List of Attachments

Attachment 1	Declaration of Assistant Attorney General John Sipos, State of New York, December 6, 2013.
Attachment 2	Statement of Timothy Mahilrajan, International Safety Research, Inc. (“ISR"), December 6, 2013.
Attachment 3	Draft Report, Consequence Study of a Beyond-Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boil Water Reactor, June 26, 2013 (ML13133A132) (“Spent Fuel Pool Consequence Study”).
Attachment 4	Official Transcript of Proceedings, Japan Lessons Learned Project Directorate, Public Meeting, September 18, 2013 (ML13277A215) (“9/18 Transcript”) (Excerpt)
Attachment 5	Various NYS & NRC communications re request for MACCS2 Input and Output Files for Spent Fuel Pool Consequence Study.
Attachment 6	“Package Description” and various MACCS2 Input and Output Files for Spent Fuel Pool Consequence Study (with date of November 13, 2012)
Attachment 7	NRC Staff SECY-13-0112, Consequence Study of a Beyond Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boil Water Reactor, October 9, 2013 (ML13256A339) (posted on public ADAMS on October 22, 2013) & [Final] Consequence Study of a Beyond Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boil Water Reactor, October 2013 (ML13256A342) (posted on public ADAMS on October 23, 2013).

UNITED STATES
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In re:	Docket Nos. 50-247-LR; 50-286-LR
License Renewal Application Submitted by	ASLBP No. 07-858-03-LR-BD01
Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 December 7, 2013

Declaration of Assistant Attorney General John Sipos
in Support of State of New York's
Motion to Reopen the Record and for Reconsideration
of Board Ruling LBP-13-13 on Contention NYS-12C

Declaration of John J. Sipos

Pursuant to 28 U.S.C. § 1746, John J. Sipos hereby declares as follows:

1. I serve as an Assistant Attorney General for the State of New York, counsel for petitioner-intervenor State of New York in this proceeding. In addition, I represent the State in various other ongoing matters before the Commission and Staff. I submit this declaration and accompanying attachments in support of the State's motion to reopen and supplement the record and for reconsideration of the Atomic Safety and Licensing Board's November 27, 2013 ruling on Contention NYS-12C contained in the Partial Initial Decision on Track 1 Contentions (LBP-13-13). In brief, as a result of the State's participation in another matter, the State has learned that NRC Staff used a materially different input value for the TIMDEC input in a MACCS2 analysis than that which it accepted and advocated for in this proceeding.

2. On behalf of the State of New York and in connection with ongoing spent fuel pool and waste issues, I attended a public meeting held in the NRC Commissioners' Meeting Room at NRC headquarters on September 18, 2013 that NRC convened to answer questions concerning the draft Spent Fuel Pool Consequence Study.[1] NRC published a transcript of the September 18 meeting ("9/18 Tr.").[2] Staff explained that it

[1] Draft Report, Consequence Study of a Beyond-Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boil Water Reactor, June 26, 2013 (ML13133A132) [Attachment 3]. Although the State had listened in on an earlier August 22, 2013 public meeting concerning the Consequence Study, NRC Staff did not provide an opportunity for the State to present its questions during that earlier meeting [Attachment 5].

[2] Official Transcript of Proceedings, Japan Lessons Learned Project Directorate, Public Meeting, September 18, 2013 (ML13277A215) ("9/18 Transcript"). An excerpt of that 245-page transcript is included as Attachment 4.

prepared the Consequence Study as an outgrowth of the accident at the Fukushima facilities and in response to a directive from the Commissioners.

3. During the September 18, 2013 meeting, I asked questions about the Consequence Study (9/18 Tr. at 89-103). In response to those questions, Staff stated that:

- the Consequence Study included a MACCS2 analysis (9/18 Tr. at 90-91);
- the Consequence Study examines a severe accident at one of the spent fuel pools at the Peach Bottom site in central Pennsylvania (9/18 Tr. at 97-98);
- NRC Staff had worked with Dr. Nathan Bixler and Mr. Joseph Jones of Sandia National Laboratories in developing the MACCS2 analysis for the Consequence Study (9/18 Tr. at 96-97, 99-100); and
- the MACCS2 computer analyses or runs for the Consequence Study were done in November or December 2012 (9/18 Tr. at 94).

At the September 18, 2013 meeting, I requested the input and output files for the MACCS2 computer runs (9/18 Tr. at 94-96). NRC stated it would consider the request. *Id.*

4. On October 23, 2013, NRC informed the State that a PDF format version of the input and output files were placed on the public side of NRC's Agencywide Documents Access and Management System (ADAMS). October 23, 2013 NRC email [Attachment 5].[3]

5. New York requested that International Safety Research, Inc. review and analyze the Spent Fuel Pool Consequence Study and the underlying MACCS analysis in connection with another matter. ISR requested native format files of the input and output files for the MACCS analysis for the Consequence Study.

[3] At about the same time, NRC Staff finalized the Spent Fuel Pool Consequence Study and presented it to the NRC Commissioners. NRC Staff SECY-13-0112, Consequence Study of a Beyond Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boil Water Reactor, October 9, 2013 (ML13256A339) (posted on public ADAMS on October 22, 2013); [Final] Consequence Study of a Beyond Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boil Water Reactor, October 2013 (ML13256A342) (posted on public ADAMS on October 23, 2013) [Attachment 7].

6. On October 28, 2013, I clarified that New York's request was for the native format files of the input and output files. October 28, 2013 NYS email [Attachment 5]. New York received a disc containing the native format files on Tuesday, November 26, 2013. November 27, 2013 NYS email & November 18, 2013 NRC letter (marked received on November 26) [Attachment 5].[4] New York forwarded the native format files on to ISR.

7. Accompanying the native format files was a "package description" explaining the contents of the disc [Attachment 6]. That project description states that there were 21 sets of MACCS2 calculations. *Id*.

8. As set forth in the accompanying statement of Timothy Mahilrajan, ISR has confirmed and informed the State that the input and output files show that NRC used a 365 day (one year) input value for the TIMDEC input for the MACCS2 code analysis that was conducted for the Spent Fuel Pool Consequence Study. Mr. Mahilrajan notes that the NRC used a one year TIMDEC input for both light and heavy decontamination scenarios in that MACCS analysis.[5] Attachment 2.

9. The State respectfully submits that NRC's use of a 365 day (one year) value for the TIMDEC input in a MACCS2 analysis is relevant and material to the State's position on Contention NYS-12. As the Board is aware, in the Indian Point proceeding Entergy used, and Staff accepted, TIMDEC input values of 60 and 120 days for light and heavy decontamination, respectively. Among other challenges to the MACCS2 analysis, the State demonstrated that using a TIMDEC input value of 365 days would more than

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[4] This email includes an incorrect parenthetical description of the date of receipt; it should have read "(Nov. 26)", not "(Nov. 27)".

[5] The term "light" decontamination refers to a decontamination factor of 3 and "heavy" decontamination referred to a decontamination factor of 15.

double the offsite economic cost risk ("OECR"). NYS000430 at 6, Table 13 (ISR analysis regarding a light decontamination scenario with a TIMDEC input of one year and a heavy decontamination scenario with a TIMDEC input of two years); Tr. 2205:20-2206:5(Lemay); State of New York's Proposed Findings of Fact and Conclusions of Law for Contention NYS-12/12A/12B/12C, at ¶ 269. Given the fact that Entergy testified that "at 11%, IP2 SAMA 025 has the smallest margin between the current benefit and the increased benefit to become cost effective,"[6] doubling the OECR would render at least one additional SAMA candidate cost-beneficial. It would also render existing cost-beneficial SAMA candidates more cost-beneficial.

10. In this proceeding, Entergy's witnesses and NRC Staff's witnesses testified, both in prefiled testimony and at the hearing, that the use of TIMDEC input values of 60 and 120 days was consistent with other MACCS2 analyses performed or accepted by NRC Staff and that, therefore, those two inputs were reasonable under NEPA.

11. In this proceeding, Mr. Jones and Dr. Bixler of Sandia National Laboratories testified on behalf of NRC Staff. Dr. Tina Ghosh and Mr. Donald Harrison also presented testimony on behalf of NRC Staff.

12. The evidentiary hearing on Contention NYS-12 concluded in Tarrytown, NY on October 18, 2012.

13. Staff stated that the MACCS2 runs for the Consequence Study were performed in November or December 2012. 9/18 Tr. at 94 [Attachment 4].

[6] ENT000450 Pre-filed Testimony of Entergy Experts Potts, O'Kula, and Teagarden on Consolidated Contention NYS-12C (Mar. 30, 2012) at 49-50 (A89) (O'Kula, Teagarden, Potts).

14. As noted, Dr. Bixler and Mr. Jones consulted with NRC Staff in the preparation of the MACCS2 analysis for the Spent Fuel Pool Consequence Study – in addition to testifying in this proceeding in support of Staff's MACCS2 / SAMA analysis for the proposed renewal of the operating licenses for Indian Point facilities.

15. Staff and its Sandia witnesses did not disclose the November 2012 MACCS2 input and output files for the Spent Fuel Pool Consequence Study to the State in this proceeding.

16. In its March 2013 post-hearing submission for Contention NYS-12C, NRC Staff stated:

> ¶ 5.41: "the inputs selected by Entergy for TIMDEC and CDNFRM have a long history of use for exactly this kind of analysis and continue to be used in the most recent state of the art reactor consequence accident ("SOARCA") analysis by the NRC Staff on these and similar types of issues. . . . These NUREG-1150 values continue to be utilized in the NRC's current analysis involving severe accident mitigation analysis and PRA analysis including the SOARCA and with the Staff's responses to tasking related to Fukushima Dai-ichi."

NRC Staff Proposed Findings of Fact and Conclusions of Law (March 22, 2013) ML13081A698.

17. Entergy made similar statements in its March 2013 post-hearing proposed Findings of Fact and Conclusions of Law:

> ¶ 216 "These [TIMDEC] values are consistent with the NUREG-1150 values and have been applied in Level-3 type PRA analyses (including SAMA analyses and the SOARCA project) for many years."
>
> ¶ 248 "The challenged TIMDEC values also are consistent with the NUREG-1150 values and have been applied in Level-3 PRA analyses (including SAMA analyses and the SOARCA project) for many years."

Entergy Proposed Findings of Fact and Conclusions of Law for Consolidated Contention NYS-12C (March 22, 2013) ML13081A743. Similar statements appear in its post-hearing reply filings. Entergy Reply to New York State's Proposed Findings of Fact and Conclusions of Law for Contention NYS-12C at ¶¶ 55, 82 ("[The TIMDEC values used for Indian Point] are consistent with the NUREG-1150 values …") (May 3, 2013) ML13123A461.[7]

18. The Board found these arguments and testimony presented by NRC Staff and Entergy persuasive in reviewing the reasonableness of the 60 and 120 day TIMDEC input values: "we note that the NRC Staff's witnesses Mr. Harrison and Dr. Ghosh testified that the NRC has examined decontamination times for more than 37 years, beginning in 1975 with the Reactor Safety Study, which discussed decontamination activities that are capable of restoring areas to habitability quickly given sufficient resources." ASLB LBP-13-13, Partial Initial Decision at 286 (citing NRC Staff NYS-12C/16B Testimony at 89 (Ex. NRC000041)).

19. One month following the conclusion of the evidentiary hearing on Contention NYS-12C, NRC Staff discontinued its 37-year practice of using only 60 days and 120 days as TIMDEC inputs for accident analyses. Staff did not disclose this to the State (or the Board) in its submissions in this proceeding.

20. In preparation for this motion, the State of New York has reviewed NRC Staff's monthly disclosures from November 2012 through November 2013. The State

[7] Entergy's pre-hearing Statement of Position stated:

> pp. 5-6: "NYS's criticisms of Entergy's MACCS2 inputs also lack merit. . . . [T]hose inputs have been subject to extensive peer review since the late 1980s and continue to be used in licensee probabilistic risk assessments ("PRAs") and SAMA analyses and state-of-the-art severe accident analyses conducted by the NRC. . . . [T]hey continue to be used today in PRA applications, including the NRC's State-of-the-Art Reactor Consequence Analyses ("SOARCA") project and licensee SAMA analyses.

Entergy Statement of Position Regarding Consolidated Contention NYS-12C (March 30, 2012) (ENT000449) ML12090A796.

found no indication that Staff disclosed the November 2012 MACCS2 input and output files for the Spent Fuel Pool Consequence Study in this proceeding.

21. Attachment 2 to this motion is the declaration of Timothy Mahilrajan of ISR. Attachments 3 to 7 to this motion are true and correct copies of NRC documents or communications between NRC and New York.

22. I declare under penalty of perjury that the foregoing is true and correct.

Executed on December 7, 2013

Signed (electronically) by

____

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

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In re:	Docket Nos. 50-247-LR; 50-286-LR
License Renewal Application Submitted by	ASLBP No. 07-858-03-LR-BD01
Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 December 7, 2013

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Declaration of Timothy Mahilrajan of International Safety Research, Inc.

Declaration of Timothy Mahilrajan

Pursuant to 28 U.S.C. § 1746, Timothy Mahilrajan declares:

1. I am a project manager with the firm International Safety Research, Inc. (or "ISR"). Dr. François Lemay is out of the country on previously-scheduled business. In Dr. Lemay's absence, I submit this declaration for ISR on behalf of the State of New York.

2. I have been employed by ISR for approximately 7 years. My education experience includes a Bachelor's of Engineering, Engineering Physics (2005), and I am currently completing a Masters in Engineering, Nuclear Engineering. I have also completed training (2008) delivered by Sandia National Laboratories on the use of WinMACCS/MACCS2 for nuclear accident consequence modeling. I have experience with the MACCS2 computer code, attended the October 2012 evidentiary hearing on Contention NYS-12 in Tarrytown, and am familiar with the evidence presented by the parties in this proceeding in connection with Contention NYS-12.

3. The State of New York requested International Safety Research to review the Spent Fuel Pool Consequence Study prepared by NRC Staff following the accident at the Fukushima nuclear facilities. Consequence Study of a Beyond Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boil Water Reactor, Attachment 7, ML13256A342 (final) October 2013, and Attachment 3, ML13133A132 (draft) June 2013.

4. The Spent Fuel Pool Consequence Study includes a MACCS2 computer code analysis of a severe accident occurring at a single spent fuel pool located at the Peach Bottom site near Lancaster in Pennsylvania.

5. After the input and output files for that study were posted in PDF format, I requested that the State provide ISR with native format files for the input and output files for the MACCS2 analysis for the Consequence Study.

6. I have reviewed the native format files for the inputs and outputs for the MACCS2 analysis for the Consequence Study and the accompanying "package description" provided by NRC. That package description states that NRC provided 7 project folders when it provided the native format files: "In this zip file, there are the 5 project folders from the base case MACCS2 (rev 3.7.0) calculations for the Spent Fuel Pool Study, and an additional 2 project folders for a sensitivity for alternative source terms based on a uniform loading distribution for an applicable amount of time after reactor shutdown. These 7 project folders represent 7 source terms, each containing 3 different applications of dose response for a total of 21 calculations." Package Description (included in Attachment 6). My review of the various input and output files (native and PDF) has confirmed to me and led me to conclude that NRC used 365 days as the value for the input TIMDEC in each of the 21 calculations examined in the MACCS2 analysis.

7. Specifically, the input and output files show that NRC Staff keyed in 3.15E+07[1] as the TIMDEC input for both the light and heavy decontamination levels. In the MACCS2 code, TIMDEC is entered in seconds, and 3.15E+07 seconds corresponds to 365 days or one year (i.e., 3.15×10^7 seconds = 8,750 hours, which rounds up to 365 days).

8. Copies of 2 of the 21 MACCS2 output files (PDF format version) are also included in Attachment 6. Red circles were added to the documents to identify where the 365

[1] 3.15E+07 is 31,500,000

day TIMDEC inputs appear. I have reviewed and compared the native format files and the PDF files of the MACCS2 calculations, and they all reflect a 365 day input for TIMDEC.[2]

9. Based on my review of the native format files, I have prepared a chart (shown below) that summarizes the location where TIMDEC appears in those files; at each of those locations, the operator assigned "3.15E+07" as the input for TIMDEC.

Line Numbers for TIMDEC Values in the MACCS2 Native Files for NRC Consequence Study

Each of the lines listed below contain a value of 3.15E+07 [seconds] for TIMDEC

Source term and fuel density	Dose threshold	Input file name	Input file line numbers	Output file name	Output file line numbers
1.4 HighDensity	LNT	Chronc1.inp	40-41	LNT 1.4 HighDensity.out	4094-4095
1.6u HighDensity	LNT	Chronc1.inp	40-41	LNT 1.6u HighDensity.out	9647-9648
2.4 HighDensity	LNT	Chronc1.inp	40-41	LNT 2.4 HighDensity.out	9095-9096
2.6 HighDensity	LNT	Chronc1.inp	40-41	LNT 2.6 HighDensity.out	4666-4667
2.6u HighDensity	LNT	Chronc1.inp	40-41	LNT 2.6u HighDensity.out	9667-9668
3.4 HighDensity	LNT	Chronc1.inp	40-41	LNT 3.4 HighDensity.out	9646-9647
3.4 LowDensity	LNT	Chronc1.inp	40-41	LNT 3.4 LowDensity.out	4295-4296
1.4 HighDensity	5 rem	Chronc3.inp	40-41	5 rem 1.4 HighDensity.out	4096-4097
1.6u HighDensity	5 rem	Chronc3.inp	40-41	5 rem 1.6u HighDensity.out	9649-9650
2.4 HighDensity	5 rem	Chronc3.inp	40-41	5 rem 2.4 HighDensity.out	9097-9098
2.6 HighDensity	5 rem	Chronc3.inp	40-41	5 rem 2.6 HighDensity.out	4668-4669
2.6u HighDensity	5 rem	Chronc3.inp	40-41	5 rem 2.6u HighDensity.out	9669-9670
3.4 HighDensity	5 rem	Chronc3.inp	40-41	5 rem 3.4 HighDensity.out	9649-9650
3.4 LowDensity	5 rem	Chronc3.inp	40-41	5 rem 3.4 LowDensity.out	4297-4298
1.4 HighDensity	620 mrem	Chronc2.inp	40-41	620 mrem 1.4 HighDensity.out	4096-4097
1.6u HighDensity	620 mrem	Chronc2.inp	40-41	620 mrem 1.6u HighDensity.out	9649-9650
2.4 HighDensity	620 mrem	Chronc2.inp	40-41	620 mrem 2.4 HighDensity.out	9097-9098
2.6 HighDensity	620 mrem	Chronc2.inp	40-41	620 mrem 2.6 HighDensity.out	4668-4669
2.6u HighDensity	620 mrem	Chronc2.inp	40-41	620 mrem 2.6u HighDensity.out	9669-9670
3.4 HighDensity	620 mrem	Chronc2.inp	40-41	620 mrem 3.4 HighDensity.out	9649-9650
3.4 LowDensity	620 mrem	Chronc2.inp	40-41	620 mrem 3.4 LowDensity.out	4297-4298

[2] The 21 MACCS2 output files (PDF) range from 79 pages to 246 pages in length. The native format files require Notepad or a similar application to open.

10. Based on information in the input and output files, it appears that NRC Staff performed the MACCS2 analyses that serve as the basis for the Spent Fuel Pool Consequence Study in mid-November 2012. The MACCS2 input and output files contain dates of "11/13/2012," "11/16/2012," and "11/17/2012") [Attachment 6]. Red circles were added to the documents to highlight these dates.

11. I declare under penalty of perjury of the laws of the United States of America that the foregoing is true and correct.

Executed in Accord with 10 C.F.R. § 2.304(4)

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December 7, 2013

Attachment 3

Draft Report

Consequence Study of a Beyond-Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boil Water Reactor (ML13133A132) ("Spent Fuel Pool Consequence Study")

June 26, 2013

Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor

Draft Report
June 2013

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FOREWORD

U.S. nuclear power plants are required to be designed with appropriate consideration of the most severe natural phenomena (e.g. floods, earthquakes, tornadoes) historically reported for their location and surrounding regions, with sufficient margin, to ensure that important safety functions can be performed. As part of our mission to protect public health and safety, the U.S. Nuclear Regulatory Commission (NRC) uses advanced computer modeling and other techniques to study more severe, and highly unlikely, events that go beyond what the plant was designed to withstand to estimate risk to the public and to explore and ensure safety margins.

On March 11, 2011, the Tohoku earthquake and subsequent tsunami in Japan resulted in significant damage to the site of the Fukushima Dai-ichi nuclear power station. Although the spent fuel pools and the used fuel assemblies stored in the pools remained intact at the plant, the event led to questions about the safe storage of spent fuel and whether the NRC should require the expedited transfer of spent fuel from pools to dry cask storage containers at U.S. nuclear power plants.

This report documents the Office of Nuclear Regulatory Research's consequence study that continues our examination of the risks and consequences of postulated spent fuel pool accidents. A spent fuel pool's robust concrete structure and stainless steel liner keep more than 20 feet of water above the spent fuel stored within it ensuring ample cooling for the spent fuel and adequate radiation shielding for plant personnel. About every two years, some used fuel is removed from the reactor and placed into the spent fuel pool. The used fuel most recently removed from a reactor is radiologically and thermally "hot". The hot fuel is distributed throughout the pool and is surrounded by older, cooler used fuel. After used fuel has cooled in the spent fuel pool for more than about five years, it has radiologically decayed such that it can be moved to dry storage casks for longer term storage.

This study compared potential accident consequences from a pool nearly filled with spent fuel and a pool in which fuel that has cooled sufficiently has been removed. The staff first evaluated whether a severe, though unlikely, earthquake would damage the spent fuel pool to the point of leaking. In order to assess the consequences that might result from a spent fuel pool leak, the study assumed seismic forces greater than the maximum earthquake reasonably expected to occur at the reference plant location. The NRC expects that the ground motion used in this study is more challenging for the spent fuel pool structure than that experienced at the Fukushima Daiichi nuclear power plant from the earthquake that occurred off the coast of Japan on March 11, 2011. That earthquake did not result in any spent fuel pool leaks. In the small likelihood that such an extreme earthquake caused a leak, the staff then analyzed how the spent fuel could overheat and potentially release radioactive material into the environment. Finally, the staff analyzed what the public health and environmental effects of a radiological release would be in the area surrounding the plant. In order to estimate the hypothetical consequences, the staff analyzed scenarios where some preplanned and improvised mitigative actions by the emergency response organization were either not successful or not implemented.

The study results for the specific reference plant and earthquake analyzed are consistent with past studies' conclusions that spent fuel pools are likely to withstand severe earthquakes without leaking. Past studies considered a wider range of earthquakes than this study. In the unlikely situation that a leak occurs, this study shows that for the scenarios and spent fuel pool studied, spent fuel is only susceptible to a radiological release within a few months after the fuel is moved from the reactor into the spent fuel pool. After that time, the spent fuel is coolable by air. This study shows the likelihood of a radiological release from the spent fuel after the analyzed severe earthquake at the reference plant to be about one time in 10 million years or lower. If a leak and radiological release were to occur, this study shows that the individual cancer fatality risk for a member of the public is several orders of magnitude lower than the Commission's Quantitative Health Objective of two in one million ($2x10^{-6}$/year). For such a radiological release, this study shows public and environmental effects are generally the same or smaller than earlier studies.

The Office of Nuclear Reactor Regulation's regulatory analysis for this study indicates that expediting movement of spent fuel from the pool does not provide a substantial safety enhancement for the reference plant. The NRC plans to use the insights from this analysis to inform a broader regulatory analysis of the spent fuel pools at all U.S. operating nuclear reactors as part of its Japan Lessons-learned Tier 3 plan. The NRC continues to believe, based on this study and previous studies that spent fuel pools protect public health and safety.

ABSTRACT

The U.S. Nuclear Regulatory Commission performed this consequence study to continue its examination of the risks and consequences of postulated spent fuel pool accidents. The study provides publicly available consequence estimates of a hypothetical spent fuel pool accident initiated by a low likelihood seismic event at a specific reference plant. The study compares high-density and low-density loading conditions and assesses the benefits of post 9/11 mitigation measures. Past risk studies have shown that storage of spent fuel in a high-density configuration is safe and risk of a large release due to an accident is very low. This study's results are consistent with earlier research conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking. The NRC continues to believe, based on this study and previous studies that spent fuel pools protect public health and safety. The study's results will help inform the Commission's evaluation of moving spent fuel from spent fuel pools to dry storage sooner than current practice.

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Rick Ennis	Steven Jones	Eric Schrader	Adam Ziedonis
Sam Hansell	Eric Powell	Bret Tegeler	

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission performed this consequence study to continue its examination of the risks and consequences of postulated spent fuel pool accidents. Pertinent research conducted over the last several decades is summarized in NUREG-1353, “Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools, April 1989; in NUREG/CR-6451, “A Safety and Regulatory Assessment of Generic BWR [boiling water reactor] and PWR [pressurized water reactor] Permanently Shutdown Nuclear Power Plants,” April 1997 and in NUREG-1738, “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants,” February 2001. The purpose of this consequence study was to determine if accelerated transfer of older, colder spent fuel from the spent fuel pool at a reference plant to dry cask storage significantly reduces risks to public health and safety. The specific reference plant used for this study is a GE Type 4 BWR with a Mark I containment.

The study’s results will help inform the Commission’s evaluation of moving spent fuel from spent fuel pools to dry storage sooner than current practice. This study’s results are consistent with earlier research studies’ conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking cooling water and potentially uncovering the spent fuel. The study shows the likelihood of a radiological release from the spent fuel after the analyzed severe earthquake at the reference plant to be about one time in 10 million years or lower. In addition, the regulatory analysis included with this study does not support accelerated spent fuel transfer to casks for the reference plant. The NRC plans to use the insights from this analysis to inform a broader regulatory analysis of the spent fuel pools at all U.S. operating nuclear reactors as part of its Japan Lessons-learned Tier 3 plan.

This study presents detailed analyses using state-of-the-art, validated, deterministic methods and assumptions, as well as probabilistic insights where practical. Previous studies have shown that earthquakes present the dominant risk for spent fuel pools, so this analysis considered a severe earthquake with ground motion stronger than the maximum earthquake reasonably expected to occur for the reference plant. The NRC expects that the ground motion used in this study is more challenging for the spent fuel pool structure than that experienced at the Fukushima Daiichi nuclear power plant from the earthquake that occurred off the coast of Japan on March 11, 2011. That earthquake did not result in any spent fuel pool leaks. This beyond-design-basis earthquake severity was selected to challenge the spent fuel pool integrity. The study considered two spent fuel configurations:

- A relatively full pool where the hottest spent fuel assemblies are surrounded by four cooler fuel assemblies in a 1×4 pattern throughout the pool (referred to as the high-density loading scenario), and;

- A minimally loaded pool where all spent fuel with at least 5 years of pool cooling has been removed so the hottest fuel assemblies are surrounded by additional water (referred to as the low-density loading scenario).

Limited sensitivity analyses of a 1x8 spent fuel configuration and a uniform configuration were also performed to better understand the potential effect of spent fuel configurations on the results.

Additionally, the study evaluated the potential benefits of strategies required in Title 10, Code of Federal Regulations (10 CFR), Part 50.54 (hh)(2) following the September 11, 2001, attacks. These “mitigation measures” are intended to maintain spent fuel pool cooling in the event of a loss of large areas of the plant due to explosions or fire.

The study evaluated 10 CFR 50.54(hh)(2) mitigation measures by analyzing each scenario twice – with and without credit for mitigation. The study shows that successful mitigation reduces the likelihood of a release. The likelihood of a spent fuel pool release was equally low for both high- and low-density fuel loading. This is because high- and low-density fuel loading contains the same amount of new, hotter spent fuel recently moved from the reactor to the spent fuel pool. In the unlikely event of an earthquake-induced spent fuel pool leak, the likelihood of fuel heatup leading to a release was more strongly affected by the fuel loading pattern rather than the total amount of fuel in the pool. In other words, the use of favorable fuel patterns such as the 1x4 pattern promotes natural circulation air coolability and reduces the likelihood of a release from a completely drained pool. Analysis also shows that for the scenarios and spent fuel pool studied, spent fuel is only susceptible to a radiological release within a few months after the fuel is moved from the reactor into the spent fuel pool. After that time, the spent fuel is coolable by air.

The study considered scenarios where some preplanned and improvised mitigative actions were either not successful or not implemented before three days, at which time the analysis was terminated. In addition to the 10 CFR 50.54(hh)(2) mitigation measures, the site emergency response organization would request support from the offsite response organizations to implement improvised additional mitigative measures, such as pumping water into the spent fuel pool using a fire truck. Analysis of these additional mitigative measures was beyond the scope of this study. Additionally, this study does not consider the post-Fukushima mitigation required by NRC in Orders EA-12-051 and EA-12-049 and currently being implemented by all U.S. nuclear power plants which should serve to further reduce spent fuel pool accident risk by increasing the capability of nuclear power plants to mitigate beyond-design-basis external events.

