/#2, with the "moderate" leakage rate, different mitigation deployment mode assumptions could lead to the deployment of sprays rather than makeup. A sensitivity study may be run for one of these cases to show the effect.

Accident Progression Analysis:

- Peach Bottom actually strives to maintain a 1x9 configuration rather than the far more typical 1x4. A sensitivity study will be performed to quantify the additional benefit (for a selected sequence) from the more favorable arrangement.
- The number of filed rack locations in the high-density loading case equals the pool capacity minus a full core offload capability. It is believed that 60 rack locations are

reserved for storage of other hardware (e.g., guide tubes), but nonetheless, 3055 stored assemblies is used. The reduction of the assumed high-density pool inventory by 60 colder assemblies will be considered by a simple assessment of the impact of these assemblies on the projected source term (e.g., ratioing of source term considering the reduction of these assemblies).

- There is some subjectivity in the modeling needed to take a 3-D Cartesian spent fuel pool rack arrangement and translate it to a 2-D cylindrical computer model (MELCOR). An example of this has to do with the prescription of heat transfer quantities (view factors, surface areas) between Rings 5, 6, and 7 in the low-density pool configuration. Modeling here took a moderate approach in establishing a nearly 1x2 ratio of Ring 5 to Ring 6 and assigning an interface that increases the surface area between Rings 6 and Ring 7. Whether these assumptions are conservative or non-conservative is highly situation (and heat transfer mode) specific. While the choices made are viewed as reasonable, the effect of an alternative assumption will be investigated.
- The current MELCOR model does not account for the effects of MCCI (should it occur) on the source term. This uncertainty will be investigated.
- Peach Bottom had a small power uprate in 2002, which is after the ORIGEN analysis used for this study was completed. In addition, as operating cycles have grown longer, the trend has been to take the fuel to higher burnups. Subsequent to the establishment of the decay heats and inventories used in this project, the licensee provided information regarding the assemblies offloaded between 2002-2011. Additional ORIGEN calculations were performed which suggest that the current study generally underestimates the long-lived radionuclide inventory while over-estimating the short-lived radionuclide inventory. A sensitivity study will be run for this new set of inventories.
- Since the study does not account for many multi-unit / concurrent accident events, a sensitivity will be run whereby a hydrogen deflagration occurs from hydrogen generated from a concurrent reactor event for a SFP scenario where such an event would not otherwise be predicted (e.g., a OCP #3 unmitigated, moderate leak case). The timing of the deflagration, and the amount of hydrogen participating, will be based on a SOARCA STSBO scenario. This calculation will be run twice, once for a high-density loading case and once for the corresponding low-density case.

Consequence Analysis:

- A set of MACCS2 calculation will be run for a low-density and high-density OCP #3 case that goes to release, which combines the SFP release with the SOARCA Peach Bottom STSBO release.

9. OTHER ISSUES

9.1. Probabilistic Risk Assessment versus Consequence Assessment

This study is a limited-scope consequence assessment that utilizes probabilistic insights. By looking at these probabilistic aspects, the results can be placed in better context, via the limited treatment of relative likelihood. While these elements provide some of the benefits of an actual risk assessment, there are several elements of a risk assessment that are specifically not performed. These are:

- Failure modes and effects analysis (except for SSCs specifically discussed in Section 4 of this report)
- Data analysis and component reliability
- Effects of dependencies
- Human reliability analysis
- System fault tree and sequence event tree development and quantification

Even so, this study does attempt to bring probabilistic insights to bear. In terms of inputs to the study these include:

- Risk information from past studies for selecting the scenarios studied
- Initiating event likelihood
- Initiating event timing effects (e.g., the relative likelihood of having an event during the various operating cycle phases and the likely configurations incurred)
- Relative likelihood of damage state characteristics and conditional probabilities associated with offsite consequence analysis (e.g., meteorological sampling in MACCS2 analysis)

In terms of assessing the results, the consideration of probabilistic insights uses the above inputs (and simple algebraic combination) to quantify different figures of merit, in order to place context on the results.

The inclusion of probabilistic aspects within the current scoping study allows for consideration of some aspects of likelihood, but will not support definitive statements on risk. To elaborate, consider the results of this study to be distinct data points along an unknown capacity function. Now consider regulatory requirements/goals (most notably the Safety Goals) to be a continuous failure functions. By comparing the capacity data points to the failure function, we can gain confidence about whether the data generally lies in this success regime or the failure regime. We can take this one step further and use past studies to compare our data points to other capacity functions. Using this approach we can draw supportable, but not definitive, conclusions.

9.2. Multi-Unit Considerations

Observations regarding a concurrent reactor event:

There are five broad interplays possible between the spent fuel pool and the reactor:

1. An initiating event which directly affects both the reactor and the spent fuel pool,

2. A reactor accident which prevents accessibility to the spent fuel pool for a prolonged period of time, leading to a spent fuel pool accident,

3. A reactor accident that includes ex-containment energetic events (most notably a hydrogen deflagration or detonation) which presents a hazard to the spent fuel pool (e.g., by causing debris to fall in to the pool),

4. A spent fuel pool accident which prevents accessibility to key reactor systems and components for a prolonged period of time, leading to a reactor accident, and

5. A spent fuel pool accident which creates a hazard for equipment used to cool the reactor (e.g., the flooding of low elevations of the reactor building due to a leak in the pool or excessive condensation from continuous boiling of spent fuel pool water).

For each of these, large seismic events and severe weather loss-of-offsite power events are logically the most relevant initiators, as they are the type of initiators that are most likely to (a) initiate an accident at the reactor and/or spent fuel pool and simultaneously (b) further hamper accessibility to key areas, key systems and components, and key resources. To the extent practicable, this study has attempted to qualitatively account for some of these effects. For example, when the reactor and SFP are hydraulically connected (during refueling), the decay heat and water volumes from both sources are considered. However, explicitly modeling multi-unit effects was not a focus of this study, due to the existing limitations with the available computational tools. The site Level 3 PRA described in SECY-11-0089 will attempt to more rigorously address these effects.