Figure ES-1 illustrates the study results in terms of the likelihood of a leak and magnitude of release from the spent fuel pool (SFP) for the severe, low likelihood earthquake considered in this study.

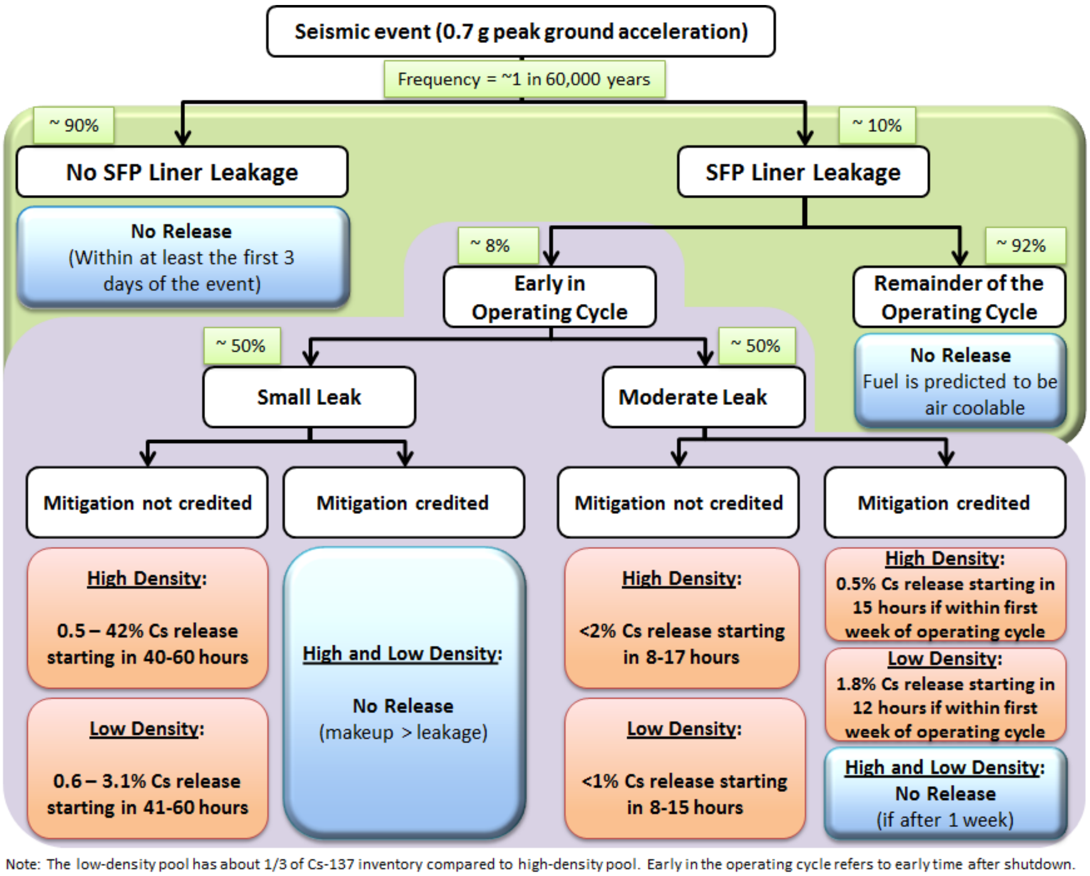

Figure ES-1: Likelihood of a leak and magnitude of releases from beyond design basis earthquake

This study considered a severe earthquake expected to occur once in 60,000 years; the pool is expected to remain intact during more likely, less severe earthquakes. The structural analysis of the pool shows the spent fuel pool in this study has a 90% probability of surviving the severe earthquake with no liner leakage (or conversely, a 10% probability of damaging the liner such that leakage will occur). The specific conditions for liner failure vary according to site conditions and spent fuel pool design. NUREG-1353 predicted the likelihood of liner failure from all potential earthquakes to be between about two and six times in a million years. NUREG-1738 predicted the likelihood of liner failure from all potential earthquakes to be between about two times in a million years and two times in 10 million years. This study considered an earthquake with ground motion roughly four to eight times stronger than that used in the plant design and predicted a liner failure likelihood of about two times in a million years.

The study examined how an accident is expected to proceed if the pool liner is damaged, concluding that pool leaks are somewhat less likely to release radioactive material to the environment than in previous studies. Depending on the size of the pool liner leak, releases could start anywhere from eight hours to several days after the leak starts assuming 10 CFR 50.54(hh)(2) mitigation measures are unsuccessful. In the event of an earthquake, releases are considered very unlikely for several reasons:

- The study finds liner damage is the only way to cause a radiological release in less than 3 days for the scenarios and spent fuel pool studied. Other possible outcomes provide time to prevent a release by taking emergency actions. This is consistent with earlier studies.

- The time period of susceptibility for a release of radioactive materials during the operating cycle is short. This study's detailed accident progression modeling differs from earlier work in showing that for the severe earthquake analyzed, draining the pool after liner failure is less likely to lead to a release. Because spent fuel can be effectively cooled by water, steam, or air, the likelihood of fuel overheating to the point of radiological release depends on several factors: how much residual heat the fuel generates, the fuel loading pattern, and the timing, location, and size of the liner leakage. If 10 CFR 50.54(hh)(2) mitigation measures aren't successful, releases could occur the first few months after the fuel came out of the reactor (or 8% of the reactor's two-year operating cycle). If 10 CFR 50.54(hh)(2) mitigation measures are successful, releases could only occur the first several days after the fuel came out of the reactor (a factor of twenty reduction in the likelihood of release).

In the unlikely event an earthquake induced liner failure does occur, this study predicts the largest releases would come from high-density loading cases without 10 CFR 50.54(hh)(2) mitigation measures. However, for each high-density loading release case, the corresponding low-density loading case also resulted in a release. The low-density cases generally resulted in a smaller release due to the smaller inventory of radioactive materials and the lower potential for hydrogen combustion. For the high-density cases, the releases are limited to a few percent of the cesium inventory, except for a few cases that predicted hydrogen combustion and resulted in releases of one to two orders of magnitude higher than the other cases. In these cases, the spent fuel heats up in a steam environment leading to oxidation of zirconium and releasing hydrogen gas into the reactor building. The mixing and reaction of hydrogen and oxygen leads to a hydrogen combustion and substantially damages the reactor building. That damage could breach structures that would retain radioactive material, along with allowing more oxygen into the building, potentially increasing the severity of the spent fuel fire. The study included a sensitivity analysis for a 1x8 loading pattern (hotter fuel surrounded by 8 cooler assemblies in a repeating pattern) which also resulted in smaller radioactive releases because the hotter assembly transfers its heat to the cooler assemblies resulting in lower peak fuel temperatures

Following the evaluation of successful and unsuccessful mitigation cases, a limited-scope human reliability analysis was performed to estimate the likelihood of successful operator actions implementing 10 CFR 50.54(hh)(2) mitigation measures to prevent fuel damage. Assumptions included post-earthquake on-site portable mitigation equipment required by 10 CFR 50.54(hh)(2) is available, minimum plant staffing are available for implementing spent fuel pool mitigation, and the work area is accessible to perform mitigation. The structural and accident progression analyses show that at least 99% of the time, the earthquake would not result in spent fuel overheating even without mitigative actions for the first seven days following the accident. For the remaining times, mitigative actions are needed to prevent fuel damage and the calculated mitigation success rates range from about 25% to 95% depending on plant conditions and assuming that the refueling floor is accessible. There are two exceptions where mitigation will be ineffective under the moderate leak scenarios: (1) if the earthquake occurs at the beginning of a refueling outage when the spent fuel is too hot for the assumed mitigation, and (2) if the earthquake occurs when spent fuel is relatively hot and the reactor and spent fuel pool are hydraulically disconnected resulting in insufficient time to deploy mitigation and natural cooling mechanisms cannot prevent fuel damage.

The study's analyses shows that a release from a spent fuel pool accident after the severe earthquake at the reference plant could occur about one time in 10 million years or lower. The factors leading to this low likelihood, as discussed above, are summarized in Figure ES-2.

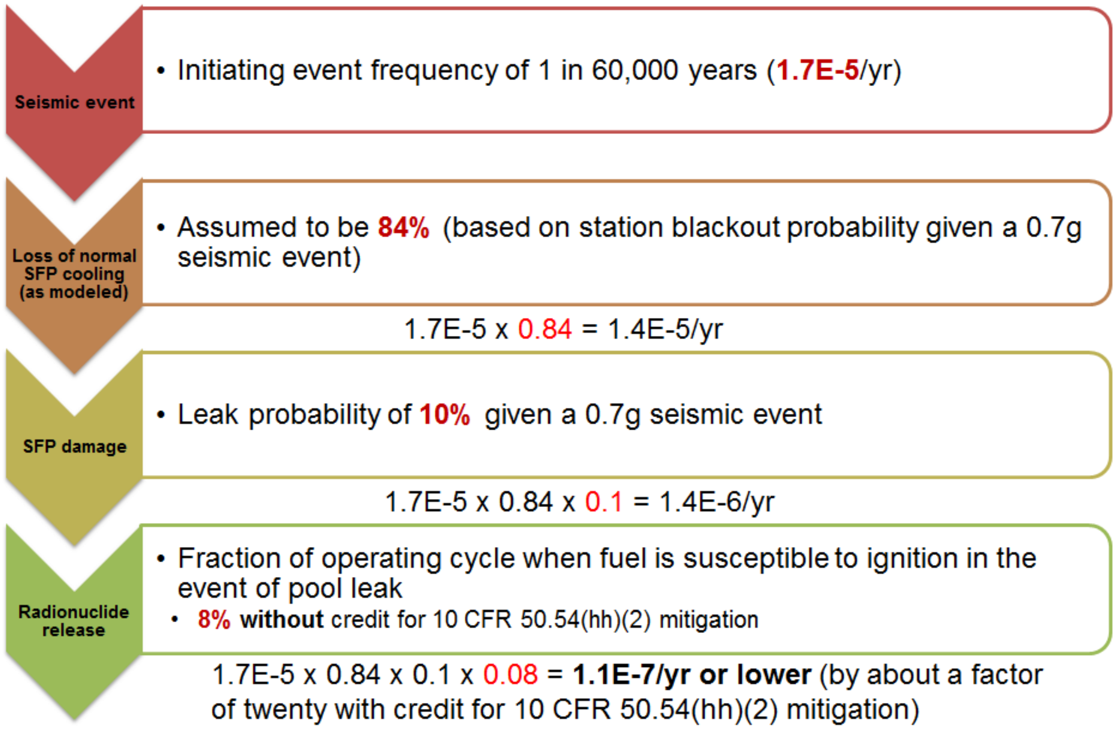

Figure ES-2: Factors Affecting Likelihood of SFP Release from a Severe Seismic Event

The study then estimated consequences to the public of a low likelihood spent fuel pool accident release. The releases of radioactive material are generally comparable to past studies. Despite the fairly large releases for certain predicted accident progressions, consequence analysis of all scenarios indicated zero early fatalities from acute radiation effects because protective actions were modeled to be effective in limiting doses to the public. The study also showed that the risk of an individual dying from cancer from the radioactive release is very low. When including the very low likelihood of a release, the risk in the analyzed scenarios that an average individual within 10 miles receives a fatal latent cancer is between about two in a trillion and five in a hundred billion per year. The risks are similar between different loading or mitigation scenarios because of modeled offsite protective actions that include evacuation, sheltering, relocation, and decontamination. Additionally, these individual risks are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough for the areas to be considered habitable.

In order to do a regulatory analysis to inform whether low density loading should be required at the reference plant, cost estimates of potential protective measures are considered along with other parameters in a cost-benefit analysis. The study shows that, while public health effects from these low likelihood spent fuel pool releases are expected to be very low for all the scenarios studied, offsite protective measures in the form of population relocation and land interdiction may be extensive. High-density loading releases without 10 CFR 50.54(hh)(2) mitigation measures are calculated to result in release frequency-weighted land interdiction values of 0.001 mi^2 per year and 0.5 displaced individuals per year which are arrived at by multiplying the estimated frequency and the estimated consequence. While the amount of land interdiction can be large, the fraction expected to be permanently interdicted is small if a release were to occur. For low-density loading or with successful deployment of 10 CFR 50.54(hh)(2) mitigation measures, considerably less land interdiction and displaced individuals are predicted.

Comparisons of the calculated individual latent cancer fatality (LCF) risk within 10 miles to the NRC Safety Goal are provided in Figure ES-3 to give context that may help the reader to understand the contribution to cancer risks from the accident scenarios that were studied. The NRC Safety Goal for latent cancer fatality risk from nuclear power plant operation (i.e., 2×10^{-6} or two in one million per year) is set 1,000 times lower than the sum of cancer fatality risks resulting from all other causes (i.e., $\sim 2 \times 10^{-3}$ or two in one thousand per year).

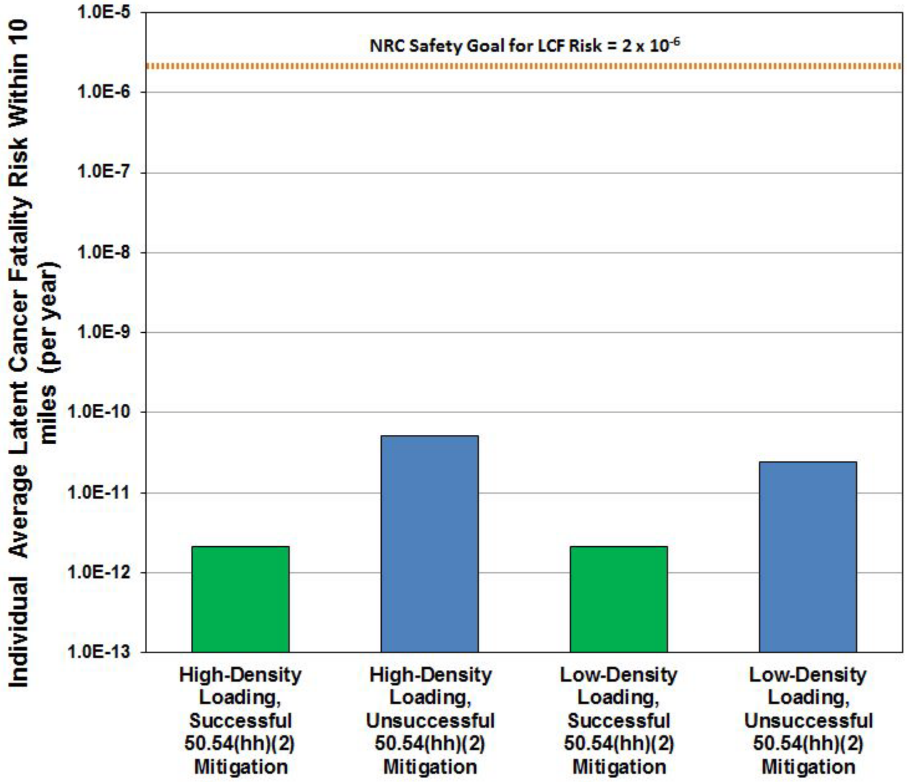

Figure ES-3: Comparison of Population-Weighted Average Individual Latent Cancer Fatality Risk Results for this Study to the NRC Safety Goal (plotted on logarithmic scale)

Comparing the study results to the NRC Safety Goal does involve important limitations. First, the safety goal is intended to encompass all accident scenarios on a nuclear power plant site, including both reactors and spent fuel. This study does not examine all scenarios that would need to be considered in a probabilistic risk assessment for a spent fuel pool, although seismic contributors are considered the most important contributors to spent fuel pool risk. Also, this study represents a mix of limited probabilistic considerations with a deterministic treatment of mitigating features. All analytical techniques, both deterministic and probabilistic, have inherent limitations of scope and method and also have uncertainty of varying degrees and types. As a result, comparison of the scenario-specific calculated individual LCF risk to the NRC Safety Goal is incomplete. However, it is intended to show how multiple spent fuel pool scenarios' risk results in the one in a trillion (10^{-12}) to one in 10 billion (10^{-10}) per year LCF range) are low. While the results of this study are scenario-specific and related to a single spent fuel pool, staff concludes that since these risks are several orders of magnitude smaller than the 2×10^{-6} (two in one million) individual LCF risk that corresponds to the safety goal for latent cancer fatalities, it is unlikely that the results here would contribute significantly to a risk that would challenge the Commission's safety goal policy (NRC, 1986).

In conclusion, past SFP risk studies have shown that high-density spent fuel storage is safe and risk of a release due to an accident is low. This study is consistent with earlier research conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking. This study estimated that the likelihood of a radiological release from the spent fuel pool resulting from the selected severe seismic event analyzed in this study is on the order of one time in 10 million years or lower. For the hypothetical releases studied, no early fatalities attributable to radiation exposure were predicted and individual latent cancer fatality risks are projected to be low, but extensive protective actions may be needed.

The study results demonstrated that in a high-density loading configuration, dispersing hotter fuel throughout the pool or successful mitigation generally prevented or reduced the size of potential releases. Low-density loading reduced the size of potential releases but did not affect the likelihood of a release. When a release is predicted to occur, early and latent fatality risks for individual members of the public do not vary significantly between the scenarios studied because protective actions, including relocation of the public and land interdiction, were modeled to be effective in limiting exposure. The beneficial effects in the reduction of offsite consequences between a high-density loading scenario and a low-density loading scenario are primarily associated with the reduction in the potential extent of land contamination and associated protective actions. The regulatory analysis for this study indicates that expediting movement of spent fuel from the pool does not provide a substantial safety enhancement for the reference plant. The NRC plans to use the insights from this analysis to inform a broader regulatory analysis of the spent fuel pools at all U.S. operating nuclear reactors as part of its Japan Lessons-learned Tier 3 plan. The NRC continues to believe, based on this study and previous studies that spent fuel pools protect public health and safety.

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ABBREVIATIONS AND ACRONYMS

ac	alternating current
ACRS	Advisory Committee on Reactor Safeguards
AEF	annual exceedance frequency
ANL	Argonne National Laboratory
BEIR	biological effects of ionizing radiation
BEF	biological effectiveness factor
Bq	Becquerel
BWR	boiling-water reactor
C	Celsius
CEC	Commission of the European Communities
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
Ci	curies
Cs	cesium
CSCM	continuous surface cap model
CV	control volume
DBE	design basis earthquake
dc	direct current
DDREF	dose and dose rate effectiveness factor
DF	decontamination factor
DLTEVA	delay to evacuation
DLTSHL	delay to shelter
DOE	U.S. Department of Energy
DURBEG	duration of beginning phase
DURMID	duration of middle phase
E	East
EAL	emergency action levels
EAS	emergency alert system
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ESPEED	speed (WinMACCS input variable)
ETE	evacuation time estimate
FAQ	frequently asked questions
FEMA	Federal Emergency Management Agency
FGR	federal guidance report
FSAR	final safety analysis report
GE	General Electric
GEIS	generic environmental impact statement
GI	generic issue
GNF	Global Nuclear Fuel
gpm	gallons per minute
GSI	Generic Safety Issue
GWD	gigawatt-day
HCLPF	high confidence of low probability of failure
HEP	human error probability
hr	hour
HPS	Health Physics Society
HRA	human reliability analysis

I	iodine
ICE	inadvertent criticality event
ICRP	International Commission on Radiological Protection
INL	Idaho National Laboratory
IPEEE	individual plant evaluation for external events
ISFSI	independent spent fuel storage installation
ISRS	in-structure response spectra
K	Kelvin
KI	potassium iodide
LCF	latent cancer fatality
LLNL	Lawrence Livermore National Laboratories
LNT	linear no-threshold
LOOP	loss of offsite power
MACCS2	MELCOR Accident Consequence Code System, Version 2
MCCI	molten core-concrete interaction
MCi	megacuries
MPC	multi-purpose container
MTU	metric tons of uranium
MW	megawatts
MWD	megawatt days
N	North
NCRP	National Council on Radiation Protection and Measurements
NAS	National Academy of Sciences
NRC	Nuclear Regulatory Commission
OCP	operating cycle phase
ORNL	Oak Ridge National Laboratory
ORO	offsite response organization
OSC	operational support center
PAG	protective action guides
PBAPS	Peach Bottom Atomic Power Station
PGA	peak ground acceleration
PPG	pool performance guidelines
PWR	pressurized water reactor
PRA	probabilistic risk assessment
QHO	quantitative health objectives
RB	reactor building
REM	Roentgen Equivalent Man
RHR	residual heat removal
S	South
SAE	site area emergency
SBO	station blackout
SIP	shelter in place
SOARCA	State of the Art Reactor Consequence Analyses
SNL	Sandia National Laboratories
SFP	spent fuel pool
SFPS	Spent Fuel Pool Study
SSC	structures, systems, and components
SSE	safe shutdown earthquake
TSC	technical support center
TSG	technical support guideline
TR	technical report (EPRI technical reports)

USGS	United States Geological Survey
W	West

1. INTRODUCTION AND BACKGROUND

All operating commercial nuclear reactors in the United States are of the light-water reactor design. They utilize upright fuel assemblies (roughly 12 feet in length) with low-enriched uranium oxide fuel (less than 5-percent uranium-235). The fuel assemblies which are composed of numerous fuel rods (typically 80-100 rods for boiling-water reactor fuel and 200–300 rods for pressurized-water reactor fuel) are placed in the reactor for two to three operating cycles. Each operating cycle typically lasts 18 to 24 months. At the end of their "life," the assemblies are placed in large pools of water near the reactor that are roughly 12 meters (m) (40 feet (ft)) deep. For facilities licensed to operate an independent spent fuel storage installation (ISFSI), the fuel assemblies are later loaded into casks and moved to the ISFSI as necessary to accommodate future core offloads. 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 (SFP)) and "dry storage" (to describe storage in casks).

SFPs in the United States were originally designed to store one to two reactor cores worth of spent fuel, so that the fuel could "cool down" (become less thermally and radioactively "hot") before its movement to a reprocessing facility or permanent geological repository. Owing to the abandonment of spent fuel reprocessing as well as delays in the identification, licensing and construction of a repository, U.S. nuclear power plants "re-racked" their SFPs in the 1980s and 1990s to allow for the storage of larger numbers of spent nuclear fuel assemblies (i.e., roughly four reactor cores worth for the plant studied in this study). Throughout this time (including present day), 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. The basis for this position is discussed later in this section.

Stakeholders have periodically challenged the NRC's position that SFPs provide adequate protection of public health and safety. To understand the basis for these challenges, it's first necessary to understand two basic facts about spent nuclear fuel:

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

The list below presents some less-obvious considerations from the perspective of the benefits and disadvantages associated with transitioning from high-density storage to low-density storage. The list is subdivided into two parts—those considerations that are covered within this study and those that are not.

This study includes the following considerations:

- Removal of older fuel from the SFP will decrease the inventory of longer lived radionuclides, such as cesium-137, present in the SFP.

- Removal of older fuel will result in less radioactive material would be present in the pool if a radioactive release occurred, which would be expected to reduce potential offsite consequences.

- Removal of older fuel reduces the overall heat load in the pool while decreasing the amount of metal mass to act as a heat sink should the fuel become uncovered, which can have competing effects on accident timing depending on the type of accident (e.g., a boiloff event versus a complete draindown).

- 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-percent of the total pool volume—because most of the pool is occupied by water, not fuel).[1]

This study does not explicitly address the following considerations, though some are discussed further in APPENDIX B:

- Discharging large amounts of fuel (and thus greatly increasing the amount of fuel contained in the ISFSI) would increase the number of casks required to store the existing spent fuel inventory.

- Expedited discharging of fuel from the SFP to dry storage increases the frequency of postulated cask drops, which in turn increases the frequency of causing damage to the pool or cask 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 into casks that are not currently approved for shipping or long-term storage may require that fuel to be repackaged 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 from various stakeholders. Issues related to the existing dry cask storage infrastructure, worker dose, and economics are discussed in (NAC, 2011) and (EPRI, 2012). Section 1.6 of this report provides more information on each of these studies.

The first set of bulleted considerations is generally advantages associated with expedited fuel movement to casks, while the latter set of bulleted considerations is generally disadvantages. 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. By investigating the pros, we are informing ongoing discussions as to whether fuel movement from spent fuel pools to dry cask storage should be expedited and if any of the "pros" are more

1 The additional water can have a non-intuitive negative impact in certain situations. For a leak at the bottom of the SFP, the additional water at the elevation of the fuel causes it to take longer to "clear" the baseplate (i.e., for the level of the receding water to drop below the bottom of the baseplate). In situations where natural circulation of air under and up through the racks is effective for preventing fuel heatup, this actually temporarily inhibits cooling of the fuel. While this does require a specific set of conditions to be relevant, it is raised here because it does actually arise in one of the scenarios realized later in this report.

compelling than past studies suggest. If they are, then the issue can be addressed more holistically.

1.1 Project Impetus

Various risk studies (most recently NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," issued February 2001) have shown that storage of spent fuel in a high-density configuration in SFPs is safe and that the risk is 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 security assessments after the events of September 11, 2001, 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, the NRC issued a new regulation, 10 CFR 50.54(hh)(2), that requires reactor licensees to develop and implement guidance and strategies intended, in part, to maintain or restore SFP cooling capabilities following certain beyond design basis events.

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, Revision 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants - Draft Report for Comment," issued July 2009. 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 the 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.

One of the objectives of this study is to inform the NRC's Fukushima lesson's learned Tier 3 activity on whether regulatory action needs to be taken to require expedited transfer of spent fuel. NRC analyzes low likelihood (beyond the design basis) events to estimate risk to the public and to explore and ensure safety margins. The results of the study will be used to inform the evaluation of what future regulatory actions the NRC might undertake, including whether expedited transfer of spent fuel from spent fuel pools into dry cask storage is justified. To help inform whether regulatory action needs to be taken in this area, the NRC has prepared an example of a regulatory analysis of the reference plant studied in this report (see APPENDIX D). A regulatory analysis is an analytical tool used by NRC decision-makers to help determine whether the NRC should implement a proposed regulatory action. The regulatory analysis is intended to inform NRC decision makers whether there is a substantial increase in the overall protection of the public health and safety, and whether the direct and indirect costs of implementation are justified in view of a potential substantial increase in protection. For the example regulatory analysis, the Spent Fuel Pool Study (SFPS) results are used as quantitative inputs to the safety goal screening criteria in accordance with the NRC regulatory analysis guidelines (NUREG/BR-0058), wherein the quantitative health objectives are used as a surrogate of the safety goal. The example regulatory analysis also contains estimates of benefits and costs, which are quantified when possible, together with a conclusion as to whether the proposed regulatory action is cost-beneficial. "Cost-beneficial" means that the benefits of the proposed action are equal to, or exceed, the costs of the proposed action. Accident consequences such as land interdiction and population relocation reported in this study are used to estimate the costs resulting from an accident (e.g., costs of interdiction measures, such as decontamination, cleanup, and evacuation) as part of the cost-benefit analysis.

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.

1.2 Technical Approach

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

(1) A condition representative of the following: (i) high-density loading in the SFP using a 1x4 pattern (see Figure 34 for an illustration of what is meant by this terminology), (ii) a relatively full SFP, and (iii) current regulatory requirements relating to fuel configuration and preventive/mitigative capabilities; and

(2) A condition where fuel with more than 5 years of cooling has been moved to dry cask storage (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 (e.g., seismic response, decay heat, radionuclide inventory) is based on readily available information that primarily stemmed from sources such as the study reported in NUREG-1150, “Severe Accident Risks: An Assessment of Five Nuclear Power Plants,” issued December 1990; seismic information developed by the U.S. Geological Survey (USGS); the post-9/11 security assessments[2]; and the State-of-the-Art Reactor Consequence Assessment (SOARCA) described in NUREG-1935. Later in the project, the licensee provided additional information that generally corroborates the assumptions made in this study.

A BWR plant was chosen for this analysis for a mix of reasons including availability of computer models for a BWR plant, a perception of greater external stakeholder interest in elevated (relative to grade) SFPs [3], and the fact that the nuclear reactors that felt the higher tsunami waves and stronger ground motions from the March 11, 2011, Tohoku earthquake, which includes those at Fukushima Daiichi, were all BWR reactors. In the context of a seismic event, 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 BWR design is not intended to suggest that these designs are more vulnerable to SFP accidents. In reality, there are differences between the major design types (PWRs versus BWRs) that make each more or less susceptible to SFP accidents on a scenario-specific basis.

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2 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.

3 SFPs at pressurized water reactor and BWR/6s (which have Mark III containments) are generally at or near grade elevation, with many being partially below grade. In the Mark I and Mark II designs, the SFP is oriented such that the top of the SFP is at the same elevation as the top of the primary containment vessel, which results in them being well above grade.