Observations regarding a multi-unit event

Along with the possibility of a concurrent spent fuel pool and reactor accident, there is the possibility for a concurrent accident at the spent fuel pool of one unit with an accident at the spent fuel pool or reactor of the other unit. Again, a large seismic event or a severe weather LOOP are the events that are most likely to lead to a multi-unit event. In general, if accidents at both spent fuel pools proceed in similar manners and similar timeframes, and both pools have similar inventories of spent fuel, then the resulting source term from a dual-unit event would be roughly twice the single-unit source term. In reality, this type of perfect symmetry is highly unlikely because the two (or more) pools are very unlikely to have the same total pool heat load or peak assembly heat load (recall that for multi-unit sites, the reactors did not usually start operation at the same time and outages are intentionally staggered). Even if such a symmetry existed, the offsite consequences would not follow a linear scaling, due to a number of non-linearities associated with that portion of the analysis. Again, capturing such effects was not a focus of this study, and future work (SECY-11-0089 Level 3 PRA) will attempt to more rigorously treat these effects.

9.3. Seismic Risk for Dry Cask Storage

A logical question to ask is what effect the studied (challenging but very low frequency of occurrence) seismic event would have on the fuel in dry cask storage at Peach Bottom. For the reasons outlined earlier, this study does not attempt to quantitatively address that question. Even so, that issue can still be addressed to the point of providing perspective.
For low to moderate seismicity sites (such as Peach Bottom), weight, cask geometry and friction protect against sliding or tip-over. For reference, NUREG-1864 (a pilot dry cask storage probabilistic risk assessment performed by the NRC) concludes that for the conditions and type of cask studied, the PGA necessary to postulate damage to the cask (from tip-over) was 1.35g, which is well beyond the 0.71g SFPSS considers. The cask considered in the NUREG-1864 study (Holtec HI-STORM 100) has a slightly higher aspect ratio (width to height ratio) than the cask used at Peach Bottom (TN-68). Comprehensive parametric evaluations are reported in NUREG/CR-6865 for a cask similar to the HI-STORM 100 and for earthquakes with varying ground shaking characteristics including characteristics similar to those expected for the Central and Eastern United States (CEUS). These evaluations show that tip-over of the cask studied would not start for an earthquake about 2.3 times the earthquake considered in the current study. A margin of 230% is considered sufficient to exclude a dry cask tip-over assumption in the current study even taking into account the differences in the aspect ratios of the casks.

As a real world example, the August 2011 earthquake at the North Anna site (which utilizes TN-32 casks), caused a cask to slide up to 4.5 inches, with no tip-over. Similarly, the Fukushima Daiichi site utilized on-site dry cask storage, with no significant damage occurring from either the earthquake or the tsunami. It is also noted that casks have been evaluated for hypothetical drops and tip-over with lateral inertia loads on the casks of the order of 30g, which are at least an order of magnitude less than lateral inertial loads from earthquakes.

Even postulating a seismic event sufficiently large for a tip-over, assuming mechanical damage to the fuel from the tip-over, and assuming a cask breach from the tip-over, the resulting radioactive release would be a limited "gap" release at ground level with no energetic heat content. Such a release, even considering releases from multiple casks, would involve far less radioactive material than is associated with a reactor or spent fuel pool fuel melt scenario. To have a sizable release from a cask in the seismic scenario, it would be necessary to have breaching of the cask and then heating of the fuel to temperatures of the order of 900 C. These temperatures are much higher than those that heating by decay heat and low-temperature cladding oxidation (particularly in the oxygen-limited configuration of a cask breach) could achieve.

9.4. Other Considerations Related to Expedited Fuel Movement

As stated up front, this study focuses on offsite consequences associated with a release from the spent fuel pool, for the specified earthquake. The set of information provided in this report helps to frame specific aspects of a much larger issue, that of whether expedited movement of spent fuel results in a sufficient benefit to warrant its requirement. As mentioned at the start of the report, there are many other considerations that play in to this larger decision. These include, but are not limited to, the following issues:

- Current regulatory requirements under 10 CFR Part 72 limit (from a practicality perspective) the ability to transfer fuel with less than roughly 3 to 5 years of cooling
- Discharging large amounts of fuel (and thus greatly increasing the amount of fuel contained in the ISFSI) might require a Part 72 rulemaking effort (e.g., to accommodate increases in the design-basis accident site area boundary allowable dose limits) and potentially increase the risk associated with the ISFSI
- Expedited discharging of fuel from the spent fuel pool to dry storage increases the frequency of postulated cask drops, which in turn increases the risk of causing damage to the pool that could lead to a radioactive release
- Expedited discharging of fuel increases occupational doses for workers involved with the management and transfer of the spent fuel
- Earlier movement of fuel increases the probability that fuel will have to be re-packaged later for shipment to the eventual long-term repository or interim storage site

Issues related to design-basis accidents and risk posed by dry cask storage have received, and continue to receive, attention. Issues related to the existing dry cask storage infrastructure, worker dose, and economics are discussed in [NAC, 2011] and/or [EPRI, 2010]. Section 1.6 provides more information on each of these studies.

9.5. Inadvertent Criticality

Inadvertent criticality events (ICEs) may be possible for specific combinations of conditions (e.g., during reflood of a drained pool for a region of the pool storing higher reactivity fuel assemblies where the boron poison in the rack panels has been significantly displaced as a result of the earthquake). If such an event affected a region of the pool (as opposed to only a portion of a particular assembly), and if it occurred at a point in the accident where the fuel was only partially covered, the event could have an important impact on onsite dose rates. Design requirements and safety analyses ensure that the spent fuel stored in the pool, under normal conditions, will not result in a critical configuration. For the selected plant (and other US spent fuel pools), this is ensured through a combination of assembly spacing and neutron poison material (e.g., Boraflex). If a seismic event did cause reconfiguration of the fissionable material via either (i) direct movement of the fuel, (ii) direct degradation of the poison material, or (iii) indirect affect on either the fuel or poison material due to high temperatures associated with an induced accident, there are several 'advantageous' considerations that should to be kept in mind. These are:

1. the reactivity of fresh BWR fuel is suppressed due to the high content of burnable absorbers,

2. the majority of the fuel in the spent fuel pool has low net reactivity due to having gone through more than one operating cycle in the reactor,

3. the fuel with the highest net reactivity will likely be the once-burned assemblies which will stay in the reactor during a "shuffling" refueling outage (but would not stay in the reactor for a full core offload),

4. critical configurations of low-enriched Uranium fuel require the presence of a neutron moderator (in this case water or steam) such that an ICE would not happen in the presence of air,

5. BWR SFPs do not use borated water so the fact that the SFP may be refilled with unborated water is not a deviation from the norm, and

6. low-enriched uranium fuel assemblies (which are used in all US reactors) are geometrically designed to maximize reactivity (moderator/fuel geometry) in the reactor and so any significant alteration of the geometry of a given assembly will likely be in the direction of a less criticality-prone configuration

Having said this, there are a few counter considerations that should also be kept in mind:

1. the poison material in the rack panels contribute significantly to the net reactivity of the SFP configuration (i.e., they are a key component to ensuring sub-criticality for high reactivity assemblies)

2. the affects of large seismic events on already degraded SFP rack poison material are not easy to quantify,

3. the rack panels and poison material have a lower melting temperature than the cladding and fuel, and

4. the possibility of a criticality event cannot be summarily dismissed.

Finally, the <u>offsite</u> consequences of a criticality event (especially if it occurs when overlying water is present) are believed to be less severe from a public health and safety standpoint than the offsite consequences from a potential large release of radioactive material associated with a prolonged uncovering of the fuel in the SFP resulting from not attempting to re-flood. In consideration of all of the above, common accident management practices in the US call for the use of any available water in responding to fuel uncovery in either the reactor or SFP. The current phase of this study follows that precedent, while recommending that future work be done to better understand the specific combinations of conditions that could lead to ICEs during a large seismic event.

9.6 Use of Contiguous (Uniform) Fuel Arrangements During An Outage

The plant studied has pre-arranged the spent fuel pool such that discharged assemblies can be placed directly in to a 1x4 (actually 1x8 in the case of this plant) arrangement for the last two outages, for both operating units. This is in keeping with the relevant regulatory requirements. However, those regulatory requirements do allow for the fuel to be stored in an unfavorable configuration for some time following discharge if other considerations prevent pre-arrangement.

There is a regulatory requirement associated with the time window by which the 1x4 arrangement must be achieved, however the specific time requirement is not publicly-available information (because it could be potentially useful to an adversary). This section posits a situation where the fuel is unfavorably arranged during the outage to demonstrate the effect of this aspect on the results.

The layout of assemblies for the OCP1 and OCP2 uniform configuration is shown in Figure 95 and Figure 96. For the 1x4 configuration (see Figure 42), the effective area between Rings 1 and 2 was determined by the number of panels (i.e., 352 panels for 88 assemblies) since each

assembly in Ring 1 is completely surrounded by Ring 2 assemblies. In the uniform configuration (Figure 95), the surface areas between Rings 1 and 2, and Rings 3 and 4 were effectively reduced by about an order of magnitude assuming that all the assemblies in Rings 1 and 3 formed an approximate square. In the 1x4 pattern, the boundary area (per unit axial length) for Rings 1 and 3 were based on 4 panels per assembly. In the uniform pattern, the number of panels per assembly are estimated as 0.4 for Ring 1 ( ) and 0.3 ( ) for Ring 3. It should be noted that this is a stylized representation of a uniform configuration by limiting the areas (and thus total heat transfer) between the hot rings and the rest of the assemblies in the pool.