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 Specificity and Familiarization

This study is intentionally based on plant-specific information for a particular site, as opposed to attempting to define a generic site that might bound a set of plants. This approach was taken because it provides the best context for examining SFP accident progression and release phenomenology in a realistic fashion, for the purpose of providing a better understanding of the factors that affect the characterization of SFP beyond-design basis accidents. The decision to proceed in this manner was deliberately made in reaction to persistent criticisms regarding the realism of past studies (due to their goal of broad applicability in order to support their intended purposes). Because this study strives to be site-specific, it does not account for the variability in design and operation across the operating fleet, but rather, represents one point within that spectrum.

In almost all situations where plant-specific design and operational information is used, it is based on Unit #3 of the Peach Bottom Atomic Power Station (PBAPS), circa 2011. Nevertheless, the SFPS makes some assumptions that are not representative of PBAPS because either (a) insufficient information was available at the time the modeling decision was made or (b) the PBAPS situation was viewed to be atypical. Regarding the former exception, the initial phase of the work was expedited to achieve early insights. Some modeling assumptions were confirmed in parallel to ongoing work, and in instances where newer information provided additional perspective on the modeling assumption, this is noted in the report. Regarding the second exception, the major example of this is that the study assumed the fuel is configured in a 1x4 pattern rather than in the 1x8 pattern used at PBAPS as discussed further in Section 5.1. In some situations, the 1x8 pattern is predicted to have a beneficial effect on the amount of radiation released (Section 9.2). Additionally, sensitivity analyses presented in Chapter 9 explore the effect of some important parameters on the study results. Due to these exceptions, the analysis contained in this report is best described as being performed for a “reference plant” which is largely based on PBAPS.

PBAPS has two General Electric (GE) Type 4 BWRs with Mark I containments, Units #2 and #3. This study uses Unit #3 when unit-specific information is required. Unit #1 is no longer in operation. Units #2 and #3 each have a dedicated SFP, 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, SFP, 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 currently under review (as of January 2013).

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 needed. The site also has an 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) that has State-specific protective action guidelines. Detailed site-specific information relevant for this study is covered in the remainder of this report, including figures that show the reactor building layout, SFP layout, etc.

1.4 Basic Scenario Development

The following key aspects of the way this study is conducted should be mentioned at this point.

- 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 (a low-density loading configuration in the pool) are analyzed. A situation in which 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 (it should be mentioned that BWR fuel is channeled, which reduces the benefit of cross-flow if the pool were to become drained).

- The study focuses on the SFP, not the reactor, though for instances in which the two are hydraulically connected, both are considered to a certain extent.

- In estimating the likelihood and consequences of radiological release, the study does not attempt to quantify the likelihood of successful deployment of mitigation, but rather treats every scenario considering both the case with successful mitigation deployment and the case with unsuccessful mitigation deployment (also referred to as mitigated and unmitigated later in the report)[4]. These results are then used to drive a human reliability analysis (Chapter 8) which provides information about what plant conditions impact mitigative reliability, and what range of likelihoods are expected.

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

1.5 Rationale for Focusing on Consequences of a Seismic Hazard

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

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, by means of 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 include the following:

[4] Note that the shorthand of "mitigated" and "unmitigated" still refers to whether mitigative actions are successfully deployed, not whether the accident itself leads to a release.

- failure modes and effects analysis (except for SSCs specifically discussed in this Chapter)
- data analysis and component reliability (e.g., consideration of random failures)
- effects of dependencies
- HRA as part of the accident progression and recovery; a limited scope HRA is performed in (Chapter 8
- 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 the following:

- 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 to put the results in context.

The inclusion of probabilistic aspects within the current study allows the study to consider some aspects of likelihood, but will not support definitive statements on risk. To elaborate, this study focuses on a specific portion of the overall risk profile, that of large seismic events between 0.5 and 1g. In comparing the results of this study to those of previous studies, one can corroborate or challenge the continued applicability of prior estimates for this piece of the risk profile. Since large seismic events have been shown in the past to be a prominent contributor to risk, this comparison helps to predict whether a comprehensive risk assessment would be expected to result in an overall decrease or increase in the estimated risk. Using this approach, the results of this study can draw supportable, but not definitive, conclusions about overall consequences and risk.

For the present study, because of (1) the relative simplicity of the SFP and its supporting infrastructure as compared to a reactor and its supporting infrastructure and (2) the much lower assembly decay heats, the majority of potential SFP accident risk is believed to emanate from either of the following two events:

(1) events that have the potential to cause a sizable leak in the SFP

(2) events that might preclude operator action to cool or inject water into the pool for an extended period of time (i.e., days)

When one considers the various possible initiators, the first criterion points to the following:

(1) very large (i.e., well beyond the design-basis) seismic events (Note that these events almost certainly initiate a loss-of-offsite power and may fail emergency on-site power.)
(2) heavy load (e.g., cask) drops
(3) inadvertent aircraft crashes

In addition to these, the second criterion also points to the following:

(4) extended loss-of-offsite power (LOOP) events caused by severe weather (e.g., severe storms, hurricanes, tornados), within design-basis seismic events or other grid upsets, with concurrent loss of emergency onsite alternating current (ac) power (either because of the same event or because of coincidental hardware failures)

(5) lack of accessibility caused by a reactor accident that has released radioactive material outside of primary containment (or an accident involving the other SFP)

Note that sabotage events have been excluded from the scope of this study.

Items #(1) (seismically-induced station blackout), #(2) (cask drops), and #(4) (extended 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. The frequency and consequences of a reactor accident is not considered and the effect of a reactor accident on a spent fuel pool scenario is partially considered here, but not rigorously (see Section 2.2 of this study for more information).

Past studies have had generally similar conclusions about the relative contribution to risk from the various initiating events considered. Table 1 summarizes fuel uncovery frequencies from NUREG-1353, “Regulatory Analysis for the Resolution of Generic Issue 82, “Beyond Design Basis Accidents in Spent Fuel Pools,”” issued April 1989, and NUREG-1738. 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	7×10^{-6}	2×10^{-6} (LLNL) 2×10^{-7} (EPRI)[2]
Cask / heavy load drop	3×10^{-8}	2×10^{-7}
LOOP – severe weather	-	1×10^{-7}
LOOP – other	-	3×10^{-8}
Internal fire	-	2×10^{-8}
Loss of pool cooling	6×10^{-8}	1×10^{-8}
Loss of coolant inventory	1×10^{-8}	3×10^{-9}
Inadvertent aircraft impacts	6×10^{-9}	3×10^{-9}
Missiles – general	1×10^{-8}	-
Missiles - tornado	-	$< 1 \times 10^{-9}$
Pneumatic seal failures	3×10^{-8}	-

[1] These numbers have not been multiplied by the stated conditional probability of having a Zirconium fire of 0.25.
[2] NUREG-1738 presented results for the two different seismic hazard models in wide use at the time (the Electric Power Research Institute (EPRI) and Lawrence Livermore National Labs (LLNL) models).

For these reasons, 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 Chapter 3). This range of ground motions represents a good compromise between more likely events that would not be expected to lead to any consequences and less likely events that would lead to greater consequences (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 configuration from an isolated pool to a pool that is hydraulically connected to the reactor vessel (and back again)—these configurations will be referred to as pool-reactor configurations to distinguish from the different spent fuel loading configurations;

- may have spent fuel temporarily offloaded from the reactor;

- will have spent fuel permanently offloaded 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 neutron absorber monitoring)

- may have older spent fuel offloaded into storage casks and transferred to an ISFSI;

- will experience changes in the peak assembly fission product decay power (of interest for draindown events and spray mitigation) because of the above considerations 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 considerations as well as radioactive decay.

To rigorously represent these changing conditions, the study breaks 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 approach allows for more accurate representation of the annualized frequencies of offsite consequences. The specific selection of these phases is described further in Section 5.2 of this report.

1.7 Overview of Past Studies

A number of past studies have been performed to look at various aspects of spent fuel and SFP safety, security, and risk. The major regulatory activities are shown in Figure 1. 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.

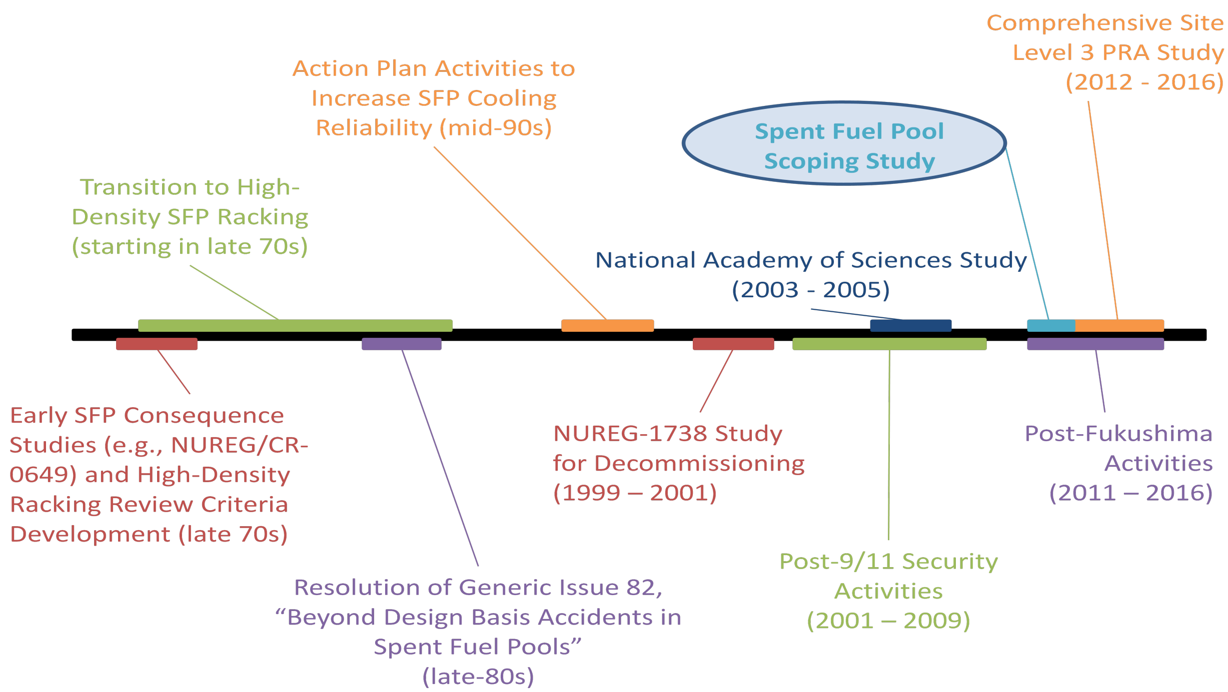

Figure 1 Graphical overview of significant SFP-related activities

In March 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 an SFP and to predict the conditions under which clad failure would occur. The report concluded that the likelihood of clad failure caused by rupture or melting following a complete drainage is extremely dependent on the storage configuration and the spent fuel decay period. Furthermore, 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 (GI)-82, "Beyond Design Basis Accidents in Spent Fuel Pools," culminated in the publishing of several related reports: NUREG/CR-4982, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," issued July 1987 (NRC, 1987), NUREG/CR-5281, "Value/Impact Analysis of Accident Preventive and Mitigative Options for Spent Fuel Pools," issued March 1989 (NRC, 1989a), and NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools,'" (NRC, 1989b). In particular, NUREG/CR-5281 investigated options including limited low-density re-racking of spent fuel, installation of water sprays above the SFP, and installation of redundant cooling, makeup systems, or both. The results of these studies indicated that the measures were, in general, not likely to be cost effective because of the low likelihood of an SFP 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 ensure the safe handling of heavy loads in the vicinity of SFPs, thus reducing the likelihood of the structural failure of the pool and rapid loss of water inventory resulting from a cask drop event.

The latter report (NUREG-1353), which draws from the preceding reports, concludes that if the decay heat level is high enough to heat the fuel rod cladding to about 900 degrees Celsius (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 based on differences in assumed rack geometry. 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 SFP risk is derived from beyond-design-basis earthquakes, this risk is no greater than the risk from core damage accidents caused by seismic events beyond the safe-shutdown earthquake (SSE). Therefore, reducing SFP risk resulting from events beyond the SSE would still leave at least a comparable risk from 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 and issued an Idaho National Laboratories (INL) study entitled, "Loss of Spent Fuel Pool Cooling PRA: Model and Results," (INL, 1996). This study considered a dual-unit plant and the following initiators:

- loss of SFP cooling
- LOOP
- loss of SFP water inventory (did not include heavy load drops)
- loss of primary (reactor) coolant
- seismic events

The results of this study indicated that, for the plant studied, the annual probability of SFP boiling is $5x10^{-5}$ and the annual probability of internal plant flooding associated with SFP accidents is $1x10^{-3}$. Qualitative arguments are provided to show that the likelihood of core damage from SFP boiling accidents is low for most U.S. commercial nuclear power plants. The INL study also showed that, depending on the design characteristics of a given plant, the likelihood of either (1) core damage from SFP-associated flooding or (2) spent fuel damage from pool dryout may not be negligible. Section 6.3.4 further discusses this issue.

The next year, three additional reports were issued: (1) NUREG-1275, Volume 12, "Operating Experience Feedback Report: Assessment of Spent Fuel Cooling," (NRC, 1997a), (2) "Follow-up Activities on the Spent Fuel Pool Action Plan," (NRC, 1997b), and (3) NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants," (NRC, 1997c). The first of these reports 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 one event per 100 reactor years. Loss of SFP cooling with a temperature increase greater than 20 degrees Fahrenheit (F) occurred at a rate of approximately three 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 offloads 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 SFP cooling event early in the outage.

In the second of these reports (known as 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 SFP design issues were sufficiently low that further analyses were not warranted. In one instance in which 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.

The third report (NUREG/CR-6451) presents a regulatory assessment for generic BWR and PWR plants that have permanently ceased operation. In addition to an assessment of regulatory requirements in the context of decommissioning, this study looked at the potential offsite consequences for four phases of decommissioning (hot fuel in the SFP, cold fuel in the SFP, all fuel in dry cask storage, and no spent fuel onsite). The following conclusions are based on an assumption that for the second configuration (cold fuel in the SFP), a zirconium fire would not occur, and that consequences are driven by an accident where a single fuel assembly is dropped during movement within the SFP (akin to the design-basis fuel handling accident). The report concluded that, "Since the estimated consequences of the Configuration 1 representative accident sequence approximate those of a core damage accident, it is recommended that all offsite and onsite emergency planning requirements remain in place during this period, with the exception of the Emergency Response Data System requirements of Part 50, Appendix E. Subject to plant specific confirmation, the offsite emergency preparedness (EP) requirements are expected to be eliminated for Configuration 2, on the basis of a generic boundary dose calculation. Part 50 offsite EP requirements can also be eliminated for Configurations 3 and 4 because the spent fuel has been transferred to an ISFSI (subject to Part 72 requirements) or transported offsite."

Several years later, the NRC re-visited these issues by conducting an SFP 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 (NRC, 2001). The results of the study indicated that the risk at SFPs 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 decision-making on SFPs. However, there are some important conservatisms associated with this study that need to be considered if it is applied outside of its intended context (e.g., exemption requests from NRC requirements for offsite emergency preparedness for decommissioning reactors). These conservatisms include: (1) the use of assumed and often bounding configurations, (2) simplified treatment of the thermal-hydraulic response, (3) conservative assumptions regarding structural response, and (4) emergency preparedness response representative of a decommissioned site.

On the heels of the aforementioned study, the agency also released NUREG/CR-6441 in March 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, 2002a). 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 airflow, and the building ventilation rate. Note that the SFP 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 (owing to its mechanistic treatment of severe accident phenomena), with supporting analysis using the COBRA-SFS, FLOW3D and Fluent codes, along with confirmatory experiments at Sandia National Laboratories (SNL).

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,” dated April 21, 2003 (hereafter referred to as the 2003 Alvarez paper) (Alvarez et al., 2003). In response, the NRC issued a review of the paper (also in 2003) which concluded that the assessment performed of possible SFP 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 relied 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 5 years old be placed in dry casks through an expedited 10-year program costing many billions of dollars, was not justified.

Continued discussions on the issue of SFP safety and security led to a 2004-2005 National Academies study, documented in “Safety and Security of Commercial Spent Nuclear Fuel Storage,” issued in 2006 (NAS, 2006). This study was Congressionally mandated (e.g., see [Congress, 2005]). The National Academies committee was briefed on numerous occasions by the 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., site-specific assessments of licensee response to develop strategies to maintain or restore SFP cooling capabilities).

The NRC's initial response to the study was documented in a letter from the NRC Chairman (Nils Diaz) to Senator Peter Domenici, dated March 14, 2005 (NRC, 2005a). In that response, NRC expressed its appreciation for the insights of the National Academies committee, noting that many of the conclusions mirrored the NRC's conclusions from prior work, which guided NRC initiatives. However, the NRC disagreed with some of the conclusions from the National Academies study, including the finding that the NRC might determine that the earlier movement of spent fuel from pools to dry cask storage would be prudent, depending on the outcome of plant-specific vulnerability analysis. "The Commission views the results of security assessments completed to date as clearly showing that storage of spent fuel in both SFP and in dry storage casks provides reasonable assurance that public health and safety, the environment, and the common defense and security will be adequately protected. The NRC will continue to evaluate the results of the ongoing plant-specific assessments and, based upon new information, would evaluate whether any change to its spent fuel storage policy is warranted." The NRC's position on each finding or recommendation that it disagreed with is contained in the report to Congress that accompanied the March 2005 letter.

In parallel to the National Academies study, the NRC continued performing the aforementioned security assessments, which were completed in 2006-2008. While the results of these studies are not publicly available because of their nature (i.e., containing sensitive information that could be useful to an adversary), the conclusions of the studies were integrated into the NRC's regulatory licensing and oversight processes (e.g., 10 CFR 50.54(hh)(2) as a result of the Power Reactor Security Rulemaking). 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 March 2007 (NRC, 2007). The report's analysis indicates 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 multipurpose 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 1.8×10^{-12} during the first year of service, and 3.2×10^{-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 leak sizes.

Two other documents of regulatory interest were issued in 2008 and 2009. The first was the denial of two PRMs, as documented in SECY-08-0036, "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)," dated March 7, 2008, 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 SFPs.

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 (NRC, 2009). This document reevaluated SFP environmental considerations related to SFPs 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 SFPs (as quantified in NUREG-1738) can be comparable to those from reactor accidents at full power (as estimated in NUREG-1150 (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 SFP accidents; and 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 Daiichi Accident" (NRC, 2011a). This report makes two sets of conclusions and recommendations related to spent fuel pool safety. The first occurs in the section of the report on prolonged loss of ac power. In this section, the task force stated the following:

> The Commission's [station blackout] SBO requirements provide assurance that each nuclear power plant can maintain adequate core cooling and maintain containment integrity for its approved coping period (typically 4 or 8 hours) following an SBO. Also, if available, the equipment used for compliance with 10 CFR 50.54(hh)(2) would provide additional ability to cool either the core or the spent fuel pool and mitigate releases from primary and secondary containment during a prolonged SBO. The implementing guidance for SBO focuses on high winds and heavy snowfalls in assessing potential external causes of loss of offsite power, but does not consider the likelihood of loss of offsite power from other causes such as earthquakes and flooding. Also, the SBO rule does not require the ability to maintain reactor coolant system integrity (i.e., PWR reactor coolant pump seal integrity) or to cool spent fuel….
>
> The Task Force concludes that revising 10 CFR 50.63 to expand the coping capability to include cooling the spent fuel, preventing a loss-of-coolant accident, and preventing containment failure would be a significant benefit.

The task force went on to recommend orders requiring reasonable protection of the equipment provided pursuant to 10 CFR 50.54(hh)(2) and the acquisition of additional sets of equipment as needed to address multiunit events. The task force also recommended a rulemaking

> to revise 10 CFR 50.63 to require each operating and new reactor licensee to (1) establish a minimum coping time of 8 hours for a loss of all ac power, (2) establish the equipment, procedures, and training necessary to implement an "extended loss of all ac" coping time of 72 hours for core and spent fuel pool cooling and for reactor coolant system and primary containment integrity as needed, and (3) preplan and prestage offsite resources to support uninterrupted core and spent fuel pool cooling, and reactor coolant system and containment integrity as needed, including the ability to deliver the equipment to the site in the time period allowed for extended coping, under conditions involving significant degradation of offsite transportation infrastructure associated with significant natural disasters.

The second set of conclusions and recommendations is included in the section of the report on SFP Safety, where the task force concluded the following:

> 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: (1) SFP instrumentation, (2) safety-related ac power for SFP makeup, (3) technical specification revision regarding onsite ac power for SFP makeup and instrumentation, and (4) a seismically-qualified spray capability. The task force also recommended rulemaking or licensing actions (or both) to require the above actions.

The U.S. nuclear industry has also undertaken various studies related to spent fuel storage and transportation. Examples include the following:

- Electric Power Research Institute (EPRI) TR-1003011, "Dry Cask Storage Probabilistic Risk Assessment Scoping Study," issued in 2002
- EPRI TR-1009691, "Probabilistic Risk Assessment (PRA) of Bolted Storage Casks: Updated Quantification and Analysis Report," issued in 2004
- EPRI TR-1021049, "Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling," issued in 2010
- EPRI TR-1025206, "Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling, Revision 1," issued in 2012

The last two reports are of particular interest for the present effort. EPRI TR-1021049 assesses the cost and risk impacts (from a worker dose perspective) associated with transfer of spent nuclear fuel from SFPs to dry storage after 5 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 as a result of the additional handling of spent fuel. With respect to SFP accidents, the report estimates that an additional 711 dry storage packages would have to be handled, as compared to the case without expedited fuel movement, thus increasing the risks associated with cask movement (based on a need to reduce the number of assemblies in some casks when loading more recently-discharged fuel to maintain overall heat load limits). A report prepared by NAC International entitled, "NAC White Paper on Establishing a Balanced Perspective on Wet and Dry Storage of Used Fuel at U.S. Reactors," dated July 7, 2011, makes similar arguments with respect to the impacts of expediting fuel movement (NAC, 2011).

The updated EPRI study, EPRI TR-1025206 (EPRI, 2012), revised the 2010 study to evaluate the dose and cost impacts of accelerating transfer of spent fuel considering two scenarios – one where the campaign takes 10 years and one where it takes 15 years. The report also adds estimates of the reduction in Cs-134 and Cs-137 inventories in the SFP due to accelerated transfer of spent fuel. The updated report estimates the worker doses to be much higher than the 2010 study projected (3 to 4 times higher), while the costs are roughly equivalent. The reduction in Cs-134 and Cs-137 inventories reported range from 43% to 53%. None of the industry studies attempt to calculate offsite consequences associated with postulated SFP accidents, which is a significant difference between those studies and the study documented in this report.

Regarding the amount of fuel older than five years, and its associated decay heat, the table below compares industry averages reported in the NAC study with those from the study presented in this report.

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%	58-90%
5-9	22%	27%	22%	6-22%
10-14	16%	18%	9%	2-8%
15-19	15%	19%	6%	1-7%
20-24	10%	17%	3%	1-4%
25-29	6%	1%	1%	0-1%
30-34	4%	-	<1%	-
Remainder	4%	-	<1%	-

The NAC white paper and the latter (2012) 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 SFP heat load (and thereby the cooling requirements and mitigative time available for beyond-design-basis SFP 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 range of values in this study represent snapshots during the operation cycle). That said, the values from the reference plant (across the representative operating cycle) only strengthen the argument that the SFP heat load is driven by the fuel less than five years old.

1.8 Potential Follow-On Work and Related Activities

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, the following:

- NRC Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (NRC, 2012g)

- NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (NRC, 2012h)

- 10 CFR 50.54 (f) letters to licensees to review seismic hazards (NTTF Recommendation 2.1)

- 10 CFR 50.54 (f) letters to licensees to review onsite shift minimum staffing levels for emergency response and performance of mitigating strategies in accordance with Order EA-12-049 (NTTF Recommendation 9.3)

- SECY 12-0095, Recommendation AR 5 "Expedited Transfer of Spent Fuel from Spent Fuel Pools to Dry Storage"

- an ongoing rulemaking related to security requirements for ISFSIs

- reevaluation of the role of defense-in-depth in regulatory decision-making

- reconsideration of the use of land contamination and economic consequences in the context of regulatory decision-making

- assessment of the effects of seismic events and accident conditions on neutron absorber materials used in SFPs

- performance of a site Level 3 PRA for Vogtle Units #1 and #2 (operating PWRs), including consideration of both wet and dry storage per SECY-11-0089.

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.

1.9 Layout of Remainder of This Report

The remaining sections of this report provide the following information:

- 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
- human reliability analysis
- sensitivity studies to investigate selected assumptions
- comparison of results with past wet and dry storage consequence and risk studies
- summary of backfitting screen analysis

Finally, Appendix A provides details on the emergency response models, Appendix B provides a gap analysis related to the larger question of assessing the impacts of expedited fuel movement, Appendix C provides study-related correspondence with the US NRC’s Advisory Committee on Reactor Safeguards (ACRS) and Appendix D provides a regulatory analysis for expedited transfer at the reference plant.

2. MAJOR ASSUMPTIONS

2.1 Study 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 combined with the fact that the nuclear reactors that felt the higher tsunami waves and stronger ground motions from the March 11, 2011 Tohoku earthquake, which includes those at the Fukushima Daiichi nuclear power plant, were all BWR reactors. Its selection does not denote a belief that this type of design is more vulnerable.
	The **beyond design basis earthquake** is assumed to occur. This is an unlikely event.	The earthquake studied has an estimated frequency of occurrence of one time in 60,000 years. The likelihood of the event is included in the reporting of frequency-weighted consequences.
	The **reliability of mitigation** is not included in the likelihood estimates provided in Chapter 5 through Chapter 7.	The accident progression and consequence analyses were originally conducted without the benefit of a human reliability analysis. The results were then used to frame a human reliability analysis, which is provided in Chapter 8.
	Multi-unit / concurrent reactor accidents are not, in general, considered.	Specifically, the reactor (and its decay heat) is 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 human reliability analysis presented in Chapter 8 does consider multi-unit effects, and Sections 9.3 and 2.2 further discuss this assumption.
	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 1.5 for more information on this assumption.
	Multi-unit aspects are only considered for certain aspects of the study.	See Section 2.2 for more information on this assumption.

Topical Area	Major Assumption	Comment
	Inadvertent criticality events are not considered.	See Section 2.3 for more information on this assumption.
	Other considerations associated with expedited fuel movement.	See APPENDIX B: for a qualitative consideration of fuel movement and APPENDIX D: for quantitative considerations.
Seismic Hazard Characterization	**Vertical PGA** equal to the horizontal PGA and vertical spectral accelerations equal to the horizontal spectral accelerations	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 which indicates that for the seismic bins of interest (high PGA, low likelihood hazard) the contributors to the hazard would be earthquakes with magnitudes less than 6 at about 20 km from the site.
	Seismic hazard models - this study used the existing USGS 2008 model instead of the model in the ongoing program.	A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (NRC, 2012b), it was not available at the start of this scoping study. In addition, the GMPE update is still in progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is appropriate to use for this study because it was the most recent and readily available hazard model for the selected site at the start of the study.

Topical Area	Major Assumption	Comment
Structural and Related Initial Damage State Characterization	**In-structure response spectra (ISRS)** for the study are obtained by scaling ISRS developed for the seismic PRA for PBAPS for the NUREG-1150 study.	Given the differences in the ground motions for the NUREG-1150 PRA and for this study, the use of this scaling is likely to be at or somewhat past the limit of acceptability. The scaling was nevertheless used because it is consistent with the practice of expedited studies of risk or margin. In addition, the assumption (and the uncertainties that it introduces) were deemed to be consistent with the uncertainties in other approximations used in the structural and seismic assessments for the study.
	A **static nonlinear pushover analysis** is used to estimate the overall response of the SFP, concrete strains and cracking, and related liner strains. This analysis used equivalent seismic forces, including hydrodynamic forces, based on elastic ISRS.	As the structure cracks and behaves in a nonlinear manner, it becomes sensitive to frequencies less than its elastic frequencies and dissipates energy, especially if the structure can respond in a non-brittle failure mode. For the ground motions considered in this study, lower frequencies of vibration tends to correspond to lower loads on the structure because spectral accelerations also decrease as the frequencies of vibration decrease. The approach used is, therefore, thought to be conservative for the frequency content of the ground motion expected at the site. Some of this conservatism was accounted for, in part, by: (i) considering a higher damping ratio for the SFP/Reactor-building system (10-percent) than that used for the design basis loads, (ii) calculating equivalent loads using a reduced concrete stiffness, and (iii) by considering a small range of reduction for the calculated liner strains that also accounts for possible ground motion incoherency. A preferred approach to account for this conservatism and also ground motion incoherency effects would involve the sampling of representative ground motion time-histories, both vertical and horizontal, and their use in time-history analyses of the coupled reactor building and SFP structures. This approach, ideally also modeling the small embedment of the reactor building using soil-structure interaction analysis, would more rigorously account for the anticipated conservatisms referred to above but was deemed to be outside the scope of the study.
	Aftershocks are not likely to induce subsequent additional damage to the SFP	The main event would crack the SFP in this study, however it is expected that the SFP's structure would remain stable after the earthquake and resistant to additional loading cycles at this level.

Topical Area	Major Assumption	Comment
	No significant **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 as a direct result of the seismic event itself.
	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 (i.e., do not depend on air pressure, ac power, or dc power) that are unlikely to fail under the earthquake.
	Failure of nearby dams is not explicitly addressed.	The Conowingo Dam is located approximately 9 miles downstream of the site. Failure of that dam could not flood the site. It could lead to additional complications for accident management strategies relying on the river as a water source. The Holtwood Dam is located approximately 6 miles upstream of the site. Failure of this dam, or partial failure in combination with the probable maximum flood, is considered in the plant's Updated Final Safety Analysis Report (UFSAR), in Section 2.4.3.5. Based on the UFSAR, should a complete failure in the upstream dam take place, the rise in the Conowingo Pond level at the site is not expected to exceed grade level, since the pond is about 1 mi wide at the site and the water level would be relieved by the downstream dam.
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.	This assumption is not expected to have a significant effect on the results. See Section 5.2 for more information.

Topical Area	Major Assumption	Comment
	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 (along with the larger volume of water) until the point that water level drops below the fuel transfer canal (and the reactor well and SFP become hydraulically disconnected). That being said, radiological release 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. 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.	This fuel would be placed in 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. See Section 5.2 for additional information.
	Use of a **1x4 pattern**, rather than the 1x8 pattern currently in use at PBAPS.	The 1x8 pattern currently in use at PBAPS is believed to be atypical and is not required by regulation. The timing of obtaining the actual pool configuration, along with modeling conveniences associated with how the MELCOR SFP model is currently designed, also played a role in the decision to use the 1x4 pattern. In cases where the use of a 1x8 pattern might affect study conclusions, this is identified, and Section 9.2 investigates this assumption.
	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 pattern (1x4). Additionally, to get the full benefit of low-density racking, BWR fuel would likely need to have the channel boxes removed.
	Effects on results if a **contiguous storage pattern** were used during the outage.	See Section 9.3 for more information on this assumption.

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. Note that Section 5.4 does include information about dose rates on the refuel floor associated with uncovering a single assembly in the lifted position.
	50.54(hh)(2) **mitigation capacities** (i.e., 500 gpm makeup delivered or 200 gpm spray delivered) are based on the generic NRC-endorsed capacities in NEI-06-12, Revision 2.	For PBAPS, the capacities of the available equipment are somewhat 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
	Mode of mitigation deployment (i.e., use of makeup versus spray)	For OCP 1 and OCP 2 with the "moderate" leakage condition, makeup is deployed. Other, equally reasonable assumptions about mitigation deployment could result in the deployment of sprays instead (which have a potential advantage in terms of mitigation for these conditions). This difference in mode of deployment shows the potential benefit of the additional instrumentation required by NRC Order EA-12-051. A sensitivity study related to this assumption is presented in Section 9.3, for a uniform pattern.
	Use of ac power fragility as a surrogate for **loss of normal SFP cooling** and makeup availability	This study used the ac power fragility from NUREG-1150 of 0.84 as a surrogate for the conditional probability of normal SFP cooling and makeup not being available. This simplifying assumption was made in light of the fact that the study is not a PRA (but rather a consequence analysis with probabilistic considerations) and that this value already approximates the upper bound of 1. In reality, the availability of normal SFP cooling and makeup would be a combination of the AC fragility, the fragility of the actual equipment and its support equipment, and operator actions to recover the equipment, which could result in a conditional probability higher than the value used here.

Topical Area	Major Assumption	Comment
Accident Progression Analysis	The study uses best-estimate **ruthenium release rates** calculated by the MELCOR code. 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 spanning a very large range of uncertainty rather than mechanistic and integrated modeling. Section 6.1.5 of this report provides additional information.
	Radionuclide releases occur only if the fuel has become uncovered by 48 hours and the radiological release has commenced before 72 hours. Otherwise, the study assumes the scenario results in no offsite consequences.	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, the project staff judged 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 of the report for additional discussion on mitigation assumptions in this study and Section 9.8 for time truncation sensitivity.
	The study does not consider **debris** entering the pool as a result of any modeled **hydrogen combustion event**.	Such debris could be generated and could fall into the pool. However, the occurrence of a hydrogen combustion event in this study denotes that the fuel in the SFP 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 combustion event from a concurrent reactor accident has the potential to generate debris which could impair SFP natural circulation air or steam cooling (should the fuel in the SFP become uncovered) for conditions in which the fuel might otherwise be cooled by means of these passive cooling modes. However, this latter situation is inherently tied to the study’s lack of a comprehensive treatment of multiunit aspects.
	Aerosol resuspension inside the reactor building, such as from hydrogen deflagration, will not be significant.	Hydrogen burns in the refueling 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.

Topical Area	Major Assumption	Comment
	The study does not consider the effects of **molten core-concrete interaction (MCCI).**	The MELCOR code models heat transfer from the debris to the pool floor, as well as the fission product release from hot debris. In some cases, the debris temperature remains above typical concrete ablation temperatures (~1500 K). MCCI may occur in selected scenarios in which the fuel relocated to the bottom of the pool following the failure of the rack baseplate and its temperature exceeded the concrete ablation temperature. These cases involve large-scale debris relocation and large releases of volatile fission products. Even without MCCI, the fuel in debris form continues to release fission products resulting in very large releases of volatiles. Section 9.5 of this report presents a sensitivity calculation.
	The effective time dependent **decontamination factor** (DF) of the reactor building can be used to reasonably estimate a cumulative release.	The use of an effective DF is based on a new methodology (see Section 6.1.5 of the report) for SFPs 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, and not just the chemical group, to allow the offsite consequence code to process the source term.
	Criteria for **release of radionuclides in the fuel cladding gap**	MELCOR does not have a fuel cladding deformation and strain model. It uses a value of 900°C for widespread cladding failure. NUREG-1738 cites a temperature range of 700–850°C for rod ballooning and burst; however, the security assessment work mentioned in Section 1.7 showed that rod ballooning has a low impact on the timing to breakaway oxidation and the impact on the peak cladding temperature response was relatively insignificant. In addition, NUREG-1738 assumes 900°C as the temperature at which the onset of significant fission product release is expected. In general, there may be some fuel cladding failures at lower temperatures but MELCOR is mostly concerned with larger thermal release from the fuel. In this sense, the gap release temperature of 900°C is taken as a surrogate for start of rapid fuel heat up associated with breakaway oxidation and initiation of zirconium fire and its propagation.
Offsite Consequence Analysis	Calculated results are from **atmospheric-type releases** only	Atmospheric releases are the primary scope of the project.

Topical Area	Major Assumption	Comment
	A straight-line **Gaussian plume** segment dispersion model is used for the atmospheric transport	The current model is a straight-line transport of plume segments; therefore, it does not capture the effects from changes in wind direction after the plume segment has been released. Despite this, the atmospheric transport model in MACCS2 has compared favorably to Lagrangian particle tracking models [NRC 2004]. This is because the use of ensemble averages of many meteorological conditions, such as the consequences reported in this study, has been shown to make reliable weather-averaged results.
	Distance truncation (from point of release)	Health effect risk estimates (e.g. latent cancer fatality risk and early fatality risk) are with respect to distance. The reported latent cancer fatality risk includes all distances that have doses above the modeled dose limit for habitability, as determined by the Pennsylvania Code Title 25 § 219.51. Total health effect estimates are not a function of distance, and have no distance truncation.
	The effect of **low dose radiation** on latent cancer fatalities is uncertain, and therefore a range of dose truncations are reported.	See Section 7.1.3 for more information on this assumption.
	The **public** will behave in an orderly fashion during a severe accident, and can be represented by cohorts.	See Section 7.1.4 more information on this assumption.
	The seismic event has a limited effect on **emergency response**.	The study assumed that the seismic event would not significantly affect emergency response. This is based on an assessment in NUREG-1935 of the same site and seismic event that assumed the damage to local infrastructure is limited to 12 bridges, partly due to the few large structures in the area. Also, the extended loss of ac power is assumed to be limited to the EPZ (~10 miles) due to the assumption that the strength of the seismic event is from the proximity of the seismic event to the site, rather than being a wider impact from a larger magnitude. See section 7.1.4 for more details.

Topical Area	Major Assumption	Comment
	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 guidance is currently being drafted. In addition, guidance could recommend the development of localized cleanup goals after an accident, to account for sociopolitical, technical, and economic considerations. Given that 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 to decide what land is to be decontaminated. This is consistent with previous studies. See Section 7.1.5 for more information on this assumption.
	A single value for **habitability** is used for all affected areas.	See Section 7.1.5 for more information on decisions regarding land interdiction and associated relocation of the public.

2.2 Multi-Unit Considerations

Observations Regarding a Concurrent Reactor Event:

There are four broad interplays that can be defined between the SFP and the reactor:

(1) an initiating event that directly affects both the reactor and the SFP

(2) a reactor accident that prevents accessibility to the SFP for a prolonged period of time (e.g., due to high radiation fields), leading to a SFP accident

(3) a reactor accident that includes ex-containment energetic events (e.g., a hydrogen combustion event) or other ex-containment interplays (e.g., steaming through the drywell head that affects refuel floor combustible gas mixtures) and creates a hazard to the SFP (e.g., by causing debris to fall in to the pool) or otherwise changes the SFP event progression[5]

(4) an SFP accident that prevents accessibility to key reactor systems and components for a prolonged period of time or 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

5 For instance, a hydrogen combustion event caused by a reactor accident that affects the refuel floor superstructure can lead to additional oxidation (for an otherwise oxygen-limited situation), which in turn may result in higher releases from the SFP. Note that this can also include positive effects, in the sense that steam leaking through the drywell head can serve to steam inert the refueling floor.

pool or excessive condensation from continuous boiling of SFP water), leading to a reactor accident

For each of these interplays, large seismic events and severe weather SBO events are logically the most relevant initiators, as they are the type of initiators that are most likely to initiate an accident at the reactor and SFP, while simultaneously hampering further 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. The study also explores these effects on mitigation (Section 8), and addresses some aspects of the uncertainty associated with this treatment (Section 9). However, explicitly modeling multiunit effects was not a focus of this study, because of the existing limitations with the available computational tools. An ongoing project described in SECY-11-0089 will attempt to more rigorously address these effects in the framework of a multiunit Level 3 PRA for Vogtle Electric Generating Plant Units 1 and 2.

Observations Regarding a Multiunit Event:

Along with the possibility of a concurrent SFP and reactor accident, there is the possibility for a concurrent accident at the SFP of one unit with an accident at the SFP or reactor of the other unit. Again, a large seismic event or a severe weather SBO are the events that are most likely to lead to a multiunit event. In general, if accidents at both SFPs 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 unlikely because the two (or more) SFPs are very unlikely to have the same total pool heat load or peak assembly heat load. (Recall that for multiunit sites, the reactors did not usually start operation at the same time and outages are intentionally staggered.) Even if this symmetry did exist, the offsite consequences would not follow a linear scaling because of a number of nonlinearities associated with that portion of the analysis. Again, capturing these effects was not a focus of this study, and future work (the SECY-11-0089 Level 3 PRA) will attempt to more rigorously treat these effects.

2.3 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. Further, if an ICE were severe enough to produce significant heat, the fuel will be harder to cool and short-lived radionuclides will be produced. 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 reference plant (and other U.S. SFPs), 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 by means of either (1) direct movement of the fuel, (2) direct degradation of the poison material, or (3) indirect effect on either the fuel or poison material because of high temperatures associated with an induced accident, there are several “advantageous” considerations, including the following:

- The reactivity of fresh BWR fuel is suppressed by the high content of burnable absorbers.

- The majority of the fuel in the SFP has low net reactivity since it has gone through more than one operating cycle in the reactor.

- 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).

- 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.

- 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.

- Low-enriched uranium fuel assemblies (which are used in all U.S. reactors) are generally 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.

There are also a few counter considerations:

- 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 subcriticality for high reactivity assemblies).

- The effects of large seismic events on already degraded SFP rack poison material are not easy to quantify.

- The rack panels and poison material have a lower melting temperature than the cladding and fuel.

- Termination of a SFP ICE during an event that required deployment of mitigation equipment could be difficult.

- The possibility of a criticality event cannot be summarily dismissed.

Finally, the offsite 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 reflood. In consideration of all of the above, common accident management practices in the United States call for the use of any available water in responding to fuel uncovery in either the reactor or SFP. This study shows the 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.

3. SEISMIC HAZARD CHARACTERIZATION

3.1 Basis for Probabilistic Estimates

The primary sources of information for seismic hazard estimates at nuclear power plant sites include (1) the NRC/ LLNL (Bernreuter et al., 1989; Sobel, 1994) model (hereafter referred to as the LLNL model); (2) the EPRI model (Toro et al., 1989); and (3) the USGS model developed in the mid-2000s (Peterson et al., 2008) (hereafter referred to as the USGS 2008 model. The implementation of the individual plant evaluation for external events (IPEEE) program utilized both the LLNL and EPRI models (NRC, 2002b). The National Seismic Hazard Mapping Project utilized the USGS 2008 model. The NRC also utilized the USGS 2008 model for the seismic hazards estimates used in screening level assessments for Generic Issue 199 (GI-199) (NRC, 2012a).

The seismic hazard assessment in this study is the US Geological Survey (USGS, 2008) hazard model. A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (NRC, 2012b), it was not available at the start of this scoping study. In addition, the GMPE update is still in progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is appropriate to use for this study because it was the most recent and readily available hazard model for the selected site at the start of the study.

Figure 2 (PGA) and Figure 3 (1-, 5-, and 10 Hz spectral acceleration) graphically show comparisons of hazard estimates for the reference site (a rock site) with the three information sources listed above. These comparisons are provided to compare the model used in this study to well-known and extensively documented information sources (LLNL model and EPRI model) that were used in past SFP risk studies. The comparisons support the following observations:

- For the PGA, the USGS 2008 model predicts higher annual probability of occurrence for high-level, low-probability events, specifically for events with PGAs greater than about 0.35g.

- For moderate PGAs, from about 0.1g to 0.35g, the LLNL model is higher than the USGS 2008 model. For events above about 0.35g, which are lower probability events, the USGS 2008 model is higher than the LLNL model until both hazard estimates differ by factors of about 2.5 at 0.75g and about 3 at 1.0g.

- The EPRI model hazard estimates are lower than those from the USGS 2008 model for all PGAs. Specifically, hazard estimates based on the USGS 2008 model are about 2 times greater at 0.2g with the difference increasing to about 10 times at 1.0g.

- Thus, in terms of PGA, the seismic hazard estimates used for this study are about 2.5 times greater than LLNL model estimates and about 6 times greater than EPRI model estimates at 0.75g.

- Curves for the USGS 2008 model and the LLNL model are comparable for each representation, with the USGS 2008 model sometimes being higher (higher annual probability of occurrence) and the LLNL model 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 this is the fact that the 5-Hz LLNL model curve, which is higher than the 10-Hz curve.

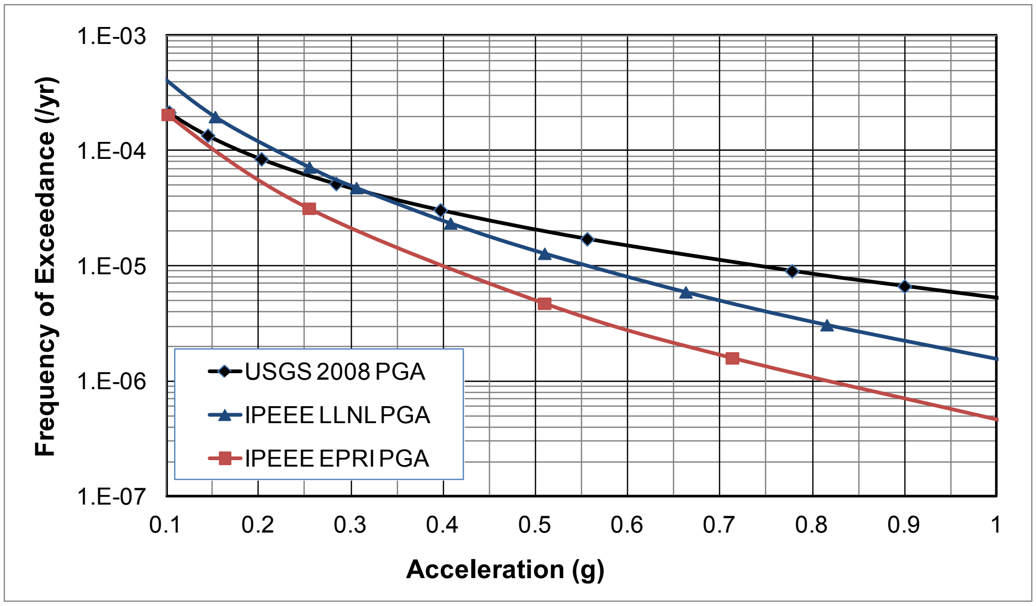

Figure 2 Comparison of PGA exceedance frequencies at the reference plant

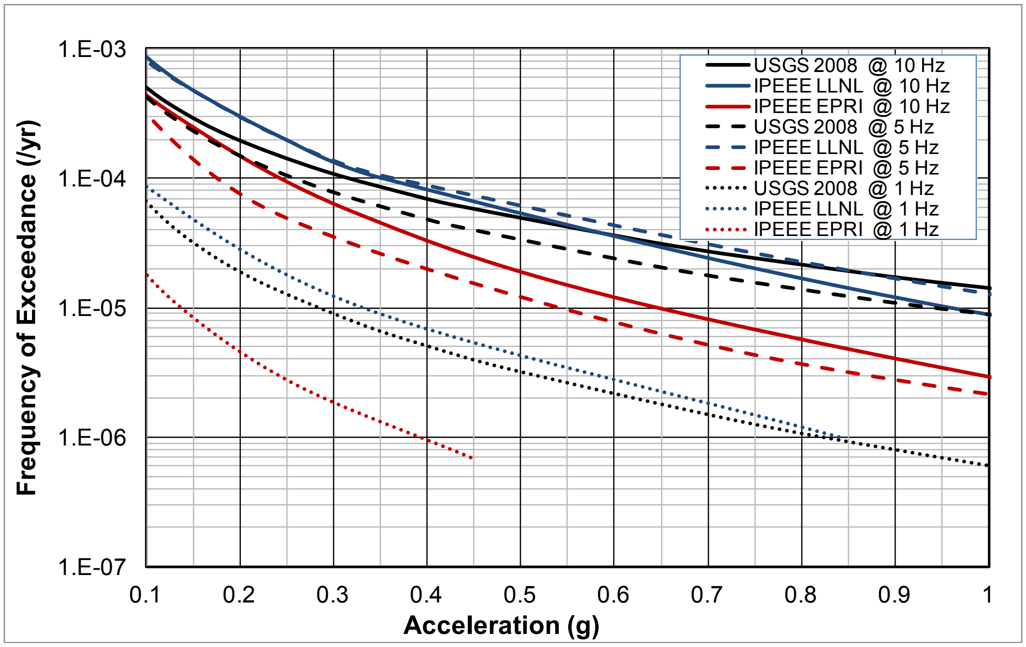

Figure 3 Comparison of spectral exceedance frequencies at the reference plant (rock hazard curves)

A comparison of the annual frequency of exceeding a given PGA for all Mark I sites (Figure 4) shows that the reference plant 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 5), the reference site is in the upper half of the group.

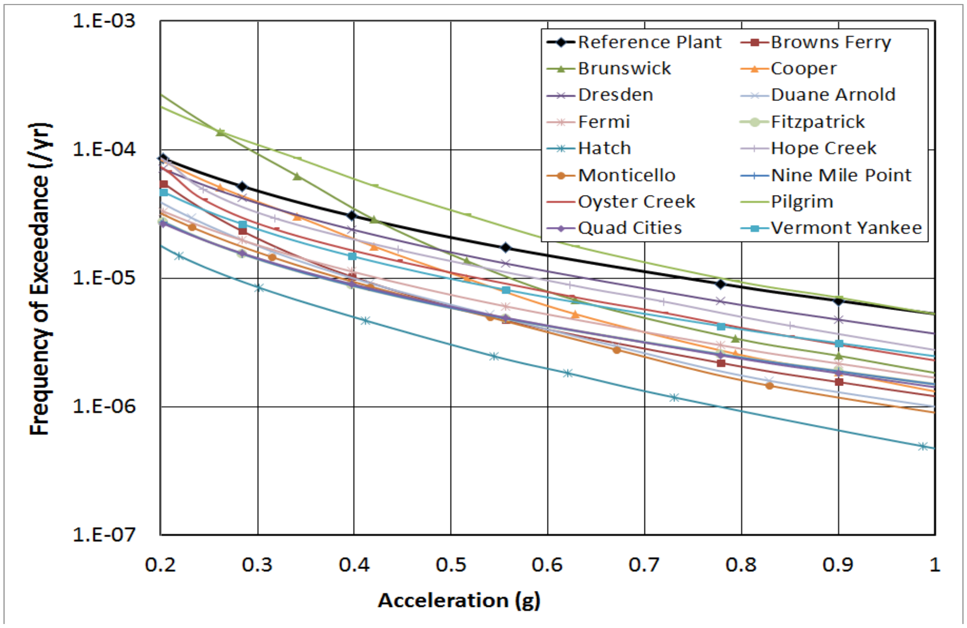

Figure 4 Comparison of annual PGA exceedance frequencies for U.S. Mark I reactors (USGS 2008 model) (rock hazard curves)

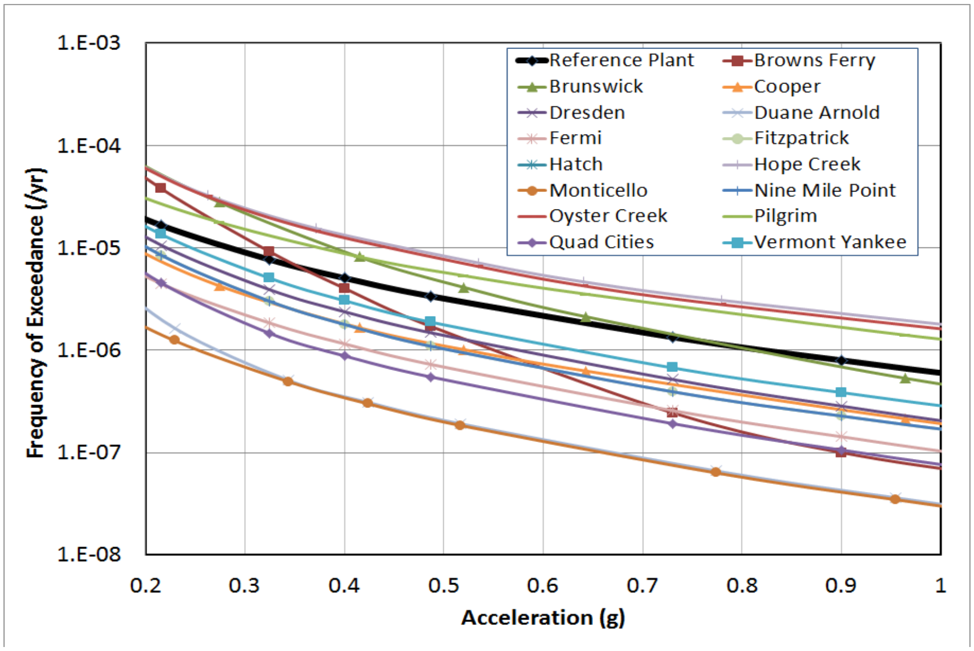

Figure 5 Comparison of annual exceedance frequencies for 1 Hz spectral accelerations for U.S. Mark I reactors (USGS 2008 model) (rock hazard curves)

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). Revision 1.01 of the NRC handbook entitled, "Risk Assessment of Operational Events, Volume 2—External Events," issued January 2008 (NRC, 2008b), recommends the use of at least three bins unless plant-specific considerations require more bins. The SFPS used four bins.

Table 4 shows the resulting bins, along with the tabulated frequencies for various spectral and peak accelerations. Note that for bin 4, the representative bin PGA has been set to 1.2g by convention, whereas for the other bins, it is the geometric mean of the interval endpoints. Figure 6 shows these results graphically.

Table 4 Seismic Bins and Initiating Event Frequencies

Bin #	Bin Range (g)	Bin PGA (g)	Approximate Initiating Event Frequency (USGS 2008 model) (/yr)
1	0.05 - 0.3	0.12	$5.2x10^{-4}$
2	0.3 - 0.5	0.4	$2.7x10^{-5}$
3	0.5 - 1.0	0.7	$1.7x10^{-5}$
4	> 1.0	1.2[1]	$4.9x10^{-6}$

[1] Assumed based on PRA modeling convention

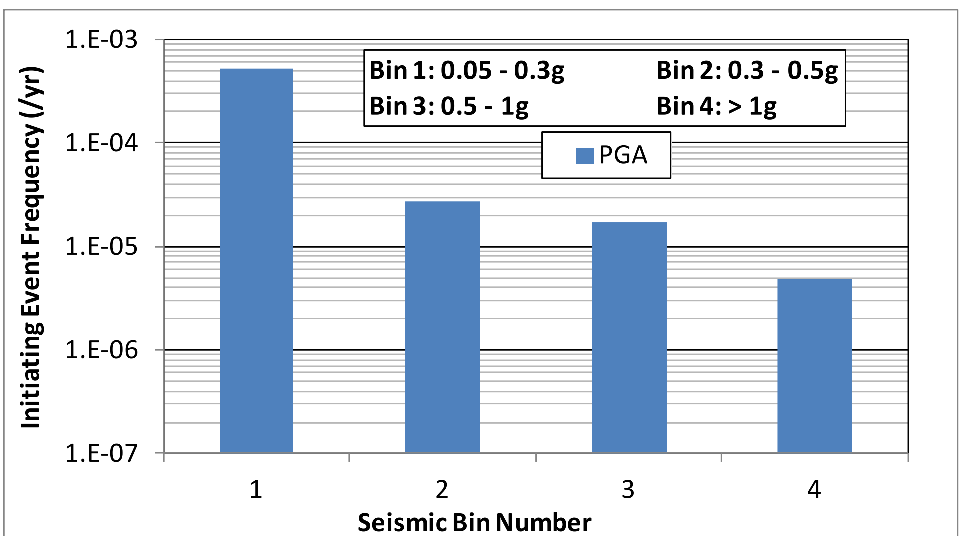

Figure 6 Comparison of seismic initiating event frequencies

Based on this information, and on a review of the results of past studies which indicate that damage to the SFP and other relevant structures, systems, and components (SSCs) is not credible for events in bins 1 and 2, this study focused on bin 3. The project staff concluded 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 (NRC, 2001)) indicates that events in bin 3, with initiating annual frequencies on the order of 1×10^{-5} to 2×10^{-5}, could challenge the integrity of the SFP (i.e., of causing a leak) at the reference plant. Thus, this is the initiating earthquake chosen for this study. It is an event that is no more severe than events considered in past reactor and SFP PRAs and consequence studies.

This study therefore considers a challenging, but very low probability earthquake as the initiating event, selected based on the considerations indicated above. This decision translates into a seismic event with a PGA several times greater than that associated with the design-basis earthquake, currently called the safe-shutdown earthquake or SSE. The PGA for the reference plant SSE is 0.12g. (This is about a magnitude 5.3 earthquake at about 25 kilometers (km).) While the probability of occurrence of this earthquake was not used in its selection, 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 (NRC, 2001; Prassinos et al., 1989) and engineering judgment, was that the ground motions associated with the SSE (bin 1) would not be large enough to damage the SFP at the reference plant.

The information above coupled with the review of previous studies (NRC, 2001) suggests that the frequency of a seismic event that could challenge the integrity of the SFP at the reference plant is on the order of 1.7×10^{-5} per year (i.e., approximately one event in 60,000 years) or less. Table 5 contrasts this frequency against other sources of information. The Mineral, VA, earthquake of August 23, 2011, which occurred near the North Anna nuclear power plant, can serve as a point of reference. In that case, the NRC staff concluded, using data from USGS instruments, that the PGA at the North Anna site was about 0.26g (NRC, 2011b). Using the USGS 2008 information for North Anna, the hazard frequency for an event with this PGA is about 1.2×10^{-4} per year (one event every 8,300 years). This frequency places the Mineral, VA, event in bin 1.