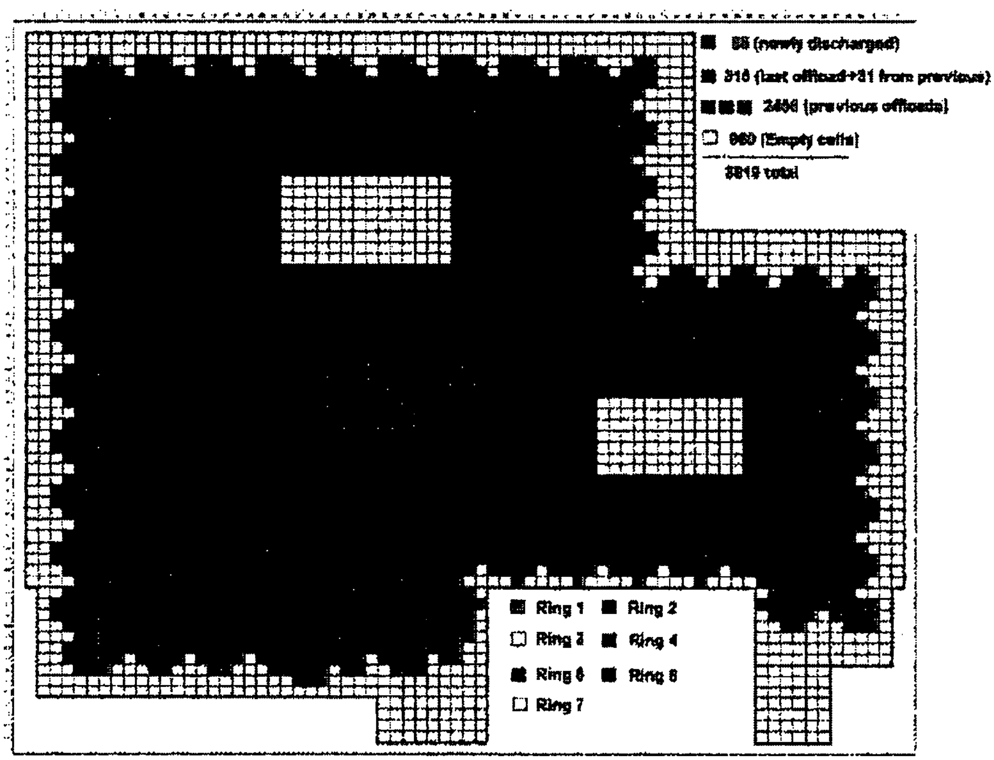

Figure 95: Layout of assemblies for OCP1 high density (uniform) model

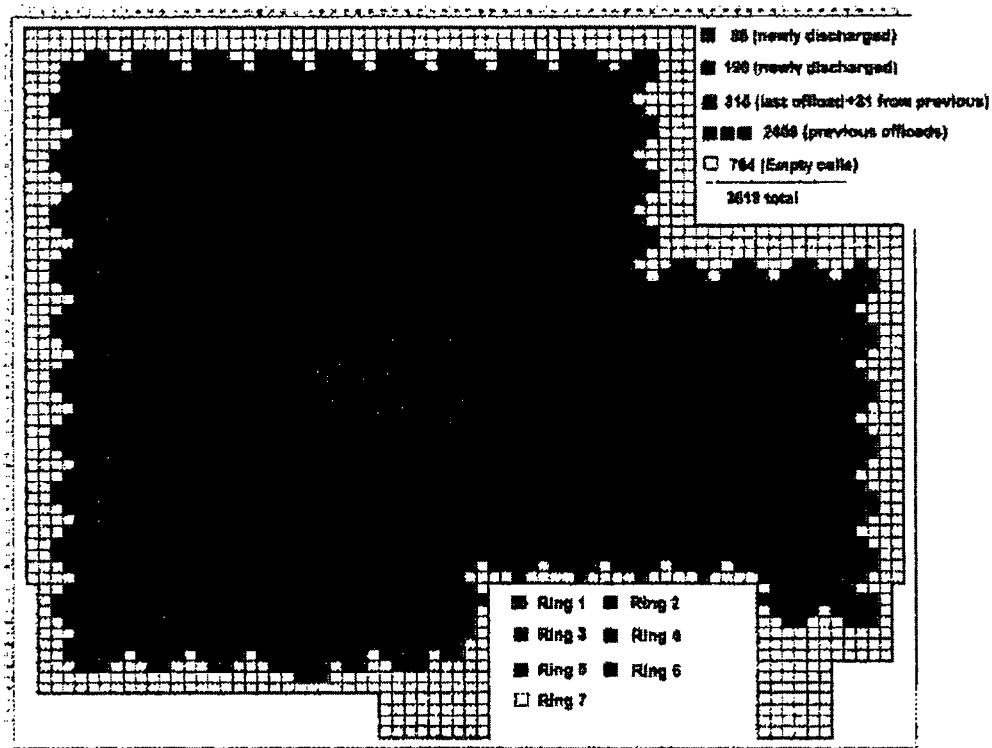

Figure 96: Layout of assemblies for OCP2 high density (uniform) model

Unmitigated Moderate Leak (OCP1 Uniform) Scenario

The results of the calculation for the uniform OCP-1 are shown in Figure 97 and Figure 98. A comparison of the heatup with the 1x4 geometry (Figure 65) shows the higher temperatures in the uniform Ring 1 configuration as there is less surface area between Ring 1 and the colder assemblies in Ring 2. The overall thermal response, however, is comparable. At about 30 hours, Ring 1 experiences a gradual heat up as the oxygen in the building is depleted, and formation of debris restricts air flow through the assemblies. Eventually, all the fuel in Ring 1 collapsed and forms a debris bed. There is continuous release from Rings 1 and 2 and the overall Cs release to the environment is about twice of that in the 1x4 geometry (see Figure 70).

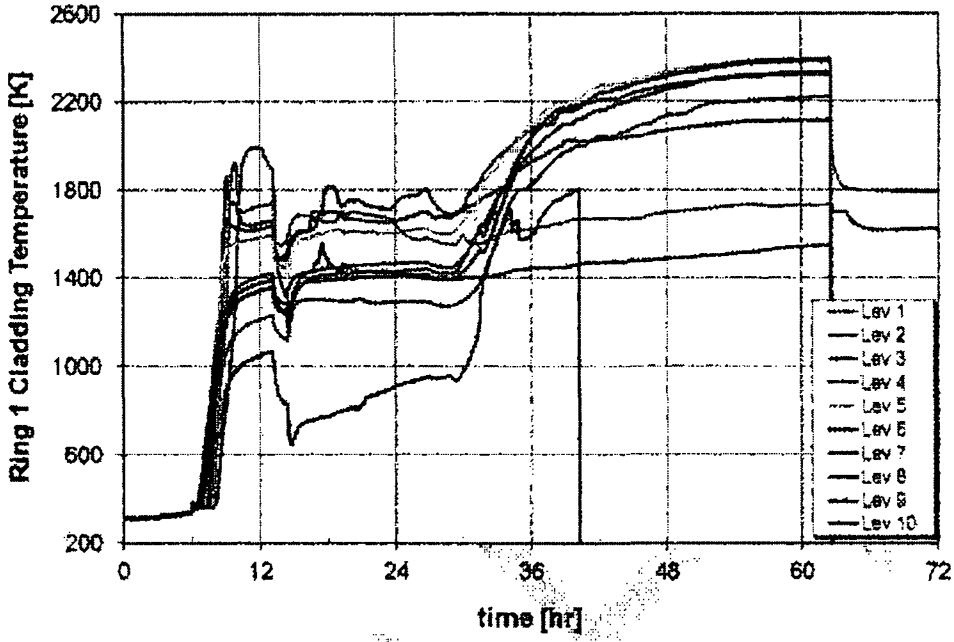

Figure 97: Ring 1 clad temperature for unmitigated uniform high density moderate leak (OCP1)