Table 5 Comparison of Seismic Frequencies from Various Sources

Source	Estimated initiating event frequency of a large seismic event[1]	Notes
USGS 2008	1.7×10^{-5}/yr (one event in 60,000 years)	Frequency of seismic bin 3 of 4 (0.5 to 1g)
The reference plant's SPAR-EE Model (v3.21, Rev. 1)	1.3×10^{-5}/yr (one event in 77,000 years)	Frequency of seismic bin 3 of 3 (> 0.5g)
NUREG-1738[1]	1.1×10^{-5}/yr (one event in 90,000 years)	Frequency of seismic hazard between 0.51g to 1.02g

[1] Initiating event frequencies reported are those based on the LLNL models (Sobel, 1994).

In addition to the PGA, ground motions at a site are also characterized by their frequency content expressed in terms of response spectra. Section 3.3 describes the procedure used to develop the horizontal and vertical acceleration response spectra for the input ground motion for this study.

Other response spectra of interest for this study are (1) the plant's SSE response spectra and (2) the free-field response spectra used in the seismic PRA for the NUREG-1150 study. 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. Volume 1, Part 3, of NUREG/CR-4550, "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events," issued December 1990 (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. As shown in Section 3.3, the spectral shape for this study differs from the SSE response spectrum, as well as from the median response spectra considered in the NUREG-1150 seismic PRA. Frequency content for the SSE and the NUREG-1150 PRA spectra generally resemble each other.

3.3 Characterization of the Ground Motion Response Spectra

Spectral shapes developed for the safety/risk assessment results for the GI-199, which utilized the USGS (2008) model, were used to develop the free-field acceleration response spectra for this study. The free-field acceleration response spectrum developed for the GI-199 for this site has a zero-period spectral acceleration (PGA) of about 0.34g. The acceleration response spectra for the free-field ground motion for the initiating seismic event considered for this study (bin 3 in Table 4 and a PGA of 0.7 g) were derived from the GI-199 spectra shape as follows:

- Horizontal shaking: horizontal response spectrum is the GI-199 spectral shape scaled to the bin 3 PGA (zero-period acceleration) of about 0.7g (specifically 0.71g). While it is recognized that the frequency content of ground motions may change somewhat with increasing PGA levels, scaling of the spectral shape for the 0.34g PGA to the bin 3 PGA is considered a reasonable approximation for low probability hazard for this rock site and for the purposes of this study. Figure 7 compares the horizontal input acceleration response spectra for this study to the horizontal response spectrum for the plant's Safe Shutdown Earthquake (SSE) (PGA of 0.12g) for 5-percent damping.

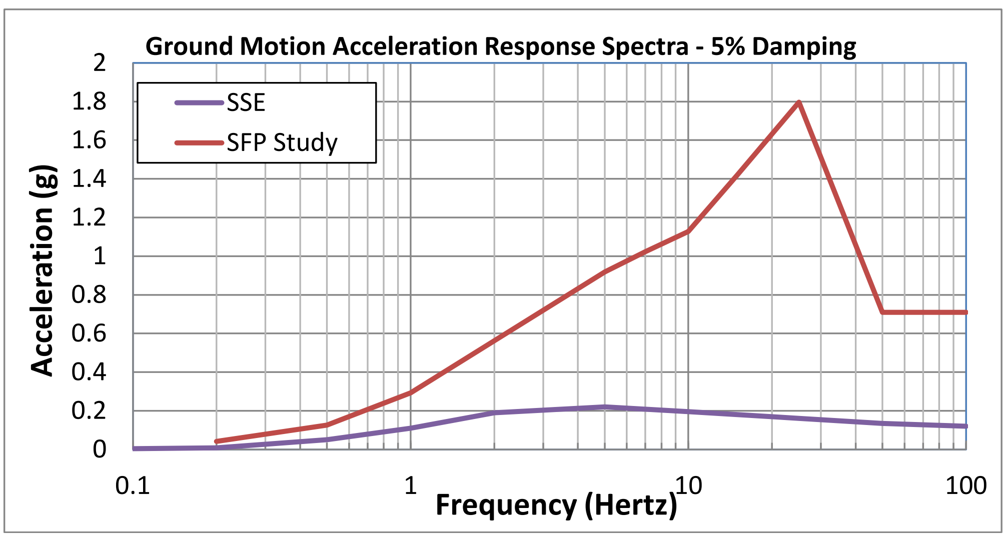

Figure 7 Input acceleration response spectrum and SSE (Horizontal Ground Motion)

- Vertical shaking: vertical spectral accelerations and the vertical PGA (0.7 g) are assumed to be the same as the horizontal spectral accelerations and PGA. 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 which indicates that for the seismic bins of interest (high PGA, low likelihood events) the contributors to the hazard would be earthquakes with magnitudes less than 6 at about 20 km from the site.

Other response spectra of interest for this study are the free-field response spectra used in the seismic PRA for the NUREG-1150 study (Lambright et al., 1990). These spectra are of interest because in-structure response spectra calculated for that ground motion were scaled, in approximation, to estimate in-structure response spectra for the free-field ground motion considered for this study. Figure 8 compares the frequency content of the horizontal response spectra (5-percent damping) for the SSE, the median response spectrum used in the NUREG-1150 study, and the spectral shape used in Spent Fuel Pool Study. For this comparison, all spectra are scaled to a PGA of 1.0g. When the three response spectra under consideration are scaled to the same PGA, the information in Figure 8 supports the following observations:

- For frequencies between about 10 Hz and 45 Hz, the spectral shape used in this study 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, which is generally the frequency range most damaging for nuclear power plant structures, the spectral shape used in this study has lower spectral accelerations than the ground shaking considered for the SSE and the NUREG-1150 study.

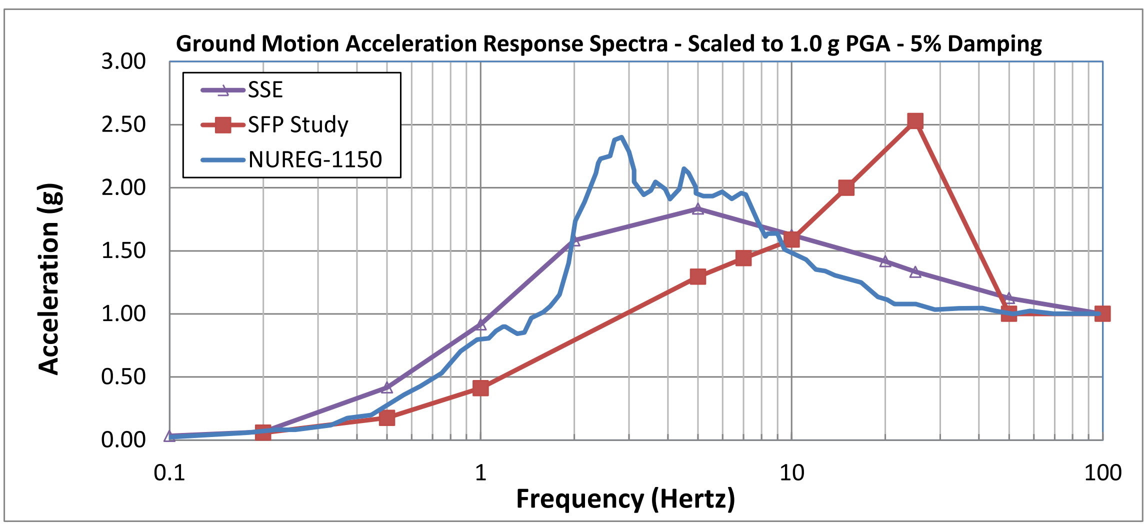

Figure 8 Response spectrum for 5-percent damping scaled to 1.0 g PGA: SSE, NUREG/CR-4550 (NUREG-1150 PRA), and this study (GI-199)

As noted above, the input horizontal acceleration response spectrum for the event considered in this study is the spectral shape derived for the GI-199 study using the USGS 2008 model (PGA of about 0.34g) scaled to a PGA of about 0.7g. Figure 9 shows the horizontal response spectra (5-percent damping) for the event considered in this study, for the SSE (0.12g PGA), and for the response spectrum used in the NUREG-1150 PRA. The NUREG-1150 response spectra shown in the figure is scaled from a PGA of 0.3g to the PGA of the event for this study. 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 in Table 4. They are also richer at the high frequencies (greater than 10 Hz), which generally tend to be less challenging to nuclear power plant structures.

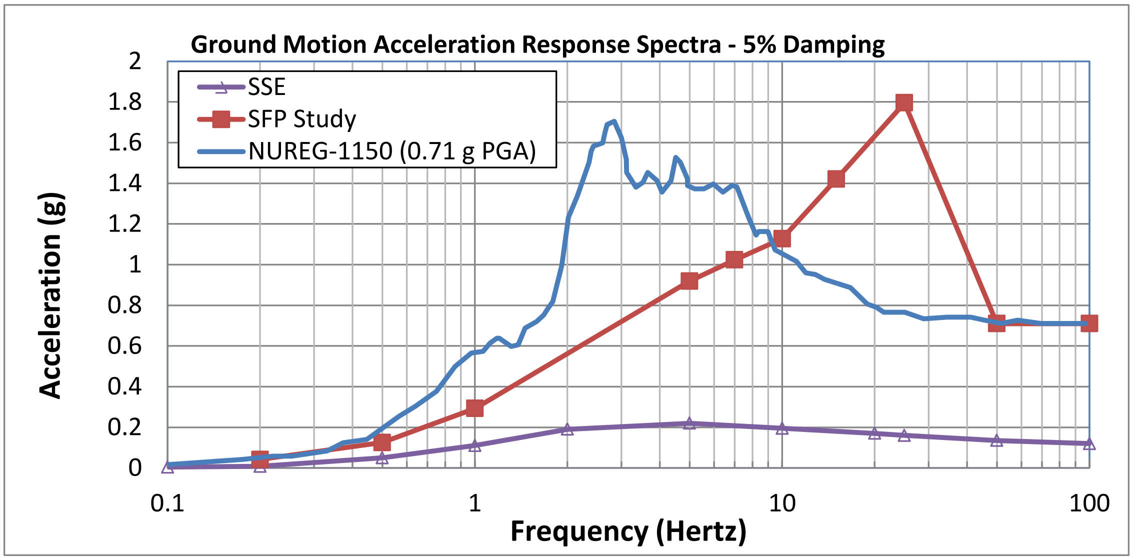

Figure 9 Horizontal response spectrum (5-percent damping): SSE, SFP Study and NUREG-1150 PRA (for 0.71g PGA)

4. STRUCTURAL ANALYSIS AND RELATED INITIAL DAMAGE CHARACTERIZATION

This section documents the structural analysis performed to estimate the initial damage states for the accident progression analysis. It provides:

- the objective,
- the approach including assumptions,
- the structural modeling and analyses, and
- the 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 that would constitute the initial conditions for the accident progression analysis. Structural and related damage states have been divided into the following two major categories:

(1) structural damage to the spent fuel pool structure with potential locations of leakage from concrete cracking and related liner tearing

(2) other damage states including:

- amount of water, if any, displaced by sloshing of the water out of the SFP
- damage to the refuel gate, support systems and penetrations, as well as 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 pool structure, namely concrete distortions, concrete cracking, and metallic liner strains at the bottom of the pool. The focus on this analysis was based on the review of past studies which indicates that damage to the SFP in those locations, if it were to occur, would be the more significant damage state in terms of loss of coolant.

Section 4.3 provides a review of the performance of SFPs at four nuclear power plant sites with a total of 20 reactors under two major recent earthquakes in Japan. This review compares relevant aspects of the seismic scenario and estimated damage states for this study with known aspects of the seismic scenario and performance of SFPs affected by those earthquakes. The review summarizes known or presumed reductions in water levels of the SFPs affected by those earthquakes associated with either water leakage from structural damage, if any, or water loss from sloshing. Although these reviews and comparisons use information available at the time of the execution of this study, they assist in the interpretation of the results obtained for the seismic scenario and SFP considered.

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 the approach reported in NUREG/CR-5176, "Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Power Plants," issued January 1989 (Prassinos et al., 1989) modified to address specific needs of this study. (The general approach is fully described in the following section.) The analyses reported in NUREG/CR-5176 were conducted in conjunction with research activities related to Generic Issue 82 (GI-82) (NRC, 2012c). Appendix 2 to NUREG-1738 (NRC, 2001), a technical study of spent fuel pool accident risk at decommissioning nuclear power plants, also addresses the seismic fragility of spent fuel pools and refers to the results and approach used in NUREG/CR-5176. The seismic evaluations in NUREG/CR-5176 considered ground motions with frequency content that differs from those considered in this study. Specifically, NUREG/CR-5176 considered ground motions with maximum response spectra amplitudes for frequencies below 10 Hz while the ground motions considered for this study have maximum response spectra amplitudes for frequencies greater than 10 Hz. This difference in the characteristics of the ground motions tends to induce conservatism in the approach when applied to this study as indicated below.

Approach

The overall approach used 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 nine steps:

(1) Obtain free-field acceleration response spectra (horizontal and vertical) for the site considered (a rock site and a reactor building with small embedment) as indicated in Section 3.3 and shown in Figure 7.

(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 SFP (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.) The SFP Study used the median-centered ISRS calculated for Peach Bottom 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 et al., 1990).

(3) Estimate ISRS for the ground motions of interest for this study at the elevations of interest (Step 2) by scaling the ISRS from previous studies (Step 2). The scaling accounts for differences in the response spectra for the NUREG-1150 seismic PRA and for this study. Given the differences in the ground motions for the NUREG-1150 PRA and for this study, the use of this scaling is likely to be near the limit of acceptability. Use of this scaling is justified on the basis that the approximations and uncertainties introduced are consistent with the uncertainties in other approximations used in the structural and seismic assessments for the study. It is expected that efforts by the NRC and industry related to Requests for Information in SECY-12-0025 (NRC, 2012f) associated with the Near Term Task Force (NTTF) Recommendation 2.1 (NRC, 2011a) will result in updated staff guidance on ISRS scaling.

(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 (1) peak vertical and horizontal accelerations of the floor and walls of the SFP structure (seismic coefficients), (2) peak vertical and horizontal hydrodynamic impulsive pressures on the floor and walls of the SFP from the water in the pool and (3) 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) Use a simplified three-dimensional (3D) finite element model of the SFP structure to estimate or verify these loads. Specifically, use elastic solid elements and special fluid elements to model the water to estimate natural frequencies and mode shapes for the SFP structure. Use this model to calculate the spatial distribution of peak vertical and horizontal accelerations of the structural components using the ISRS from Step 3 as input.

(b) Calculate hydrodynamic impulsive pressures and peak vertical and horizontal pressures on the basis of simplified methods (Housner, 1963; AEC, 1963; Malhotra et al., 2000). Use the 3D finite element model in Step 4a together with the ISRS from Step 3 as input to estimate peak hydrodynamic pressures. This provides for verification and adjustment of the hydrodynamic pressures calculated using simplified methods.

(c) Use the 3D finite element model in Step 4a together with the ISRS from Step 3 as input 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. Such analysis provides the load deformation behavior of the SFP for the loading pattern and intensity considered. 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, 2006a). 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) Based on guidance for combining effects from three spatial components of an earthquake in Regulatory Guide 1.92 (NRC, 2006a), peak vertical seismic loads were combined with 40-percent of the peak horizontal loads. A combination of peak horizontal loads on both directions with 40-percent of the vertical loads was also considered. Preliminary analyses indicated that the load combination involving peak vertical loads and 40-percent of the horizontal loads would likely 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 median 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 beyond-design-basis 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 events (NEI, 2011; Prassinos et al., 1989).

(6) Review and process the calculated structural distortions (as measured by the displacement of nearby nodes), structural deformations, concrete strains and liner strains for the following purposes:

(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 with 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 such that water leakage is controlled by the size of the tearing in the liner.

(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 through 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 as well as 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. The use of finite element analyses in Step 4 is done 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 and adjust hydrodynamic impulsive loads on the floor and walls of the SFP. The use of finite element analyses in Step 5 is done to track the development of cracking and liner strains and then relate 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 SFP structure. These conservative aspects are as follows:

- As the high-frequency structure of the SFP (fundamental frequency on the order of 15 to 25 Hz) cracks under the applied seismic loads, its natural frequencies decrease and are no longer resonant with the high frequency components of the ground motion (i.e., the frequencies corresponding to the higher spectral accelerations). Since the spectral accelerations decrease as the frequencies of the SFP structure decrease after cracking, the use of seismic loads calculated assuming elastic frequencies can introduce conservatism in the analysis for the seismic event considered. This would be the case if the SFP structure were to remain stable with only minor distortions after cracking as in the case of the SFP studied. This aspect was partially accounted for in this study through a small reduction in the spectral accelerations and by the use of a small reduction of the concrete stiffness in the calculation of the natural frequencies of the SFP structure. Assessment of the conservatism introduced by the approach used, which was outside the scope of this study, would involve the sampling of representative acceleration time-histories, both vertical and horizontal, and their use in time-history analyses of the SFP response to the seismic loads considered.
- Generally ISRS accelerations do not 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, both the structure of the reactor building and of the SFP may crack and dissipate energy thus dampening the response of the building. This effect is taken into account, in part, by reducing ISRS spectral accelerations by the ratio of spectral amplification factors for 10-percent and 5-percent damping (Newmark and Hall, 1978). The reduction of spectral accelerations implied by the use of a higher damping ratio is further justified by the decrease in the fundamental frequency of the structure related to cracking which, for the input free-field ground motion for this study (see Section 3.3 and Figure 9), would decrease the spectral accelerations.
- 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 (1) the scaling of ISRS calculated from a ground motion with response spectra markedly different from the ground motion spectra considered for this study (addressed in item 5(c)), (2) the decoupling of the response of the SFP from the response of the reactor building, and (3) neglecting the small embedment of the reactor building, as also done in previous studies, which may affect the calculation of horizontal ISRS. Follow-on subsections address these and the above approximations. 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 a better assessment of these possible conservatisms. However, these were outside 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 explicitly in the estimation of the initial stresses in the SFP components (Step 5a). The weight of those equipment and appurtenances is expected to be small in comparison to the other dead loads in the pool and accounted for by the approximations in the estimation of those dead loads. 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.

Seismic Loads

Chapter 3 of this report discusses the bases for the free-field ground motion response spectra for the seismic event considered for the SFP Study. As noted in Chapter 3 other free-field response spectra of interest in this study are those documented in NUREG/CR-4550 (Lambright et al., 1990) and used in the probabilistic risk assessment (PRA) for NUREG-1150. That report 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 motion 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 et al. (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 natural 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 dominant, elastic (uncracked) frequencies of vibration of the SFP structure, considering hydrodynamic effects of the water and the mass of the spent fuel, range from about 17 Hz (vertical response of the floor slab) to 28 Hz (horizontal deformations of the walls). 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 Natural Frequencies of Vibration for the Peach Bottom Reactor Building (Lambright et al., 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 et al. (1990) were scaled to estimate floor vertical and horizontal ISRS at the elevation at the bottom of the SFP (Elevation 195 ft) as well as horizontal ISRS at the midheight 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 study. This scaling was done using the reported median-centered

ISRS for 5-percent damping, the vertical and horizontal (EW) components for the ISRS (examination of the charts indicates that the horizontal (EW) component tends to have the higher spectral accelerations). The SFP 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 reductions in the high frequency spectral accelerations is provided at the end of this subsection.

Figure 10 shows a comparison of the vertical ISRS for the elevation at the bottom of the SFP, calculated as indicated above to the corresponding ISRS (smoothed) for the NUREG-1150 seismic PRA. Likewise, Figure 11 provides a similar comparison for the horizontal ISRS at the midheight of the SFP structure (average of the ISRS at Elevation 195 ft and Elevation 234 ft).

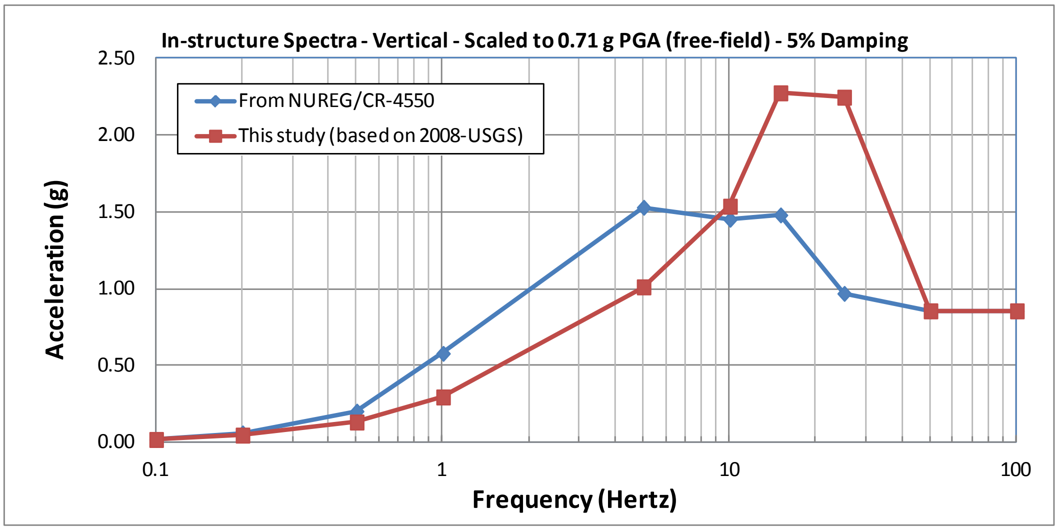

Figure 10 Vertical ISRS for 5-percent damping at Elevation 195 ft (bottom of the SFP)

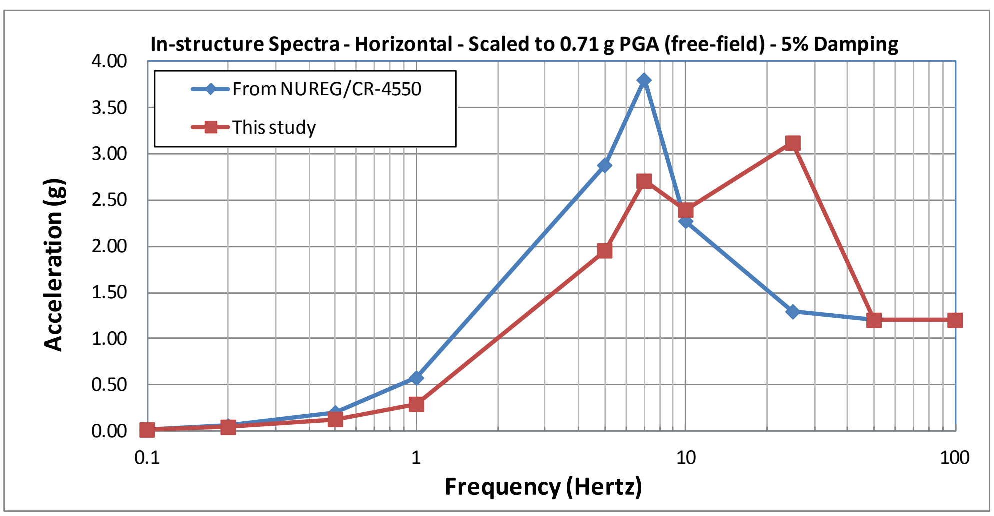

Figure 11 Horizontal ISRS for 5-percent damping midway between Elevation 195 ft and Elevation 234 ft (midheight of the SFP)

The spectra shown in Figure 10 and Figure 11 are for 5-percent damping (reactor building and equipment). Calculation of seismic load coefficients for the SFP floors and walls as well as of hydrodynamic impulsive pressures considered a reduction of these spectral accelerations. Specifically, seismic coefficients and hydrodynamic pressures calculated using the 5-percent damping ISRS were reduced by the ratio of scaling factors for 10-percent and 5-percent damping reported in NUREG/CR-0098 (Newmark and Hall, 1978). As noted above, this is done to account for, in part, the energy dissipation (damping) from cracking of the SFP and minor cracking of the reactor building. This is further justified by the reduction in the natural frequency of the SFP structure from cracking that would lead to reduced spectral accelerations for the input free-field ground motion (see Section 3.3 and Figure 9). An assumption is, for example, 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). This will absorb and dissipate energy and damp the response. ISRS obtained by reducing the 5-percent damping ISRS in this manner, herein called 10-percent damping ISRS, are shown in Figure 12 and Figure 13.

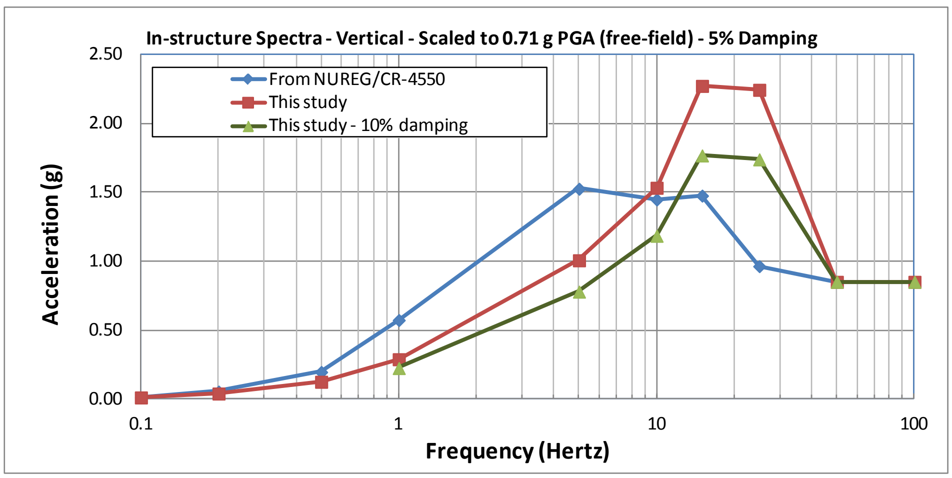

Figure 12 Vertical ISRS for 5-percent and 10-percent damping at Elevation 195 ft (bottom of the SFP)

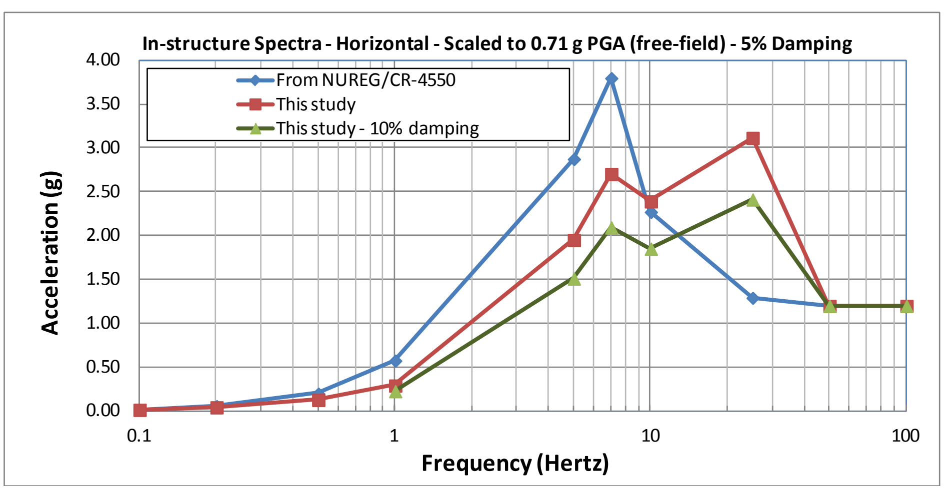

Figure 13 Horizontal ISRS for 5-percent and 10-percent damping midway between Elevation 195 ft and Elevation 234 ft (midheight of the SFP)

The scaling used to obtain the 5-percent damping ISRS does not take into account reductions on spectral accelerations for frequencies greater than 10 Hz that 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 (RC) 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 14). The foundation slab above this rock pedestal is still an 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 arguable. 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 on 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.

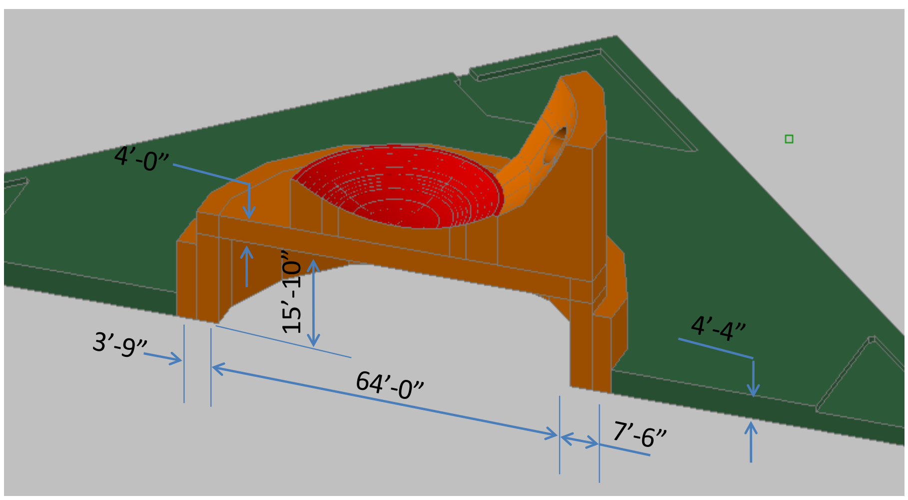

Figure 14 Schematic diagram of the reactor building foundation near the drywell

4.1.2 Description of the Spent Fuel Pool Structure

This section provides a brief description of the SFP structure and its relation to the main reactor building. The description identifies the main structural components and other aspects relevant for this study.

The final safety analysis report (FSAR) for PBAPS 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 at the ends by RC exterior walls on opposite sides of the reactor building. Figure

15 is a 3D representation of the SFP structure and dryer-separator storage pool. Figure 16 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 on the left-hand-side of that figure 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 17) 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 ft deep (above the top of the SFP floor) and about 6 ft 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, spent fuel racks, and the partition wall (south, S, wall). The E and W walls are supported by the thick RC biological shield building on the north (N) side and by the outer wall of the building (on the south side). A cavity exists between the SFP itself and the outer wall of the reactor building.

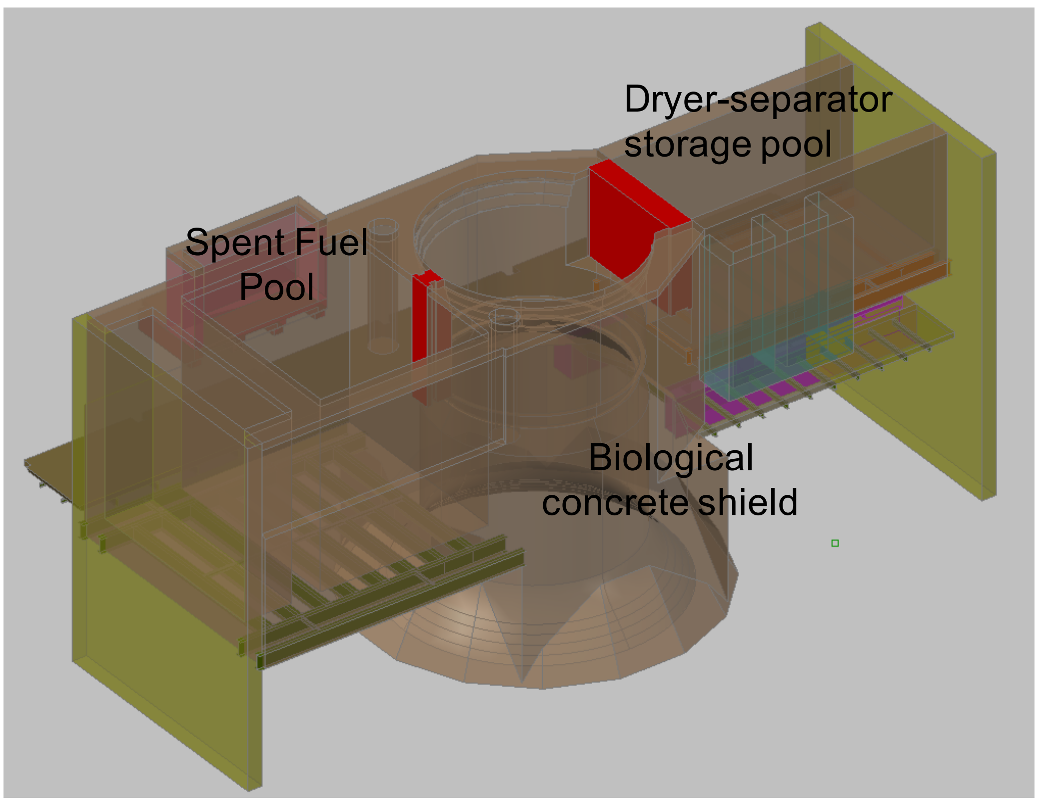

Figure 15 SFP details in cutout of 3D CAD model

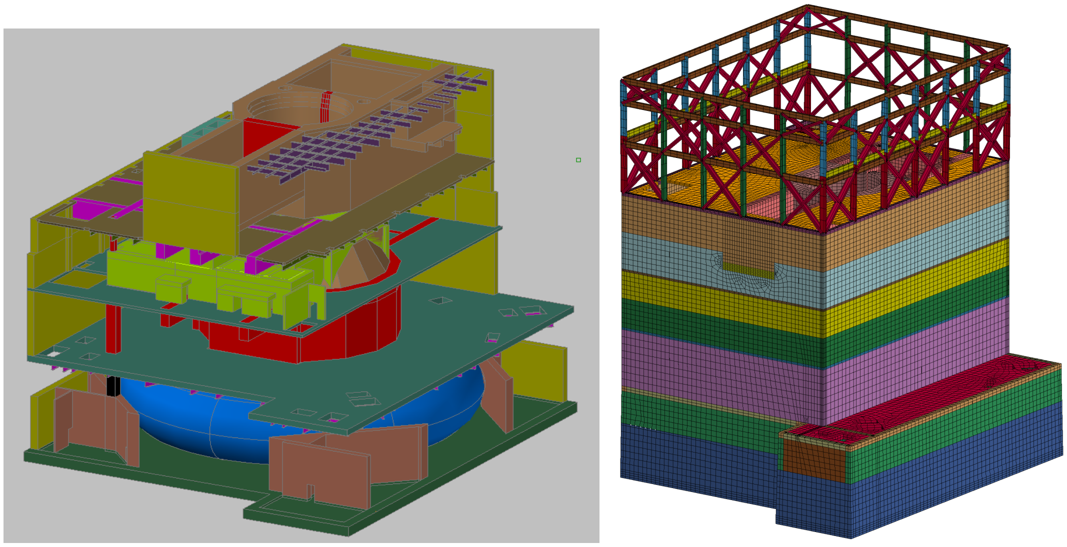

Figure 16 Cutouts of 3D CAD models of the reactor building and SFP

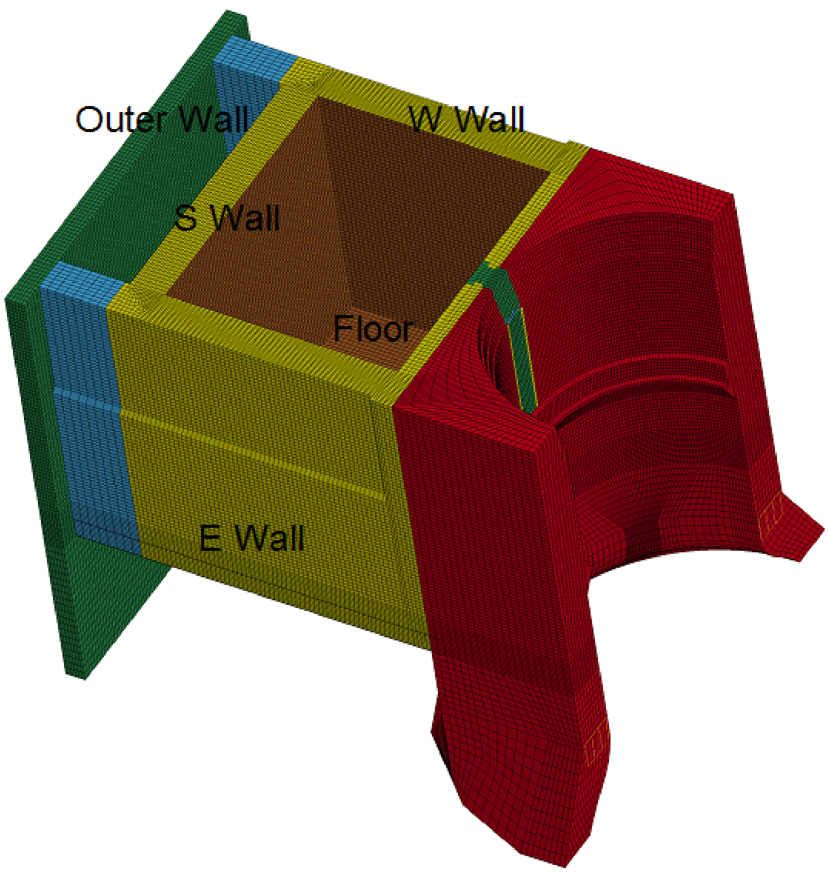

Figure 17 Finite element model of the SFP structure with labels for the floor and walls

Of interest for the study is assessment of damage and cracking to the walls identified in Figure 17 as well as to the floor of the pool from the low probability, seismic event considered in this study. The walls are RC walls with vertical and horizontal layers of #11 (1.41 in. diameter) reinforcing steel bars near each face as well as near the mid surface of the walls.

The SFP floor consists of an RC slab 6 ft 3 in. thick, with embedded heavy steel W-Shapes (I beams) as shown in Figure 15. This floor framing was used during construction and designed to carry the weight of the wet concrete but the beams and decking were left embedded in the concrete floor to the depth of the lower flange of the shapes. The beams that extend from the biological concrete shield to the outer 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. Vertical reinforcement near each face of the wall extends vertically into the floor slab and some of those bars bend and then extend horizontally into the upper half of the pool floor. This is done to provide adequate embedment to the reinforcing bars.

The floor and walls of the SFP are covered with a 1/4-in. thick stainless steel liner which is designed to preclude inadvertent loss of water and that is attached to the concrete using steel anchors, and welds to steel plates and shapes embedded in the concrete floor and side walls. Figure 18 , 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 (E and W walls), is used 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 sump drain.

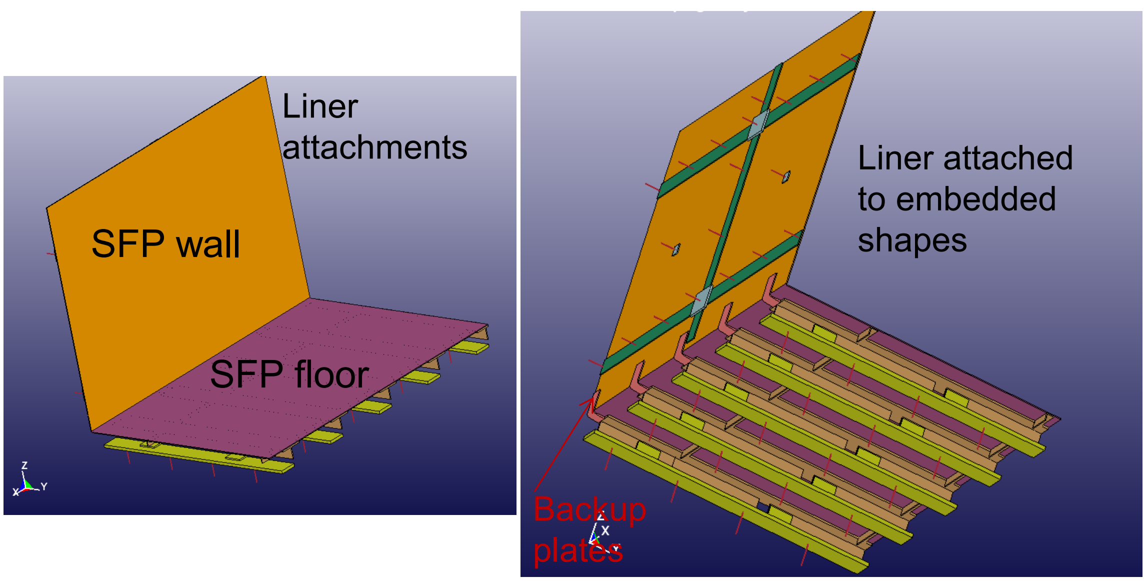

Figure 18 Outline of detailed finite element model of the SFP liner representing attachments to the SFP floor and walls (E and W walls)

According to the FSAR, 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 sentence 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 gate opening (in red in Figure 15) is covered with concrete blocks and closed by two steel gates, in which one steel gate backs the other to provide redundancy in the case of malfunction of a single gate. Each gate consists of steel plates with steel stiffeners. Each gate has polymeric seals around its perimeter that are kept under pressure by the mechanical locking system for the gates. Pressurization of the seals is not a pneumatic system that requires pressurization by electric power systems.

4.1.3 Finite Element Model Description

Step 5 of the approach described in section 4.1.1, the nonlinear pseudodynamic analysis of the SFP 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 LS-DYNA finite element software was used for the analysis (LSTC, 2007). Figure 17 shows the overall detailed finite element model. 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 major reinforcing bars for the floor and walls of the SFP structure as well as 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 LS-DYNA shell elements. In addition, the finite element model also includes the steel liner on the inside surface of the SFP. Figure 19 shows some of the components included in the finite element model.

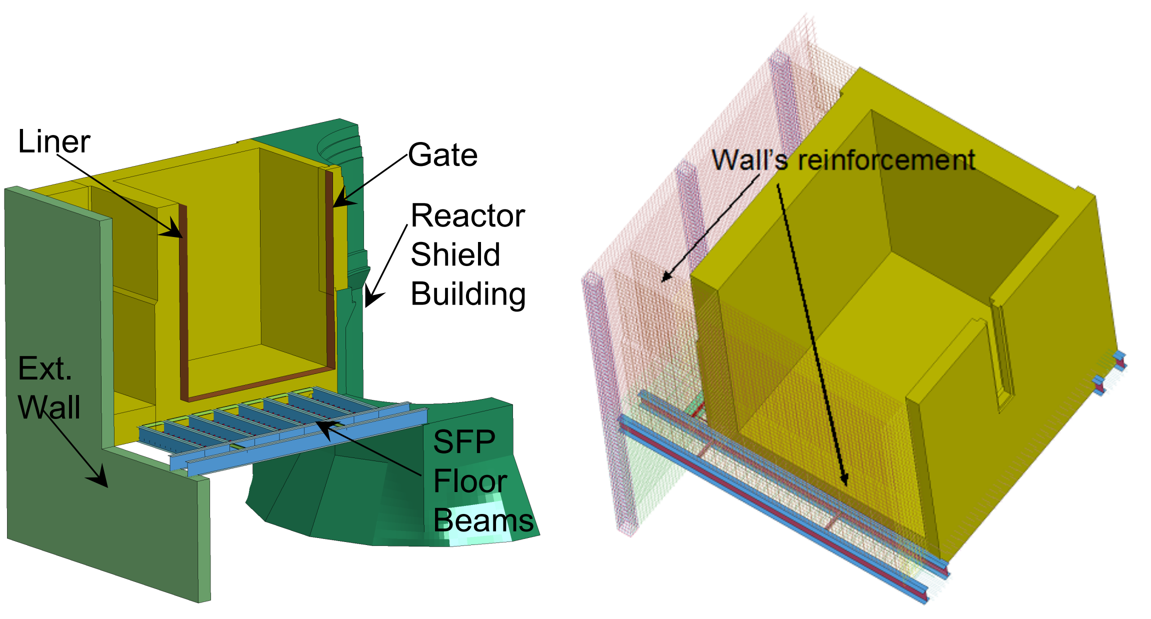

Figure 19 Cutouts of 3D finite element model showing components included in the model

Given the complexity of the structure, rather than using node-to-node modeling for the embedded shell elements modeling the steel beams, the model used the "Constrained

Lagrange in Solid" option available in LS-DYNA to represent the coupling between the embedded elements and the concrete. For the steel liner, two levels of modeling detail 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 bonded to the concrete (node-to-node connections). A more detailed model of sections of the liner (see Figure 18 above) with elements as small as 3.7 millimeters (mm) (0.15 in.) wide was subsequently used as an embedded gage to assess strain concentrations in the liner plates at the intersection of the floor and walls as discussed in the following section.

Boundary conditions for the nonlinear finite element analysis are as follows: (i) vertical and horizontal displacements fixed at the bottom of the exterior wall and of the radiological concrete shield building; (ii) horizontal displacements fixed at the edges of the exterior wall and at the edges of the radiological concrete shield building; and (iii) horizontal displacements in the direction perpendicular to the E and W walls fixed at the top of the E and W walls from the exterior wall to the radiological concrete shield building. Fixing the horizontal displacements at the top of the E and W wall in the direction perpendicular to the walls is justified on the basis of the 1 ft 7 in. thick composite floors with a reinforced concrete deck continuous with the SFP walls on each side of the SFP at the top of the E and W walls (Elevation 234 ft) and that extend to the exterior walls.

The finite elements in the model for the nonlinear analysis are as follows:

- reinforcing bars—LS-DYNA beam elements with the truss option.
- concrete—Constant stress LS-DYNA solid elements (reduced integration)
- shell elements—Belytschko-Tsay shell elements.

Two material models were used as follows:

- Concrete—LS-DYNA material model 159 known as 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—LS-DYNA 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 be the materials that would most influence 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, 2011) and a nominal concrete strength of 4,000 pounds per square inch (psi) (27.5 MPa). For the other materials, the table primarily lists nominal properties. In the case of the liner, nominal material properties were assumed for its yield strength and Young's modulus. These properties and the liner itself are not expected to have a significant effect in the overall response of the SFP structure. However, liner strains and failure strains for the liner are critical in assessing the leakage potential for the SFP. An approach to assess failure of steel liners in reinforced concrete containments is used together with simple probabilistic models to estimate the relative likelihood of the damage states as described in Section 4.1.5.

Table 7 Material Properties for the Nonlinear Finite Element Analyses

Material	Properties	
Concrete	Unconfined compressive strength Aggregate diameter Unit weight (and density) Young's Modulus (for reference)	6,400 psi (44.6 MPa) 1.5 in. (38 mm) 145 lb/ft^3 (2.33 g/cm^3) 4.5x10^6 psi (31,000 MPa)
Rebars	Yield strength (Grade 40) Yield strength (Grade 60) Young's modulus Tangent modulus Unit weight (and density) Failure strain	47,850 psi (330 MPa) 69,000 psi (475 MPa) 31x10^6 psi (2.15x10^5 MPa) 15x10^4 psi (1000 MPa) 479 lb/ft^3 (7.7 g/cm^3) 0.10
Liner and steel plate anchorages	Yield strength (Grade 40) Young's modulus Tangent modulus Unit weight (and density) Failure strain	36,000 psi (250 MPa) 30x10^6 psi (2.07x10^5 MPa) 15x10^4 psi (1,000 MPa) 479 lb/ft^3 (7.7 g/cm^3) Treated as variable
Beams	Yield strength Young's modulus Tangent modulus Unit weight (and density) Failure strain	36,000 psi (250 MPa) 30x10^6 psi (2.07x10^5 MPa) 25x10^4 psi (1,700 MPa) 479 lb/ft^3 (7.7 g/cm^3) 0.10
Anchor studs	Yield strength Young's modulus Tangent modulus Unit weight (and density) Failure strain	36,000 psi (250 MPa) 30x10^6 psi (2.07x10^5 MPa) 25x10^4 psi (1,700 MPa) 479 lb/ft^3 (7.7 g/cm^3) 0.10

This study used a simpler version of the model used for the nonlinear analysis. This 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 (ANSYS, 2011). 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 it 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 elements 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: (1) Young's modulus of 3.15x10^{6}psi (reduced to 70-percent of the Young's modulus of reference to account partially for cracking effects on stiffness) (21,700 MPa), (2) unit weight of 145 lb/ft^3 (2.33 g/cm^3), and (3) a Poisson ratio of 0.15.
- Water: (1) bulk modulus of 3.16x10^5 psi (2,180 MPa), (2) unit weight of 62.4 lb/ft^3 (1 g/cm^3), and (3) a viscosity of 1.64x10^{-7} psi-s (1.13x10^{-9} MPa-s).

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 were done to estimate seismic load coefficients for structural components and to verify the magnitude of the hydrodynamic pressures on the SFP walls.

Summary of Dead and Seismic Loads for the Finite Element Analysis

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 on the 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. Horizontal hydrodynamic loads (not shown in Table 9) considered hydrodynamic pressures from the horizontal ground motions as well as pressures on the wall from the vertical ground motions.

Table 8 Approximate Dead Loads on the SFP Floor in Terms of an Equivalent Vertical Floor Pressure

Load	Approximate equivalent floor pressure in lb/ft^2 (in kPa in the parentheses)
Weight of the floor	900 (43)
Vertical hydrostatic pressure	2,300 (110)
Weight of spent fuel assemblies and racks	1,700 (80)
Total	4,900 (230)

Table 9 Approximate Peak Equivalent Seismic Loads in Terms of an Equivalent Static Vertical Floor Pressure

Load	Approximate equivalent floor pressure in lb/ft^2 (in kPa in the parentheses)
Floor slab acceleration	1,400 (67)
Hydrodynamic impulsive vertical pressure	4,840 (230)
Dynamic forces from spent fuel assemblies and racks	1,750 (85)
Total	7,990 (385)

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 the 5-percent vertical and horizontal ISRS described in Section 4.1.2. It is noted that the (lower) natural frequencies of the SFP, considering a reduction of the concrete Young's modulus to about 70-percent of its original value, the water, and the mass of the spent fuel assemblies and racks, range from about 14 Hz (vertical motion of the floor) to 24 Hz (horizontal motion of the walls). 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 for this study shown in Chapter 3 of this report indicates 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 20 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 are for a free-field PGA of 1.0g and 5-percent damping ISRS. They were multiplied by 0.71 and by the ratio of spectral amplitudes for 10-percent and 5-percent damping 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 4-ftx4-ft 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 LS-DYNA 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. Given the significance of these pressures, deterministic response spectrum analysis with the simplified ANSYS finite element model of the SFP was used in their calculation. Figure 21 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 21 are for a free-field PGA of 1.0g and the 5-percent damping ISRS. They were multiplied by 0.71 for the PGA of interest and by the ratio of spectral amplitudes for 10-percent and 5-percent damping to obtain the values shown in Table 9. Note that water pressures from the vertical accelerations also apply hydrodynamic pressures to the walls, which decrease with height above the floor. The analysis accounted for these pressures.

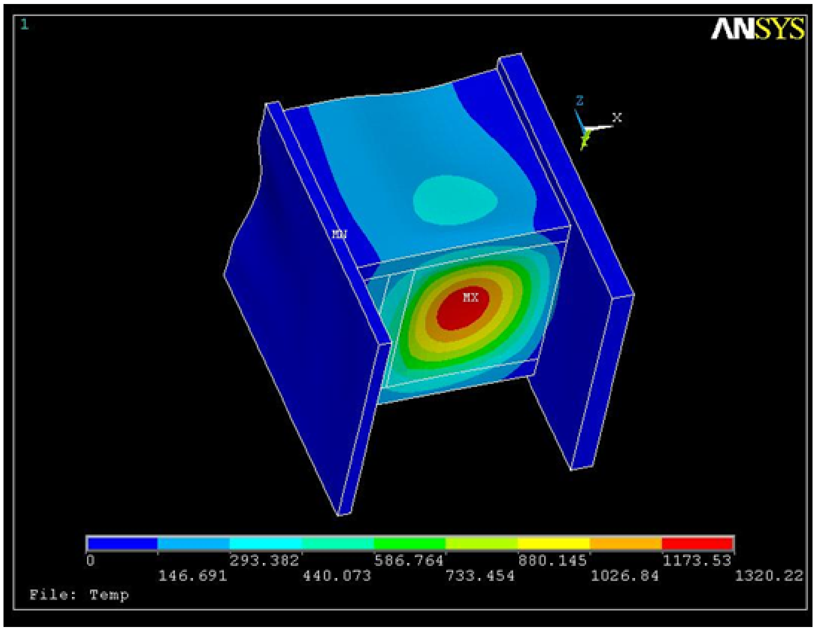

Figure 20 Estimated peak vertical accelerations (in/sec^2) of the SFP floor from response spectrum analysis and vertical ISRS as input (1.0g PGA and 5-percent damping)

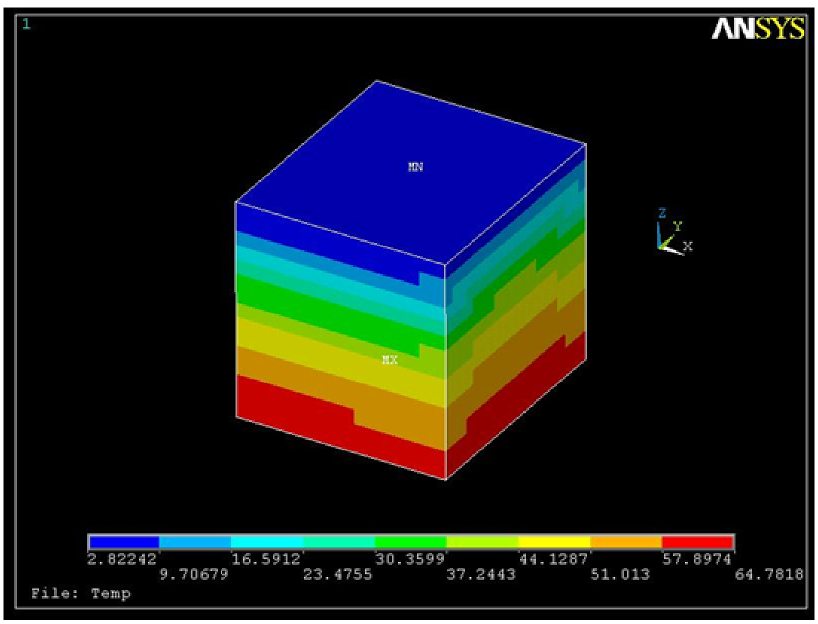

Figure 21 Estimated peak hydrodynamic pressures (psi) on the SFP floor from response spectrum analysis and vertical ISRS as input (1.0g PGA and 5-percent damping)

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 estimated in Section 4.1.3. 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 LS-DYNA software which is an explicit dynamic finite element code. Since this is an equivalent static analysis, the analysis used mass scaling (with only minor 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 they reached their full values. Subsequently, the analysis slowly and proportionally applied all the equivalent seismic static loads until they 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.

Figure 22 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 are small on the order of 0.6 in. (15 mm) or about 0.6/(40x12) = 1/800 of the clear span. Small displacements are a result of the high stiffness of the SFP structure which consists of thick RC slabs and walls (on the order of 6 ft) and comparatively short spans (from about 35 ft in the N-S direction and about 40 ft in the E-W direction).

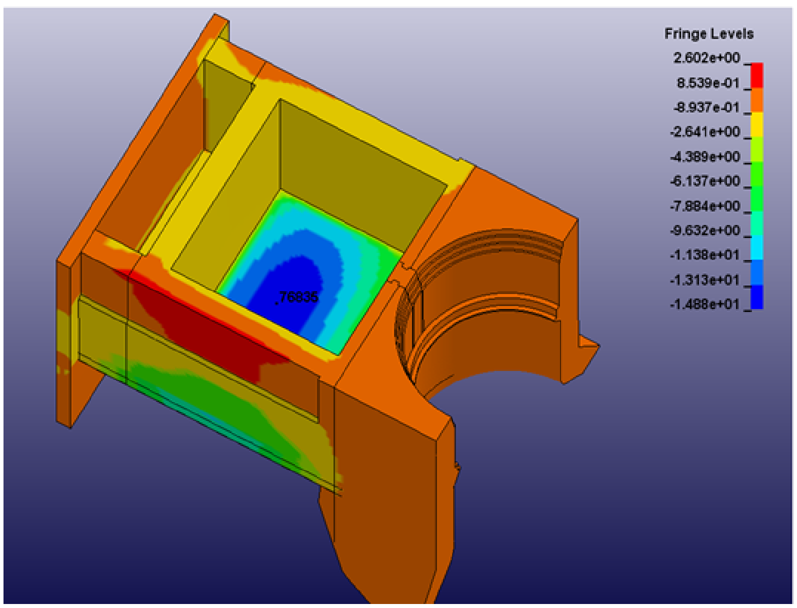

Figure 22 Contours of vertical displacements (mm) of the SFP floor and walls

Figure 23 shows vertical displacement along the outside face of the W wall. Of special interest in Figure 23 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 this is the region of possible strain concentrations in the SFP liner as shown in Figure 24. Finally, Figure 25 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 could likely develop. The crack would start as a flexure crack and develop into a mostly tension-flexure crack though the thickness of the wall accompanied by shear friction at the bottom of the wall.