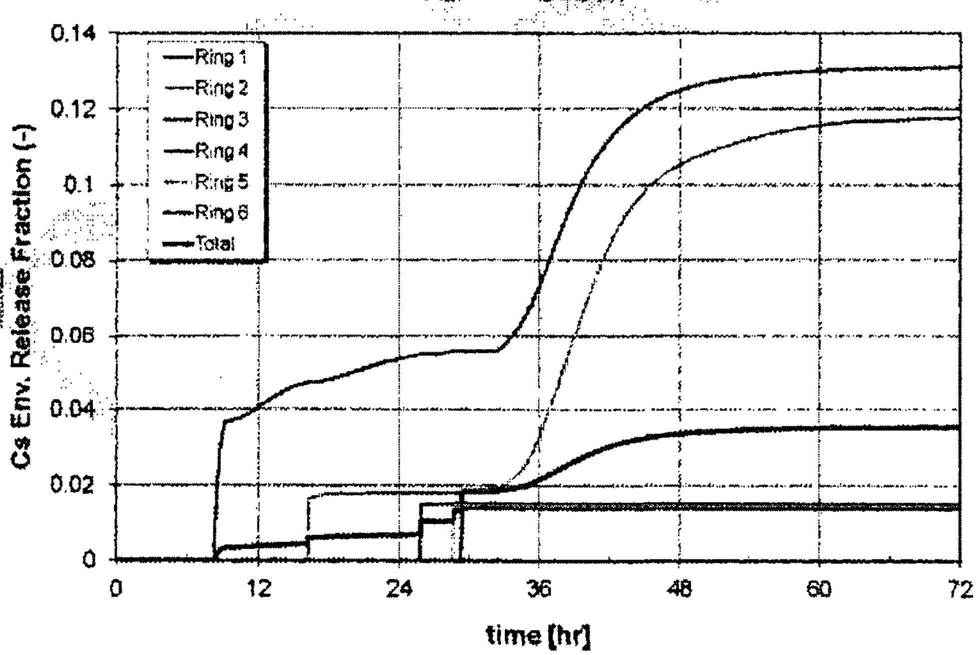

Figure 98: Cs environmental release fraction for unmitigated uniform high density moderate leak (OCP1)

Mitigated Moderate Leak (OCP2 Uniform) Scenario

For the mitigated case in OCP2 uniform configuration that had the highest Cs release fraction (1.2%), a number of calculations were performed to determine the effectiveness of mitigation. It should be noted that the same scenario in 1x4 configuration did not have any release. The overall behavior of fuel temperature is similar to 1x4 configuration cases in OCP2 (not shown) and OCP 1 (Figure 75), but the fuel is experiencing higher temperature that gradually declines. For this base case (Figure 99), temperatures are high enough to cause a gap release and more gradual release of fission products from the fuel. The calculation for the 200 gpm spray instead of 500 gpm makeup water is shown in Figure 100 that actually shows a rapid heatup before the temperatures are stabilized. A calculation was performed to test the effectiveness of a higher spray flow rate of 500 gpm and as indicated in Figure 101, the fuel temperature are stabilized at much lower temperatures without release of fission products from the fuel. In all the spray calculations performed here, the simple flow regime model was disabled because of a more stable and faster calculation and the previous results with OCP 3 had already showed that both models predict comparable maximum clad temperatures.

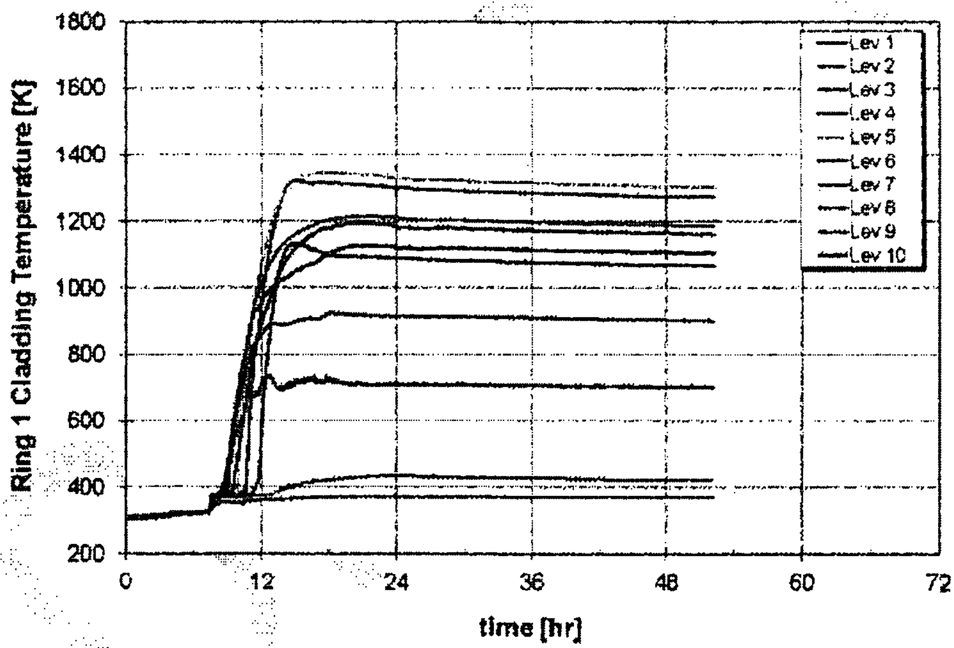

Figure 99: Ring 1 clad temperature for mitigated uniform high density moderate leak (OCP2) with 500 gpm injection