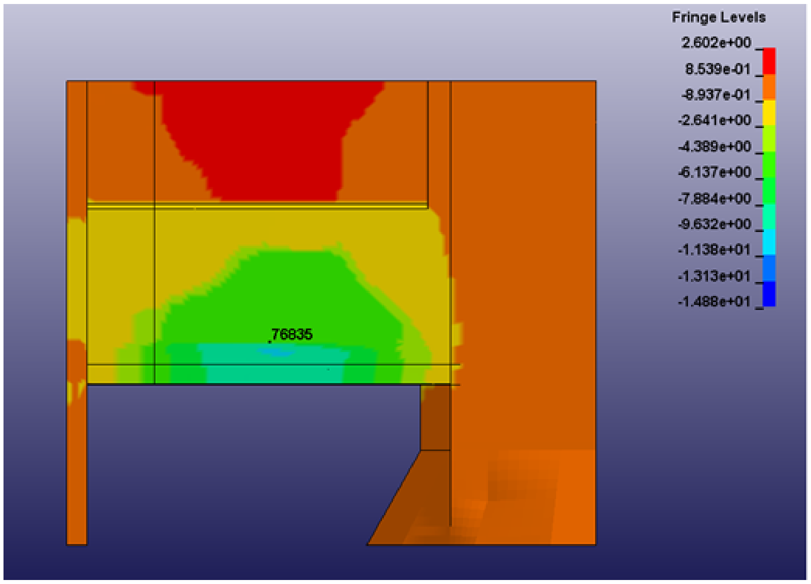

Figure 23 Contours of vertical displacement (mm) of the SFP walls

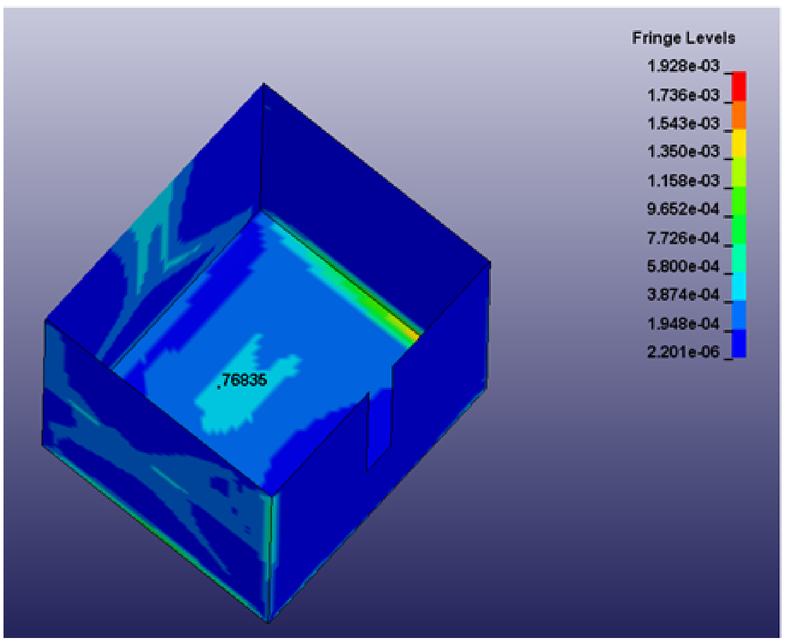

Figure 24 Liner strains (overall response not fully accounting for strain concentrations)

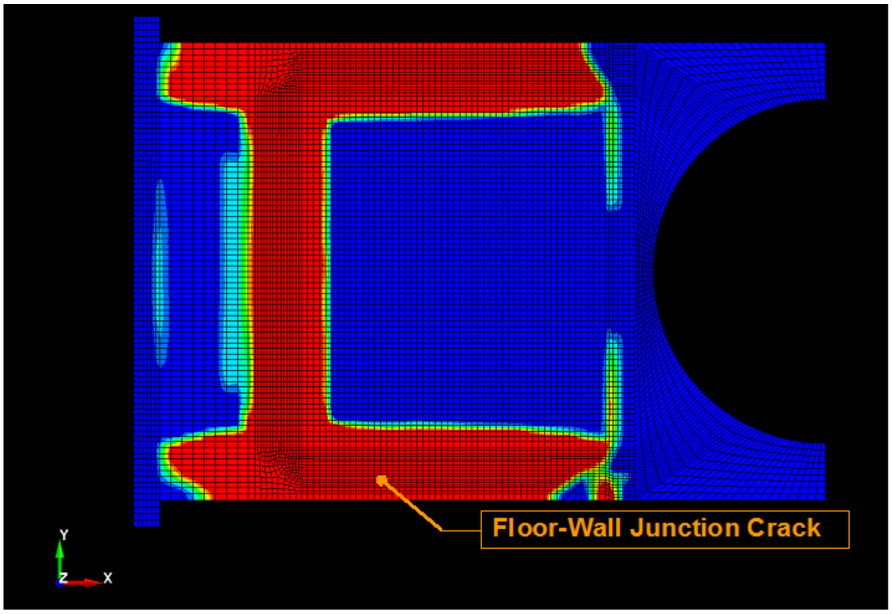

Figure 25 Region of concrete cracking initiation at the floor-wall junction

The higher liner strains in Figure 24 are, as expected, at the intersection of the SFP wall with the SFP floor, which is the region of strain concentrations. Although this is a region of strain concentrations, the liner strains shown are small, of the order of $5x10^{-4}$ to $1.9x10^{-3}$. For comparison, the nominal liner yield strain is $1.2x10^{-3}$. 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 details of 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 18. 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 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 LS-DYNA and appropriate contact definitions) to assess strain concentrations in the liner. Note that the actual liner and the liner in the model are attached to the concrete only at a few discrete locations. Elsewhere, the liner is only in contact with the concrete. 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 18) 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 26 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 does not affect the overall response of the SFP in a significant manner. However, it permits an estimation of the strain concentrations in the liner.

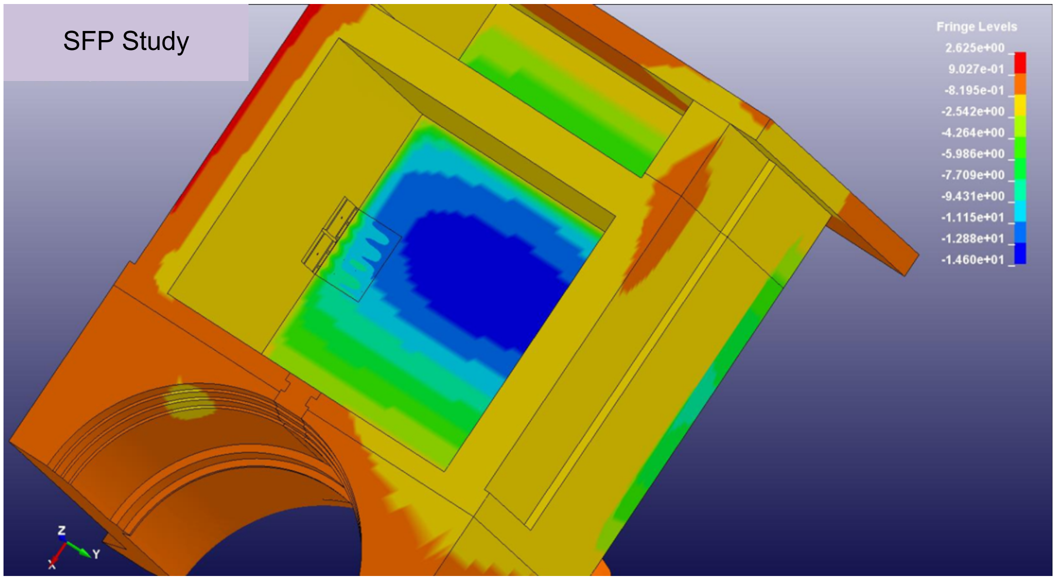

Figure 26 SFP displacements (mm) with detailed liner insert

Figure 27 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, i.e., 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 27 is about 3.7 percent (0.037). The following section uses these strains as well as estimates of the width and extent of the concrete cracking (see Figure 25), to assess liner tearing likelihoods for the scenario considered.

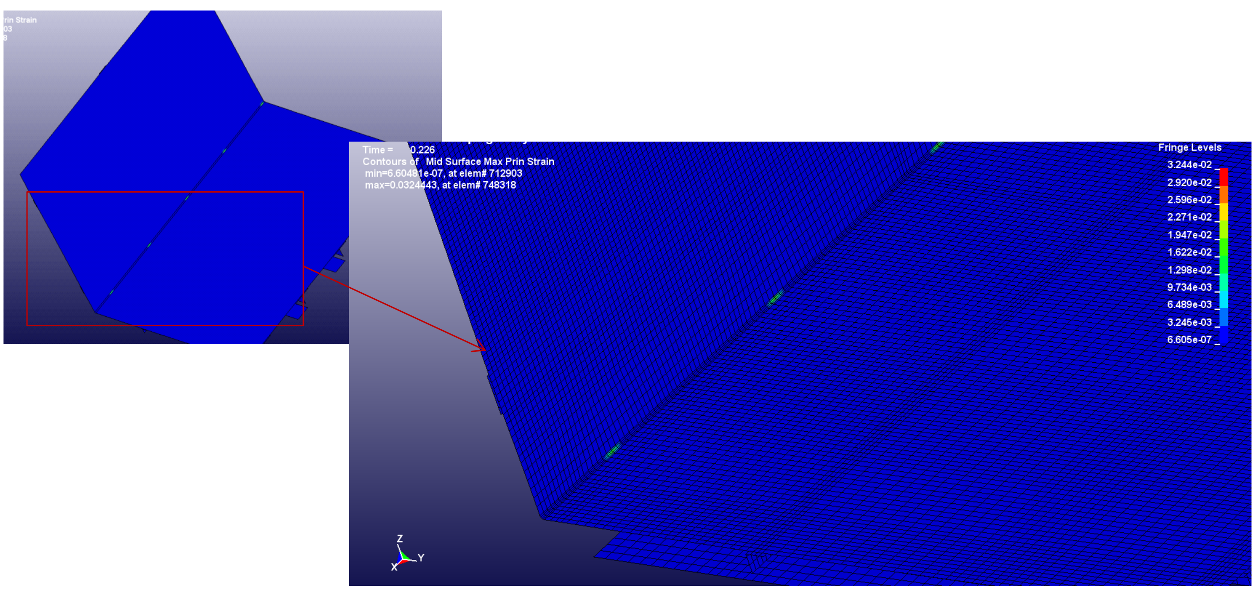

Figure 27 Strain concentrations in the SFP liner

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 4.1.4 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. The section starts with the approach used to estimate the likelihood for each damage state for the initiating seismic event considered. Then, the section provides the estimation of the leakage rates for the damage states with leakage.

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 previous 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.
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.

This study uses an approach and strain criteria, including uncertainties, for tearing of steel liners in reinforced concrete containments (Cherry, 2001 and 1996) together with uncertainties in the calculated liner strains to estimate the relative likelihoods for the three initial damage states listed. Uncertainties in the calculated liner strains account for (1) uncertainties in the ISRS spectral accelerations (of the order of 25 percent), (2) uncertainties in liner strains from uncertainties in concrete properties (namely concrete strength) and (3) an additional reduction in spectral accelerations to account for both ground motion incoherency and nonlinear effects.

The analysis used information in Cherry (1996) to estimate upper and lower bounds for the limiting failure strain, which were then used with a triangular probability density function to estimate their mean and coefficient of variation (Ang, 1984). This is expected to be a conservative assumption in that the SFP liner is of stainless steel which is likely to have larger limiting failure strains. This approach, adjusts the failure strain from coupon tests using reduction factors that account for the multi-dimensional state of stress (triaxiality effects), uncertainties in material properties, and the level of detail in the analysis used to estimate strain concentrations. Bounds in the liner limiting failure strain use a failure strain from coupon tests of 21-percent (0.21) together with a triaxiality factor of 1.75 (typical of a cylindrical state of stress). Estimation of these bounds considers a high level of detail in the model for the calculation of strain concentrations which used elements as small as 0.15 in (3.7 mm) wide. Accordingly, this study considered a range of reduction factors for the analysis detail that range from 0.4 to 0.9. Reduction factors for material properties were those reported in Cherry (1996). On these bases, the bounds on the failure strain for the purposes of estimating its mean and coefficient of variation came to be 0.045 and 0.14, and the resulting mean and coefficient of variation came out to be 0.09 and 0.20, respectively.

Maximum effective tensile strains in the liner were calculated assuming reduced material properties and were found to be sensitive to the concrete strength. Effective strains calculated for the median concrete strength and a reduced concrete strength were used to assess the derivative of this strain to the concrete strength. This derivative was then multiplied by the standard deviation of the concrete strength which was calculated using an estimated coefficient of variation for the concrete strength of 0.15 (Lambright et al., 1990) to estimate the standard deviation and coefficient of variation of the liner strain associated with uncertainties in the concrete strength. This coefficient of variation was estimated to be about 0.65.

Maximum effective concrete strains calculated for the 0.7 g PGA and for reduced seismic loads (about 70-percent of the initial loads) were used to estimate the sensitivity of the effective strain to the estimated spectral amplitudes. Nonlinear analyses were done to estimate maximum effective strains for spectral accelerations equal to about 80-percent of the original to account for effects of ground motion incoherency and further reductions from nonlinear effects. Additional uncertainty measures for the calculated strain were then estimated using the calculated strains for the base case and the case with reduced spectral amplitudes in conjunction with an asymmetric triangular distribution for the calculated strains. The assumed triangular distribution used the strain for the reduced value as the least likely value and that for the base case as the most likely value. This procedure resulted in an adjustment of the median strain (reduction factor equal to 0.93) and an additional coefficient of variation (0.09) for the liner strain. An additional coefficient of variation for the ultimate strain of about 0.25 was used to account for uncertainties in the estimate of floor response spectra ordinates (Lambright et al., 1989).

Uncertainties calculated in this manner were then used to estimate medians and coefficients of variation for the limiting failure strain (capacity) and for the induced strain (demand). Using these quantities and assuming lognormal distributions, the probability of liner tearing conditional on the occurrence of the seismic event was estimated to be less than 10-percent (Ang, 2006; Ang, 1984). This estimate indicates that the state with no leakage (no tearing of the liner) is the most likely with a relative likelihood in excess of 90-percent. The relative likelihood of the two states with leakage from the bottom of the SFP is estimated at less than 10-percent. Assigning relative likelihoods to the two damage states with leakage is subject to considerable uncertainties at this time. Accordingly, the assumption is made that both states are equally unlikely.

Concrete Cracking and Moderate Leakage Rate

Postprocessing 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. The 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 subtracting 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 of about 3.6 mm (about 0.14 in.) and an average crack length of about 33,000 mm (about 108 ft), with a non-smooth and non-uniform surface. An average crack width is used because the overall change in the crack width is not expected to be large along the perimeter of the floor.

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 envelope 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 testing are (1) an equation to estimate the leakage flow rate through concrete cracks that involves a friction factor and (2) quantification of that friction factor based on the experimental data. 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 was calculated as shown in Figure 28 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. The average flow rate for this condition to a height of about 16 ft above the SFP floor is about 1,500 gallons per minute.

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 that the liner tearing spreads to an extent such that the leakage rate through the liner is greater than the leakage rate through the concrete cracks. In this case, concrete cracking 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.

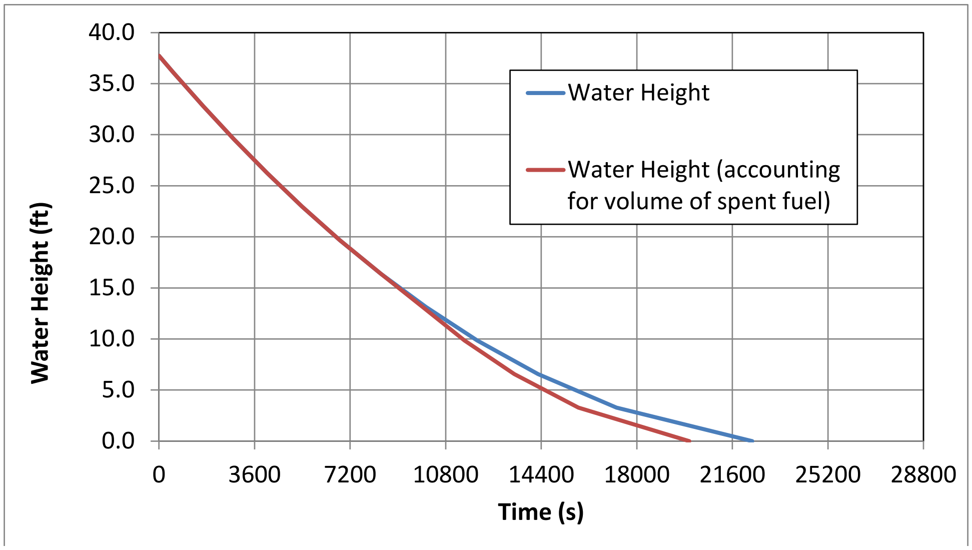

Figure 28 Moderate leakage flow rate (through concrete cracks)

Liner Strains and Small Leakage Rates

Maximum effective membrane liner strains from strain concentrations at the floor-walls junction are on the order of 0.037 (3.7 percent). These strains are localized at the backup plates, which are spaced 24 in (609.6 mm) 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 are smaller. Moreover, liner tearing or through wall (or floor) concrete cracking are not expected near the biological concrete shielding. Accordingly, tearing of the liner, if it were to occur, would be only along the base of the E and W walls.

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 0.15 in (3.7 mm) 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.

The resulting crack width for a liner tear localized at the location of the backup bar is then estimated at 0.15x0.10 = 0.015 in (0.37 mm). 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 24 in (609.6 mm), a total of 40 backup bars (20 on each wall) are used to estimate the summed length of all localized cracks as 40x4 = 160 in (4,064 mm). The estimated width for each crack, if it were to occur, is then 0.015 in (0.37 mm) and the depth of the crack is the depth of the liner which is equal to 0.25 in (6.35 mm).

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%. 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.

Estimation of a friction factor was made using data in Paul et al. (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 temperatures and the driving pressures are much higher than those applicable to the SFP. Review of other flow models reported in Paul et al. (1994) indicates that for relatively smooth cracks friction will be low. Assuming relatively smooth cracks, the equation for the flow through concrete cracks was applied for flow through steel tears together with a small friction factor (0.11) in order to estimate the leakage flow rates. For this friction factor, the estimated leakage flow through the steel cracks (small leakage flow) is as shown in Figure 29. Considerable uncertainty continues to exist in the estimation of leakage flow rates for these localized liner tears. Given the assumption that the crack surface is relatively smooth, it is estimated that the flow rates in Figure 29 would be greater than the actual flow rates. The average flow rate for this condition to a height of about 16 ft (488 mm) above the SFP floor is about 200 gallons per minute (757 liters per minute).

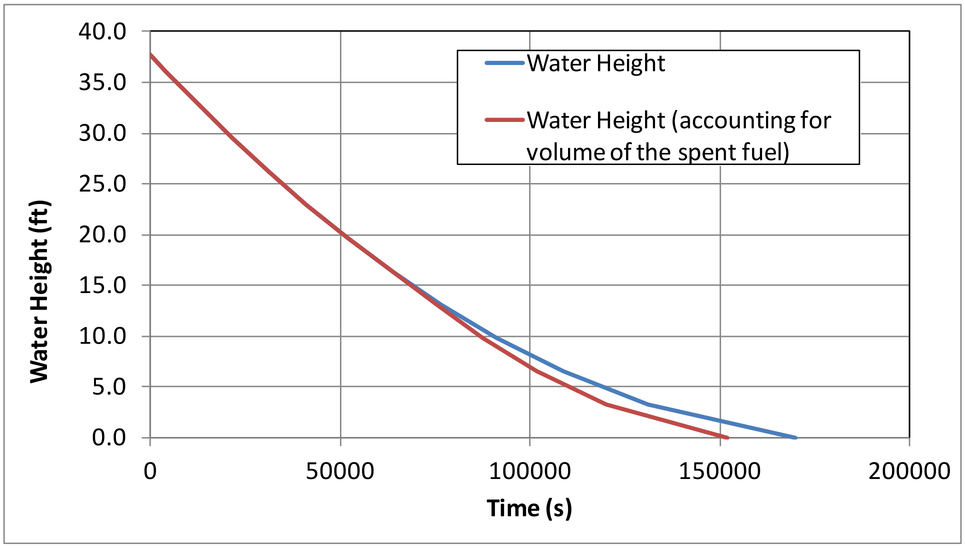

Figure 29 Small leakage flow rate (through localized steel tears)

4.2 Other Damage States

Assessment of other damage stages is primarily based on (1) finite element deterministic response spectra analysis to estimate maximum vertical displacements of the water surface (sloshing), (2) seismic fragilities used in conjunction with the NUREG-1150 seismic PRA study (Lambright et al., 1990), (3) the examination of design details for certain appurtenances such as the refueling gate, and (4) 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. Finite element analysis using the ANSYS finite element model described above, show that the natural frequencies of the sloshing modes in the two horizontal directions parallel to the walls of the SFP are about 0.27 Hz and 0.29 Hz, corresponding to periods of vibration on the order of about 3.8 to 3.5 seconds. These results resemble those obtained using analytical methods (e.g., AEC, 1963; Malhotra et al., 2000).

The free-field ground motion specified for the study does not have high spectral velocities and accelerations at the sloshing frequencies. Consequently, sloshing amplitudes are expected to be small. Deterministic response spectrum analyses with the simplified ANSYS finite element model of the SFP using the horizontal ISRS at midheight of the SFP (for the frequencies of interest to sloshing) as input and considering the low damping of the sloshing mode, show that the sloshing amplitude will not exceed about 20 in. Given that the water at the pool is about 1 ft below the top of the SFP, sloshing is not expected to cause more than 1 ft of water loss. Accordingly, an initial 1.5 ft 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 or dc 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, the study assumes that the refueling gate will not fail for the seismic event considered and will continue to maintain its intended function during the accident progression.

SFP penetrations: According to the FSAR, 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. Results of the nonlinear finite element analysis also indicate that overall distortions of the pool walls are small (on the order of a few millimeters). These distortions are not expected to lead to seismically induced damage of the penetrations that would lead 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 study assumes 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 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. In addition, the high-frequency components (greater than 10 Hz) of the motion would not be expected to induce large sliding or rocking motions.

Damage to the Reactor Building and Other Relevant SSCs

According to the fragility analysis for the NUREG-1150 seismic PRA (Lambright et al., 1990), the median fragility for the reactor building is about 1.6g. 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 for horizontal modes of vibration of the reactor building 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 with the same PGA considered in earlier evaluations of the median fragility). On these bases, seismically-induced failure or severe damage to 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 ends as a consequence of the ground shaking, that end of the crane bridge would not fall inside the SFP. Depending on the end of the crane bridge losing support, the crane could fall only a few feet from the SFP, but not inside the SFP.

A LOOP 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 study (Lambright et al., 1990) indicates a high probability of loss of onsite ac power (about 0.84). This estimate is based on either direct failure of the onsite emergency diesel generators (assumed to be sensitive to spectral accelerations around the 20 Hz frequency) or failure of either the emergency service water or the emergency cooling water systems that provide cooling water for the diesel generators. The probability of losing dc power based on the fragility of the inverters alone is estimated to be close to but less than 50-percent for the seismic event considered in this study.

4.3 Review of Spent Fuel Pool Performance under Recent Major Earthquakes in Japan

Five Japanese nuclear power plant sites with a combined total of 20 reactors and 20 SFPs were subjected to severe ground motions from two major earthquakes in the past 5 years (NERH, 2011a; NERH, 2011b; Kawamura, 2008; Sato, 2010):

- March 11, 2011, Tohoku earthquake (with moment magnitude M_w= 9.0)
  - Fukushima Daiichi (5 BWR Mark I and 1 BWR Mark II SFPs)
  - Onagawa (3 BWR SFPs)
  - Fukushima Daiini (4 BWR SFPs)
  - Tokai (1 BWR SFP)
- July 16, 2007, Niigataken Chuetsu-Oki earthquake (M_w= 6.6)
  - Kashiwazaki-Kariwa (7 BWR SFPs)

This review addresses reductions in water levels for the SFPs affected by those events that might have resulted from either water leakage from structural damage or water loss from sloshing, if any.

No leakage of water near the bottom of the SFPs has been reported for any of the 20 SFPs in those five nuclear power plants for these two major earthquakes. For the Kashiwazaki-Kariwa site, the only report of water loss (leakage or sloshing) for the seven SFPs at the site was a loss of about 320 gallons (about 1.2 cubic meters) from sloshing of the water in the SFP of Unit 6 (Kawamura, 2008).

Loss of water other than from sloshing was not reported for the SFPs of the power plants affected by the March 11, 2011 Tohoku earthquake (NERH, 2011b). According to the NERH (2011b) report, minor leaks of radioactive material (all contained inside buildings) at the Onagawa plant were attributed to sloshing of SFP water, and SFP sloshing overflow lead to a 8 in (20 cm) decrease of the water level in the SFP at Tokai. Actual decreases in SFP water levels from sloshing at the Fukushima Daiichi units are not known, but decreases in water level from sloshing have been assumed in evaluations of SFP performance (NERH, 2011b). Specifically, a water level reduction of about 1.6 ft (0.5 m) was assumed for Unit 2 as a result of sloshing induced by the ground motion while reductions of about 5 ft (1.5 m) were assumed for Units 1, 3 and 4 from sloshing associated with ground motions and explosions.

This review also 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 this review and comparison use information available at the time of the execution of this study, they assist in the interpretation of the results obtained for the seismic scenario and SFP 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 Study. A possible exception to this would be 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). Differences in the seismic design-basis loads and uncertainties regarding the construction details (e.g., out of plane shear reinforcement if any) for the various SFPs listed above add to the overall level of uncertainty in the comparisons. However, this section provides a comparison of the structure of the SFP considered in this study and the structure for the SFP of Fukushima Daiichi Unit 4, for which some structural information was available at the time of the writing of this report.

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, are not directly comparable to the free-field PGA considered in the study. However, the free-field ground motion for this study is also taken to be the foundation ground motion because the reactor building is considered to be a fixed-base structure. Additional sources of uncertainty are the type of reactor (several of the plants have Mark II containments instead of Mark I containments), site conditions (soil versus rock sites), reactor building foundation (slab thickness and uniformity) and reactor building embedment. Generally, the foundation slabs for the reactors listed above are thicker and more uniform than that for the reactor considered in the study. Also, the site for the study is a rock site and stiffer than the sites for Fukushima Daiichi and Kashiwazaki-Kariwa.

An additional source of uncertainty for the comparison is that ISRS reported for some plants may be affected by localized structural details such as the vertical response of a floor slab. Such ISRS would not be representative of the seismic loads on the SFP in the same sense as the ISRS used in this study. Precise determination of the location of the accelerometers used for the observed ISRS was not done for these review and comparison. 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 0.7g. 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 study with the exception of that for Kashiwazaki-Kariwa Unit 1.
- Vertical PGAs at the foundation slabs of all reactors are for the most part less than horizontal PGAs with the exception of Fukushima Daiini Unit 1 and Kashiwazaki-Kariwa Units 6 and 7.

- Vertical PGAs at the foundation slabs of all reactors are less than those considered in the study.
- The difference between the recorded PGAs and the PGA for the study is greater for the vertical accelerations than for the horizontal accelerations.
  - The study assumes that the vertical PGA is approximately equal to the horizontal PGA (see Section 3.3).

Table 10 Fukushima Daiichi, Measured and Design (DBGM S_S) PGAs at Foundation Slab (Tohoku, 2011 Earthquake)

Unit	Containment	Measured (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark I	460	447	258	487	489	412
2	Mark I	348	550	302	441	438	420
3	Mark I	322	507	231	449	441	429
4	Mark I	281	319	200	447	445	422
5	Mark I	311	548	256	452	452	427
6	Mark II	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 (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	BWR	540	587	439	532	529	451
2	BWR	607	461	389	594	572	490
3	BWR	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 (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark II	254	230	305	434	434	512
2	Mark II	243	196	232	428	429	504
3	Mark II	277	216	208	428	430	504
4	Mark II	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 (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark II	214	215	189	393	400	456

Table 14 Kashiwazaki-Kariwa, Measured and Design PGAs at Foundation Slab (Chuetsu-Oki, 2007 Earthquake)

Unit	Reactor	Measured (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark II	311	680	408	274	273	235
2	Mark II	304	606	282	167	167	235
3	Mark II	308	384	311	192	193	235
4	Mark II	310	492	337	193	194	235
5	Mark II	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 study is a rock site and dominant seismic event for this scenario would be an earthquake in the CEUS at a distance of about 15 km or less. Accordingly, the ground motion response spectra for the seismic scenario considered has maximum spectral accelerations for frequencies greater than about 10 Hz and at frequencies near the lower fundamental frequencies of the spent fuel pool structures.

Figure 30 includes vertical response spectra for 5-percent damping at the foundation slab of Unit 1 (the case with a horizontal PGA of about 0.7g) and Unit 4 of Kashiwazaki-Kariwa together with the corresponding response spectrum for the vertical ground motion considered for the study. This comparison indicates that the ground motion for this study has higher vertical spectral accelerations near the lower fundamental frequencies of vibration of the SFP structure. Spectral accelerations for the ground motion used in the study remain higher than those for Unit 4 down to a frequency of about 5 Hz and those for Unit 1 down to frequencies of about 4 Hz. The results shown are typical of those for the other reactors at Kashiwazaki-Kariwa (with the possible exception of Unit 6 which has significantly higher spectral accelerations between 6 and 2.5 Hz). The reactors at this plant are Mark II reactors, have reinforced concrete base slabs several times thicker than the reactor considered in this study, and are deeply embedded in the ground.