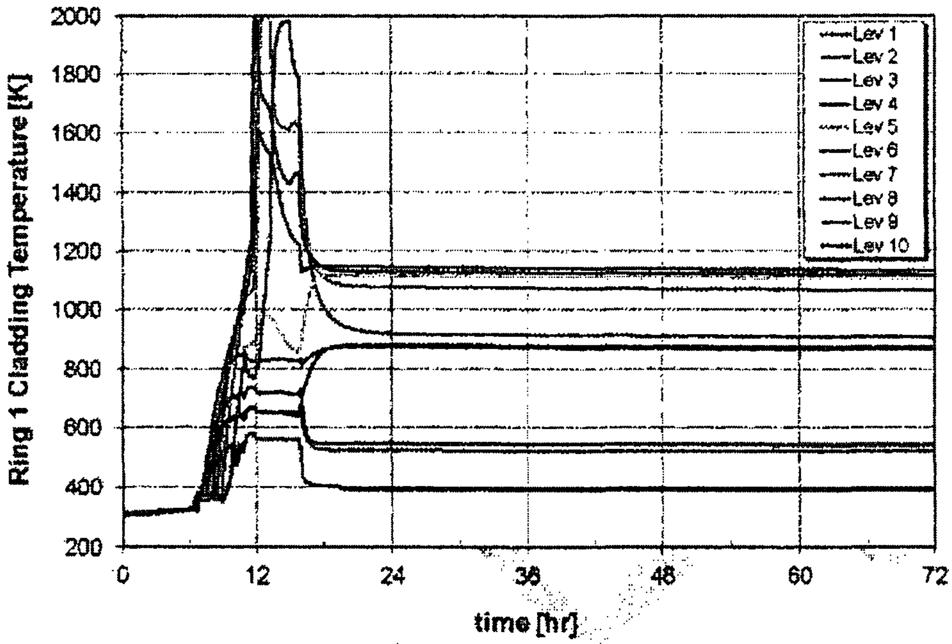

Figure 100: Ring 1 clad temperature for mitigated uniform high density moderate leak (OCP2) with 200 gpm spray

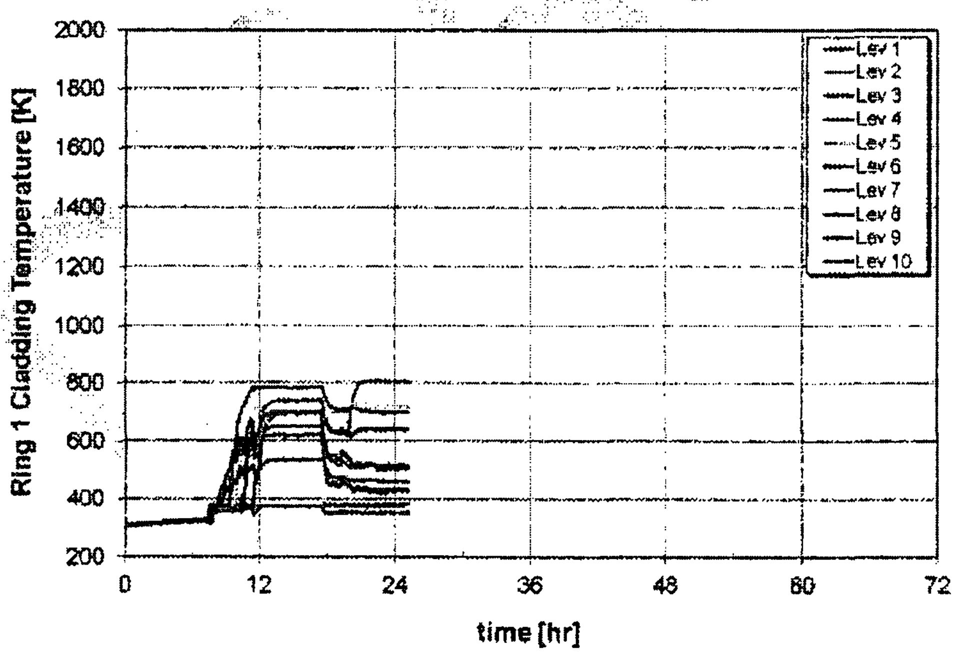

Figure 101: Ring 1 clad temperature for mitigated uniform high density moderate leak (OCP2) with 500 gpm spray

Table 44: Summary of release characteristics for high density, uniform configuration

High Density Case #	Scenario Characteristics					Release Characteristics			
	SFP Leakage?	50.54(hh)(2) Equipment?	Fuel Uncovery (hr)	Gap Release (hr)	Hydrogen Deflagration (hr)	NGs release at 72 hours	Cs-137 (MCi) Released	I release at 72 hours	I-131 (MCi) Released
OCP # 1	Small	No	39.7	52.3	No	0.8%	0.41	4.8%	0.38
	Moderate	Yes	7.4	11.7	No	0.6%	0.32	0.6%	0.05
	Moderate	No	5.9	8.2	No	3.6%	1.88	12.4%	0.97
OCP # 2	Small	No	42.6	55.2	65.4	4.2%	1.93	5.5%	0.61
	Moderate	Yes	7.3	12.7	No	1.2%	0.55	5.0%	0.56
	Moderate	No	5.9	8.8	21.65	49.1%	22.71	68.4%	7.65

10. SUMMARY & CONCLUSION

This report documents the results of applying a challenging (well-beyond-design-basis) seismic event, with a peak amplitude in the range of 0.5 to 1g and an estimated frequency of occurrence of 1 event in 60,000 years) to a selected Mark I boiling water reactor spent fuel pool. The results obtained are synopsized in the tables below...

Now let's revisit the original considerations regarding expedited fuel movement that this study sought to inform, providing insights from the work documented in this report:

- Expedited movement of fuel from the spent fuel pool to dry storage will decrease the inventory of longer-lived radionuclides such as Cesium-137.

OCP	High density (MCi)	Low density (MCi)	Ratio (low/high)
OCP # 1	57	17	0.29
OCP # 2	59	22	0.37
OCP # 3	59	22	0.37

- As a result of the above, less radioactive material would be present if a radioactive release occurred, which would be expected to reduce potential land contamination and economic impacts

  - ...

- Removal of older fuel reduces the overall heat load in the pool, which can have the effect of delaying the start of a radioactive release (and thus increasing the time available to take mitigative action) for many types of accidents

OCP	High density (kW)	Low density (kW)	Ratio (low/high)
OCP # 1	2951	2526	0.86
OCP # 2	3567	3143	0.88
OCP # 3	2571	2149	0.84

- Removal of older fuel will increase the volume available for cooling water

  - As mentioned before, this is mathematically a small effect with the older fuel comprising on the order of 5% of the total pool volume (recall that most of the pool is occupied by water, not fuel). In the scenarios studied here, a 5% difference in the initial water inventory generally would not have affected the course of the accident and the offsite consequences.

Regarding what this study tells us relative to the Quantitative Health Objectives in the Safety Goal Policy Statement, we invoke the discussion in Section 9.1, about how the results from a limited scope consequence assessment can be used to draw supportable, but not definitive, conclusions about risk. Let's start by comparing the scenario-specific frequencies from this study with those of past studies...

11. REFERENCES

[Alvarez, 2003] Alvarez, Robert, et al., "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States," as transmitted in a letter to Ashok C. Thadani dated April 21, 2003. [ML031130327]

[Congress, 2005] House Report 108-554, Energy and Water Development Appropriations Bill, 2005. Available at thomas.loc.gov

[redacted]

[EPRI, 2010] Electric Power Research Institute, "Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling," EPRI-TR-1021049, November 2010. Available at www.epri.com

[Exelon, 2011] Exelon Corporation, Letter from Michael Jesse to the U.S. Nuclear Regulatory Commission entitled, "License Amendment Request – Safety Limit Minimum Critical Power Ratio Change," dated June 8, 2011. [ML111600180]

[redacted]

[INL, 1996] Idaho National Laboratory, "Loss of Spent Fuel Pool Cooling PRA: Model and Results," INEL-96-0334, September 1996.

[NAC, 2011] NAC International, "NAC White Paper on Establishing a Balanced Perspective on Wet and Dry Storage of Used Fuel at U.S. Reactors," memorandum to U.S. Nuclear Regulatory Commission dated July 7, 2011.

[NAP, 2006] National Academies' Press, "Safety and Security of Commercial Spent Nuclear Fuel Storage," 2006. Available at www.nap.edu

[NEI, 2006] Nuclear Energy Institute, "B.5.b Phase 2 & 3 Submittal Guideline," NEI-06-12, Revision 2, December 2006. [ML070090060]

[NRC, 1979] U.S. Nuclear Regulatory Commission, "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, March 1979.

[NRC, 1989a] U.S. Nuclear Regulatory Commission, "Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools," NUREG/CR-5281, March 1989. [ML071690022]

[NRC, 1989b] U.S. Nuclear Regulatory Commission, "Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools"," NUREG-1353, April 1989. [ML082330232]

[NRC, 1990] U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990. [ML082330232]

[NRC, 1994] U.S. Nuclear Regulatory Commission, "Revised Livermore Seismic Hazard Estimate for Sixty-nine Nuclear Power Plant Sites East of the Rocky Mountain," NUREG-1488, April 1994. [ML052640591]

[NRC, 1996] U.S. Nuclear Regulatory Commission, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," NUREG-1437, May 1996.

[NRC, 1997a] U.S. Nuclear Regulatory Commission, "Operating Experience Feedback Report: Assessment of Spent Fuel Cooling," NUREG-1275, Volume 12, February 1997. [ML010670175]

[NRC, 1997b] U.S. Nuclear Regulatory Commission, Memo from L. Joseph Callan to the Commission entitled "Followup Activities on the Spent Fuel Pool Action Plan," dated September 30, 1997. [ML003706412]

[NRC, 2001] U.S. Nuclear Regulatory Commission, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, February 2001.

[NRC, 2002] U.S. Nuclear Regulatory Commission, "Analysis of Spent Fuel Heatup Following Loss of Water in a Spent Fuel Pool: A Users' Manual for the Computer Code SHARP," NUREG/CR-6441, March 2002. [ML021050336]

[NRC, 2003a] U.S. Nuclear Regulatory Commission, Letter from Ashok Thadani to Robert Alvarez with enclosure entitled, "NRC Review of "Reducing the Hazards from Stored Spent Power Reactor Fuel in the United States," dated August 19, 2003. [ML031210075]

[NRC, 2003b] U.S. Nuclear Regulatory Commission, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 Through 2002," NUREG-1774, July 2003. [ML032060160]

[NRC, 2005] U.S. Nuclear Regulatory Commission, Letter from Nils Diaz to Pete Domenici with enclosure entitled, "U.S. NRC Report to Congress on the NAS Study on the Safety and Security of Commercial Spent Nuclear Fuel Storage," dated March 14, 2005. [ML050280428]

[NRC, 2006] U.S. Nuclear Regulatory Commission, "Evaluation of Human Reliability Analysis Methods Against Good Practices," NUREG-1842, September 2006. []ML063200058]

[NRC, 2007a] U.S. Nuclear Regulatory Commission, "A Pilot Probabilistic Risk Assessment Of a Dry Cask Storage System At a Nuclear Power Plant," NUREG-1864, March 2007. [ML071340012]

[NRC, 2007b] U.S. Nuclear Regulatory Commission, "Regulatory Issue Summary 2005-25, Supplement 1: Clarification of NRC Guidelines for Control of Heavy Loads, dated May 29, 2007. http://www.nrc.gov/reading-rm/doc-collections/gen-comm/reg-issues/

[NRC, 2007c] U.S. Nuclear Regulatory Commission, "ATHENA User's Guide," NUREG-1880, June 2007. [ML071770660]

[NRC, 2008a] U.S. Nuclear Regulatory Commission, "Denial of Two Petitions for Rulemaking Concerning the Environmental Impacts of High-Density Storage of Spent Nuclear Fuel in Spent Fuel Pools (PRM-51-10 and PRM-51-12," SECY-08-0036 dated March 7, 2008 and SRM-SECY-08-0036 dated June 19, 2008. [ML080170587, ML081710446]

[NRC, 2008b] U.S. Nuclear Regulatory Commission, "Risk Assessment of Operational Events Handbook," Volume 2, Revision 1.01, January 2008.

[NRC, 2009a] U.S. Nuclear Regulatory Commission, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Draft Report for Comment" NUREG-1437 Revision 1, July 2009.

[NRC, 2010] U.S. Nuclear Regulatory Commission, "Documentation of Evolution of Security Requirements with Respect to Mitigation Measures for Large Fires and Explosions," dated February 4, 2010. [ML092990438]

[NRC, 2011a] U.S. Nuclear Regulatory Commission, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," SECY-11-0093, July 12, 2011. [ML11186A950]

[NRC, 2011b] U.S. Nuclear Regulatory Commission, "Human Reliability Analysis-Informed Insights on Cask Drops," NUREG/CR-7016, February 2012. [ML12067A135]

[NRC, 2011c] U.S. Nuclear Regulatory Commission, "Preliminary, Qualitative Human Reliability Analysis for Spent Fuel Handling," NUREG/CR-7017, February 2012. [ML12067A137]

[NRC, 2012] U.S. Nuclear Regulatory Commission & Electric Power Research Institute "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," NUREG-1921 / EPRI TR-1023001, *To be published*. [Interim Final Version at ML120790453]

[USGS, 2008]

[redacted] Wagner, K.C. and Gauntt, R.O., "Evaluation of a BWR Spent Fuel Pool Accident Response to Loss-of-Coolant Inventory Scenarios Using MELCOR 1.8.5," Sandia Letter Report, Rev. 3, June 2008 (ML082820262).

[redacted] Wagner, K.C. and Gauntt, R.O., "Additional MELCOR Analyses of BWR Spent Fuel Pool Assembly Accident Response," Sandia Letter Report, Rev. 3, June 2008 (ML081680064).

[redacted] Wagner, K.C. and Gauntt, R.O., "Analysis of Emergency Spray Mitigation of Spent Fuel Pool Loss-of-Coolant Inventory Accidents," Sandia Letter Report, Rev. 2, January 2008 (ML081680027).

[redacted] Wagner, K.C., et al., "MELCOR 1.8.5 Separate Effect Analyses of Spent Fuel Pool Assembly Accident Response," Sandia National Laboratories, Sandia Letter Report, April 2003 (ML082820136).

[redacted] Chiffelle, R., et al., "Analysis of Spent Fuel Pool Flow Patterns Using Computational Fluid Dynamics: Part 1 – Air Cases," Sandia National Laboratories, Sandia Letter Report, April 2003 (ML082820144).

[redacted] Ross, K. W., et al., "Analysis of Spent Fuel Pool Flow Patterns Using Computational Fluid Dynamics: Part 2 – Partial Water Cases," Sandia National Laboratories, Sandia Letter Report, April 2003 (ML082820239).

[Natesan,2004] Natesan, K. and Soppet, W.K., "Air Oxidation Kinetics for Zr-Based Alloys," NUREG/CR-6846, Argonne National Laboratories, June 2004.

[Durbin,2005] Durbin, S., "Hydraulic Analysis of the Spent Fuel Pool Experiment," Revision 4, Sandia National Laboratories Internal Memo dated 11/21/05.

[NRC,2012] Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project. Volume 1: Peach Bottom Integrated Analysis," NUREG/CR-7110, January 2012.

[NRC,1995] Nuclear Regulatory Commission, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465,February 1995.

[redacted] Khalil, I., and Webb, S., "Simulations of Natural Convection Airflow in a Completely Drained Pressurized Water Reactor Spent Fuel Pool using FLUENT," Sandia National Laboratories, Letter Report, Revision 1, October 2005.

[Gauntt,2010] Gauntt, R.O., "MELCOR 1.8.5 Modeling Aspects of Fission Product Release, Transport and Deposition," Sandia Report, SAND2010-1635, April 2010.

[Gauntt,2005] Gauntt, R.O., "MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide, Vol. 2: Reference Manual, Version 1.8.6," NUREG/CR 6119, Nuclear Regulatory Commission, Washington, D.C. 2005.

[Gauld,2011] Gauld, I.C., et al., "Isotopic Depletion and Decay Methods and Analysis Capabilities in SCALE," *Nuclear Technology* **174**, 2, 169, 2011.

[Rearden, 2012] Rearden, B. T., et al. , "Enhancements in SCALE 6.1," PHYSOR 2012 – Advances in Reactor Physics – Linking Research, Industry, and Education, Knoxville, Tennessee, USA, April 15-20, 2012, on CD-ROM, American Nuclear Society, LaGrange Park, IL, 2012.

[Ilas , 2006] Ilas, G. et al., "LWR Cross Section Libraries for ORIGEN-ARP in SCALE 5.1," *Trans. Am. Nucl. Soc.* **95**, 706 2006.

[Ilas , 2008] Ilas, G. and Gauld, I.C., "SCALE analysis of CLAB decay heat measurements for LWR spent fuel assemblies," *Annals of Nuclear Energy* **35**, 37–48, 2008.

APPENDIX A: FREQUENTLY ASKED QUESTIONS