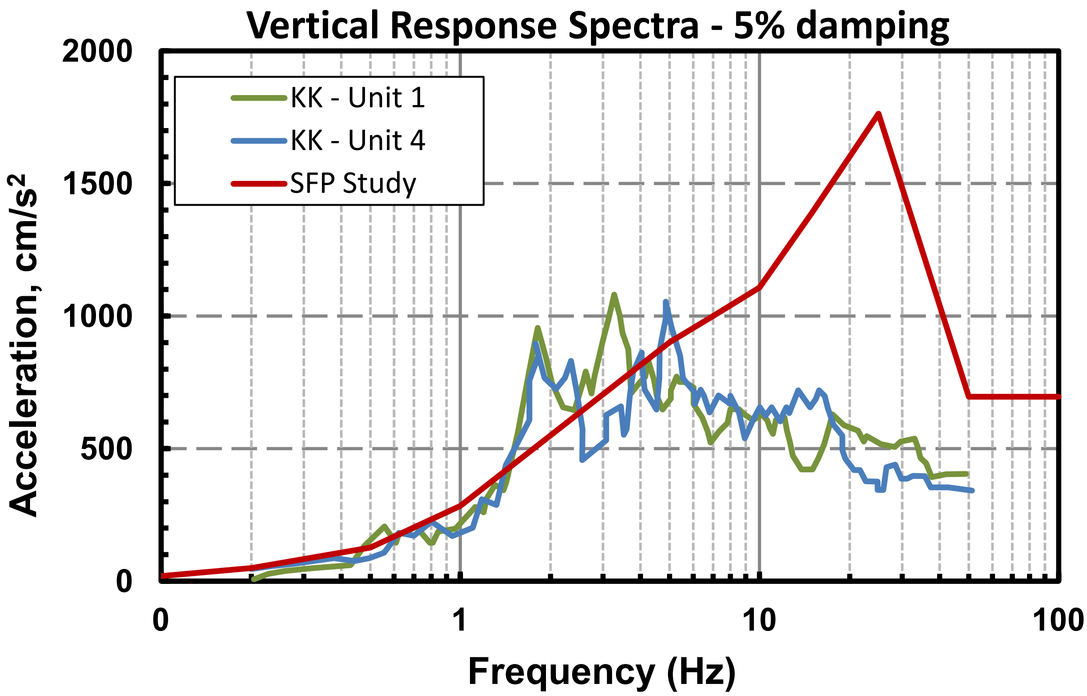

Figure 30 Vertical response spectra: Kashiwazaki-Kariwa Units 1 and 4 (foundation level) and SFP study (free-field)

With the exception of Unit 4 at Fukushima Daiichi, vertical response spectra for the reactors affected by the March 11, 2011 Tohoku earthquake were not available at the time of the study, so the comparison of foundation response spectra are, for the most part, made using horizontal spectra. Figure 31 shows horizontal response spectra for 5-percent damping at the foundation slab of Unit 1 and Unit 4 of Fukushima-Daiichi and the corresponding response spectrum for the horizontal ground motion used in the study. Comparison of those spectra indicates that the ground motion for this study (rock site) has higher horizontal spectral accelerations at the lower natural frequencies of the SFP structure (about 0.05 seconds). Spectral accelerations for the ground motion used in the study remain higher than those for Unit 1 for frequencies down to about 4 Hz and those for Unit 4 for frequencies down to about 3 Hz. The results shown are, in general, typical of those for the other reactors at Fukushima Daiichi.

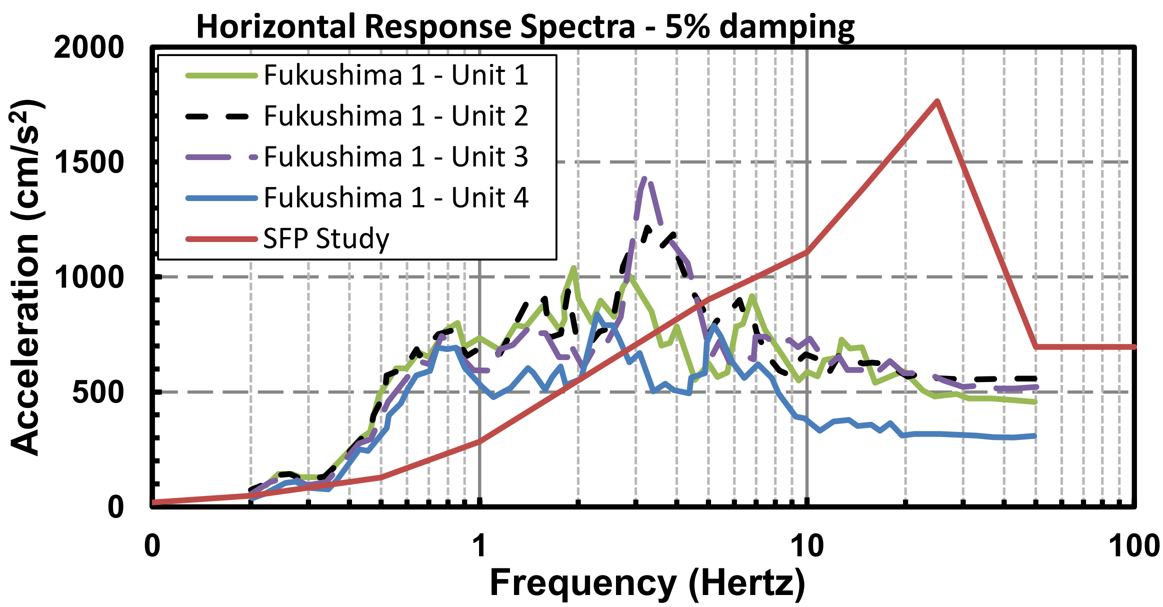

Figure 31 Horizontal response spectra: Fukushima Daiichi Units 1 and 4 (foundation) and SFP study (free-field)

Figure 32 shows vertical response spectra for 5-percent damping at the foundation slab of Unit 4 of Fukushima-Daiichi and the corresponding response spectrum for the vertical ground motion used in the study. Comparison of those spectra indicates that the ground motion for this study (rock site) has higher horizontal spectral accelerations at the lower natural frequencies of the SFP structure (about 0.05 seconds). Spectral accelerations for the ground motion used in the study remain higher than those for Unit 4 for frequencies down to about 3.5 Hz.

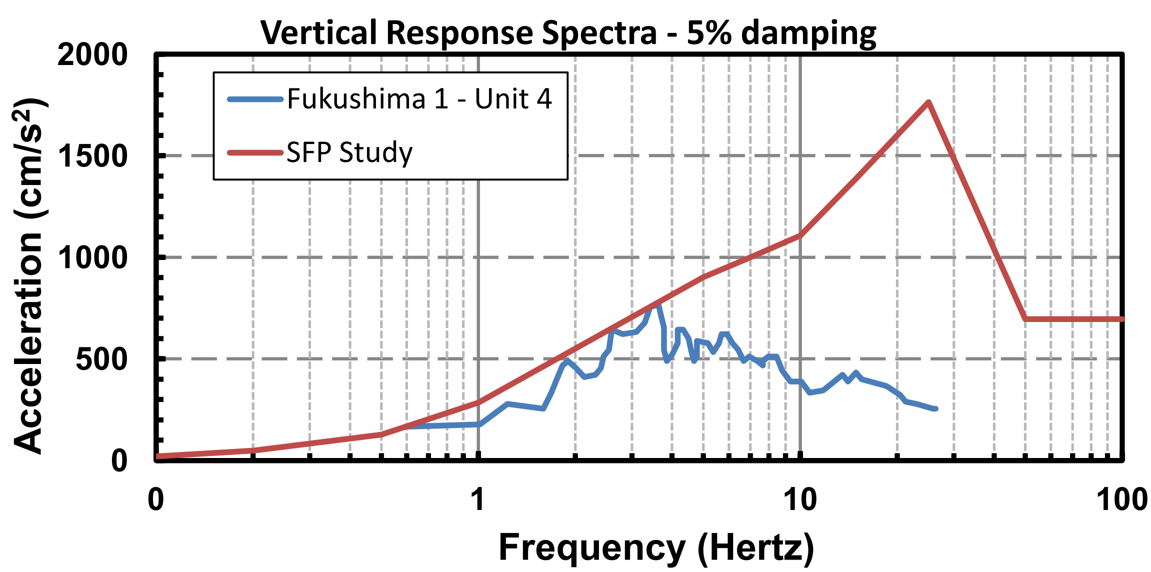

Figure 32 Vertical response spectra: Fukushima Daiichi Unit 4 (foundation) and SFP study (free-field)

Figure 33 shows vertical ISRS at an elevation at about midheight of the SFP for Unit 1 and Unit 4 of Kashiwazaki-Kariwa together with the vertical ISRS for the study. ISRS for the study are shown for 5-percent and 10-percent damping. In both cases, the ISRS for the study is higher than the observed ISRS for frequencies close to the lower natural frequencies of the SFP considered in the study. The 5-percent ISRS for this study remains above that for Unit 4 down to frequencies of about 12 Hz and approximately equal to it for frequencies down to about 7 Hz. The 5-percent ISRS for this study remains higher than that for Unit 1 for frequencies down to about 4 Hz.

The 10-percent damping ISRS for the reactor building approaches that of Unit 4 at frequencies equal to about 17 Hz. The 10-percent ISRS for this study is higher than that for Unit 1 for frequencies down to about 6 Hz and is close to it at about 12 Hz. The ISRS for Unit 1 is typical of those for the other units with the exception of Unit 3, which approaches the 5-percent damping ISRS for the study at about 11 Hz.

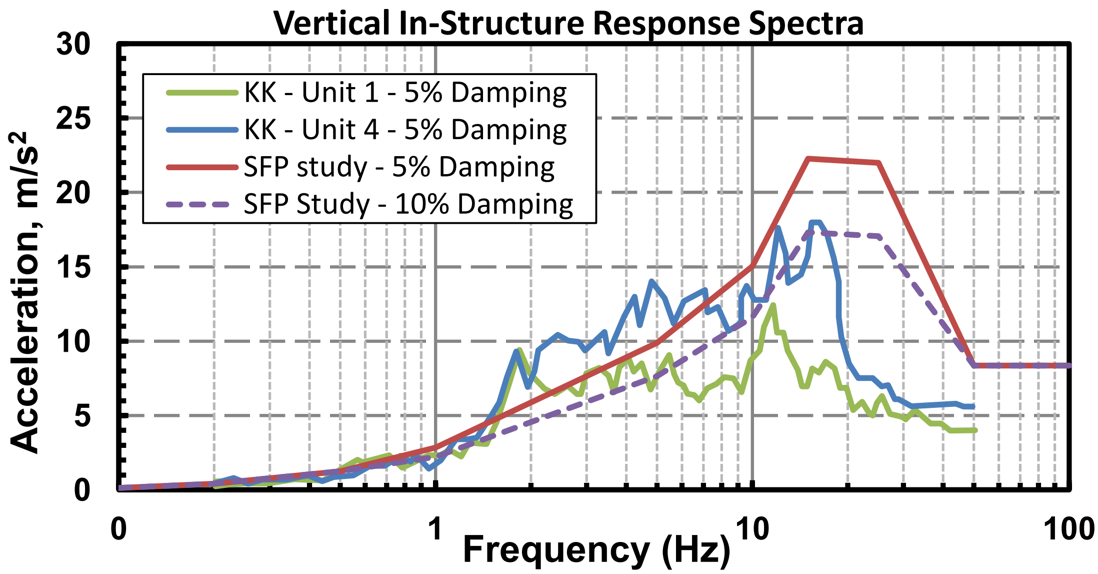

Figure 33 Vertical ISRS for Kashiwazaki-Kariwa Units 1 and 4 and for the SFP Study

The comparisons, especially the comparison of the vertical response spectra at the foundation of Unit 4 of Fukushima Daiichi and at the base of the SFP for the study, indicate that the vibratory loads for this study, especially the vertical loads, are likely to be more challenging to the SFP than those from the actual events.

<u>Structural and Construction Details</u>

The seismic design loads for the various reactors considered in this comparison differ, for the most part, from the design basis loads for the SFP considered in this study. A possible exception to this would be Unit 1 at Fukushima Daiichi, which initially considered comparable seismic design-basis loads. However, seismic design basis loads for Unit 1 were subsequently revised upwards (those are the design loads reported in this comparison).

The depth of the 20 SFPs affected by the recent earthquakes in Japan is similar to that for the SFP considered in the study. The horizontal dimensions for the SFPs in Fukushima Daiichi (EW and NS dimensions with reference to Figure 17) are also similar with the exception to the

SFP for Unit 1, which has a significantly smaller NS span that tends to make the SFP for Unit 1 less vulnerable to seismic loads. The thickness of the floor slabs for the SFPs at Fukushima Daiichi are likely similar to those for the SFP considered in the study. The SFP for which more structural details were known at the time of the writing of this report is the SFP for Unit 4 at Fukushima Daiichi (hereafter referred to as Unit 4). The following provides a comparison of structural details for Unit 4 with those of the SFP considered in this study.

- For Unit 4, the thickness of the SFP floor is about 1.5 m (about 5 ft) which is less than the thickness of the floor of the SFP considered in this study which is about 1.83 m (about 6 ft).
- Available information indicates that the reinforcement of the wall of the SFP in Unit 4 is not significantly different from the reinforcement in the wall of the SFP considered in this study.
- Although no reference is made to out-of-plane shear reinforcement for Unit 4 of Fukushima Daiichi, it is not known with certainty at the time of the writing of this report if out-of-plane shear reinforcement was provided at the edges of the floor slab of Unit 4 or at the intersection of this floor slab with the vertical walls.
- For the SFP of Unit 4 no reference is made to a grid of steel beams analogous to that embedded in the floor and bottom of the walls of the SFP considered in this study (used to support the weight of wet concrete during construction).
- Cross section drawings of the reactor building for Unit 4 indicate the possibility of a load bearing wall under the South wall (with reference to Figure 17) of the SFP of Unit 4, which does not exist for the SFP considered in this study. This difference, if confirmed, would result in a longer span for the entire structure of the SFP considered in this study.

Although there are differences between the structures of the Unit 4 SFP and the SFP considered in this study, these differences do not seem to be sufficiently significant to assert without further analysis that the Unit 4 SFP would be stronger for the same seismic demands than the SFP considered in this study. Differences in the vertical and horizontal response spectra at the foundation of Unit 4 and at the base of reactor building considered in this study (a fixed base structure) (see Figure 32 and Figure 31) indicate that the seismic forces for Unit 4 would have been significantly less than those considered in this study. The difference between these seismic demands would have been the main factor affecting the relative performance of the Unit 4 SFP (under the March 11, 2011 earthquake) and the performance of the SFP considered in this study under the hypothetical beyond design basis earthquake.

Major observations from these comparisons are:

- For the challenging events that affected 20 reactors and SFPs, leakage from the bottom of the SFPs of the 20 BWR reactors was not reported. This is consistent with the highest relative likelihood estimate for this study being that for the state with no leakage.
- Possible differences in the design and construction of the reactor buildings and SFPs, which considered higher design-basis seismic loads, and the SFP considered in this study, introduces uncertainties in these observations.
- The ground motion used in this study may be more challenging for the spent fuel pool structure than those experienced at the Fukushima Daiichi nuclear power plant from the earthquake that occurred on March 11, 2011, off the coast of Japan, which did not cause spent fuel pool leaks at the bottom of the walls.

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 SFP. 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 five operating cycles at the reference 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.)[1]	—
Cell pitch	15.95 cm (6.28 in.)	—
Open vs. closed cell	Closed cell[2]	—
# of storage locations	3,819	
Fuel loading		
Min. assem. during outage[3]	3,819 – 764 – 284 = 2,771	284 × 2 = 568
Max assem. during outage	3,819 – 764 = 3,055[4]	284 × 3 = 852
# of assem. after outage	3,819 – 764 = 3,055	284 × 3 = 852
Newer fuel (<5 years)	GE14/GNF2[5]	—
Older fuel (>5 years)	Actual, based on 2003 info.	N/A
Pattern for “hot” fuel	prearranged in 1x4[6]	1x4 “with empties”
Coherent downcomer area[7]	Yes	—
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)[9]	—

1 Later in the conduct of the study the authors became aware that the distance between the pool floor liner and the bottom of the rack baseplate is actually (on average) closer to 26 centimeters (cm) (10.25 in.), depending on adjustments made to the leveling pad during installation. For the cases studied in this report, in which the leakage location is at the junction of the floor and side wall, side calculations have shown that the results are insensitive to this difference (i.e., even at 18 cm sufficient cross-sectional flow area exists to accommodate natural circulation flow). Nonetheless, any future analyses for this site (particularly if they involve leak locations on the pool side wall), should consider correcting this error.

2 This terminology refers to a rack design in which the sides of the rack cells have panels that inhibit or prevent cross-flow, while being relatively open at the top and bottom for axial flow.

3 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 percent of the core) based primarily on the information in Exelon (2011), with a slight change from 270 to 284 assemblies for MELCOR modeling convenience.

4 Sixty of these rack locations may be reserved for storing guide tubes. The study does not address this situation, but it 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 percent of the overall SFP inventory (and less than 2 percent of the radionuclide inventory available for release).

5 See Exelon (2011) for more information.

6 See Section 9.3 of this report for a discussion of how the use of contiguous (uniform) patterns would affect the results.

7 This term is used to describe whether an open area exists within the pool (such as an unracked region, a cask laydown area, or large gaps between the edge of the racks and the pool walls) that would facilitate downflow during conditions resulting in natural circulation air flow under the rack baseplate.

8 Note that the decay heat from the fuel left in the reactor is considered when the pool and reactor well are hydraulically connected.

9 After results were calculated based on a 700 day operating cycle, the authors realized that the correct operating cycle length should be 725 days (including the 25 day outage) rather than 700 days (which didn't include the outage). This error is expected to have a small impact on the overall results.

The above table depicts a 1x4 storage pattern for the recently discharged fuel, based on the approach PBAPS has taken to meet the requirements associated with license condition 2.C.(11) and 10 CFR 50.54(hh)(2). The plant studied actually currently utilizes a 1x8 pattern. Because this pattern is believed to be atypical (relative to the fleet), it is not modeled as the base case in this study. Section 9.2 of this report provides additional analysis that shows the benefit of the 1x8 pattern. Section 9.3 discusses of how the use of contiguous (uniform) patterns would affect the results. Figure 34 illustrates the different patterns.

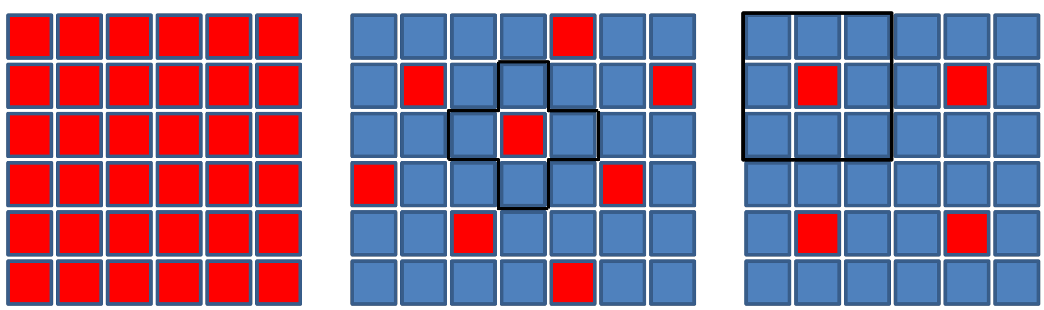

Figure 34 Illustration of SFP patterns

From left to right: Uniform/contiguous; 1x4 (used as base case);
1x8 (actually used by plant as of May 2012)
Red = a recently discharged assembly; Blue = an older, lower decay heat assembly, black outline shows repeating pattern

5.2 Operating Cycle Phase Specification

As described in Section 1.5, constant changes to the conditions in the SFP affect the consequences of a postulated accident (e.g., changes in the decay heat, changes in the inventory of fuel in the pool). Thus it is necessary to discretize this continuous behavior into a manageable set of discrete quasi-steady snapshots. Further, it must be recognized that the number of quasi-steady snapshots (or 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, defining the OCPs becomes a minimization/optimization problem (i.e., the analysis 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).

Based on these 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 a set of five OCPs, as outlined in Table 16.

Table 16 OCP Definition for the Modeled Operating Cycle

OCP #	OCP Description	Time (d)	% of opera-ting cycle	Pool-reactor configuration	Modeled 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	1x4	Existing[1] + (27% of offloaded assemblies) @ 4 days[2]	Highest powered offloaded assembly @ 4 days[2]
2	Reactor T&M / inspection and refueling	8–25	2.4	Refueling	1x4	Existing[1] + (offloaded assemblies) @ 13 days[2]	Highest powered offloaded assembly @ 13 days[2]
3	Highest decay power portion of nonoutage period	25–60	5	Unconnected	1x4	Existing[1] + (offloaded assemblies) @ 37 days[2]	Highest powered offloaded assembly @ 37 days[2]
4	Next highest decay power portion of nonoutage period	60–240	25.7	Unconnected	1x4	Existing[1] + (offloaded assemblies) @ 107 days[2]	Highest powered offloaded assembly @ 107 days[2]
5	Remainder of operating cycle	240–700; 0–2	66	Unconnected	1x4	Existing[1] + (offloaded assemblies) @ 383 days[2]	Highest powered offloaded assembly @ 383 days[2]

1 The term "existing" refers to the fuel residing in the SFP at t = 0 (before offload).
2 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

The following key assumptions in the above OCP definition warrant highlighting:

- The study does not explicitly treat the offloading of older fuel into casks (as part of the normal fuel management practices as opposed to an expedited fuel movement program). Rather, a stylized assumption is made that the 284 assemblies that would be loaded into dry casks during the operating cycle are instantaneously removed from the pool just before the outage.

- The study does not treat new fuel. This fuel would be placed into the SFP just before 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 available to participate in a propagating zirconium fire, but would have a negligible effect on the source term.[6]

- The actual time at which the snapshots are evaluated is based on the mean decay heat during the OCP, as opposed to the mean 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 OCP1, 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

One of the objectives of this study is to provide insights into the effectiveness and benefits of mitigation measures currently employed at nuclear power plants. In addition to the redundant and diverse physical systems designed to prevent severe accidents, NRC requires plant owners to have preplanned emergency measures in the unlikely event an accident occurs. When they are successfully implemented, NRC expects these emergency measures will mitigate accident consequences by preventing, delaying, or reducing a potential release of radioactive material from the SFP. These measures include a site-specific emergency plan, emergency operating procedures, severe accident management guidelines, and 10 CFR 50.54(hh)(2) mitigation measures put in place to respond to the loss of large areas of the plant due to fires or explosions. NRC requires its licensees to train and practice emergency measures to ensure that they have proper equipment, procedures, and training. NRC inspectors periodically observe these activities to help ensure that NRC regulations are met at each plant. The study assumes that the licensee's emergency response organization would implement these measures in accordance with approved emergency plans, procedures, and guidelines.

Regarding onsite mitigative actions, the assumptions chosen by the project team to define the scenarios analyzed using MELCOR and MACCS2 are described here. Two cases are modeled for each scenario, a mitigated case and an unmitigated case. In the mitigated case, the model includes what would happen if the operators are fully successful in carrying out onsite mitigating actions. However, NRC analyzes extreme events to gain insights on the safety margin provided by NRC's regulatory framework. The uncertainties associated with the response to a beyond-design-basis seismic event, and the resultant effects on the SFP, make consideration of unmitigated scenarios prudent from an informed decision-making standpoint. Thus, each scenario is also analyzed assuming that the operators are not successful in implementing onsite mitigating actions.

[6] The radioactive material that is of concern during an accident is the fission products generated while the fuel is in the reactor. The uranium dioxide (UO_2) present in fresh fuel would not contribute noticeably to the source term, and in particular, not in a SFP accident in which the temperatures during a postulated accident are lower than those during a reactor accident.

In the unmitigated case, 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 10 CFR 50.54(hh)(2)[7] equipment. The cases which assume lack of successful mitigation are presented to (1) acknowledge uncertainties in the effectiveness of these efforts during a beyond-design basis event and (2) demonstrate the effectiveness of successful mitigation. Section 5.3.2 of this report discusses further the rationale for developing results for this situation.

In the mitigated case, (1) mitigative actions associated with the regulatory requirements of 10 CFR 50.54(hh)(2) are successfully deployed, (2) additional onsite capabilities are used to extend the use of this equipment, and (3) arrival of offsite resources allows this equipment to be utilized for an extended period of time (e.g., days) until onsite capabilities can be recovered.

This study's original scope did not include an attempt to quantify the likelihood of successful execution of different mitigative actions that might take place (e.g., makeup using a portable pump, recovery of ac power). Subsequent to completion of the MELCOR (Chapter 6) and MACCS2 (Chapter 7) analyses described in the following chapters, the project staff performed a human reliability analysis for the purpose of providing context regarding human response. The HRA results provide informative data to gain insights on the likelihood of mitigation being successfully implemented as well as possible regulatory enhancements for consideration. Chapter 8 describes the HRA. Since the HRA was performed after the bulk of the analysis was completed, some of the assumptions differ from those described in this Section.

In addition to onsite mitigation, offsite support is considered in the paragraphs below.

The reference plant is supported by an offsite emergency operating facility (EOF). The emergency response organization at the EOF has access to fleetwide emergency response personnel and equipment, including the 10 CFR 50.54(hh)(2) mitigation measures and equipment from the sister plants. Every licensee participates in full onsite and offsite exercises every 2 years where response to severe accidents and coordination with offsite response organizations is demonstrated and inspected by the NRC and the Federal Emergency Management Agency. In addition, the Institute for Nuclear Power Operations and the Nuclear Energy Institute would activate their emergency response centers to assist the site as needed.

Concurrent with the industry response, the U.S. National Response Framework (NRF) would establish a coordinated response of national assets. As described in the Nuclear/Radiological Incident Annex to the NRF, the NRC is typically the Coordinating Agency for incidents occurring at NRC-licensed facilities. As Coordinating Agency, the NRC has technical leadership for the Federal Government's response to the incident. The NRF conducts periodic exercises and provides access to the full resources of the Federal Government. The NRC has an extensive, well-trained and exercised, emergency response capability and has onsite resident inspectors. The NRC would activate the incident response team at the NRC regional office and Headquarters. The focus of the NRC response is to ensure that public health and safety are protected and to assist the licensee with the response.

[7] This section of the regulations deals with the development and implementation of guidance and strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with loss of large areas of the plant resulting from explosions or fire.

However, for the large beyond-design-basis seismic event under consideration in this study, it is possible that significant damage to local infrastructure could occur, requiring emergency resources to also be needed in other areas. Additionally, radiation and other hazards (discussed in Section 5.3.2 of this report) could hinder access to the SFP and key equipment, making prevention or truncation of an ongoing SFP release challenging.

Considering the uncertainties associated with this event as described above, project staff chose a 72-hour time truncation (assumed that the event would be terminated by some means by 72 hours after initiation). The use of a time truncation is a point of uncertainty that can significantly affect the results and is further analyzed in Section 9.8 of this report. Note that like other aspects of this study, the incorporation of ongoing changes in regulatory commitments related to offsite response emanating from the Japan Lessons Learned initiatives is beyond this study's scope.

Regarding offsite support for these situations, the accident progression analysis assumes the following for the purposes of this study:

- Within 24 hours, offsite support arrives.
- Within 48 hours, actions are planned and equipment is staged.
- At 48 hours, if the fuel is not uncovered and the pool can be refilled with an injection rate of 500 gpm (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 because of the additional complexities of (1) accessing the area of the pool when the fuel is uncovered and stopping an ex-containment release in progress and (2) performing a large leak repair.

These assumptions are similar to the assumptions used in NUREG-1935, "State-of-the-Art Reactor Consequence Analyses Project" (NRC, 2012i).

Table 17 summarizes each situation.

Table 17 Summary of Mitigation Assumptions

Item	Situations with successful deployment of onsite mitigation	Situations without successful deployment of onsite mitigation
Installed accident mitigation equipment	Damaged by the event; recovery/repair not credited	
10 CFR 50.54(hh)(2) equipment	Successfully deployed 2 hours after diagnosis	Not credited
Other onsite resources	Successfully deployed to extend operation of 10 CFR 50.54(hh)(2) equipment	Not credited
Offsite resources	Successfully deployed for terminating the accident at 48 or 72 hours (see Section 9.8)	
Emergency preparedness	Effective (see APPENDIX A: of this report for more details)	
Mitigation equipment being considered under NRC Order EA-12-049, dated March 12, 2012	Not considered; may be substantively similar to 10 CFR 50.54(hh)(2) capabilities within the context of this study	

5.3.1 Approach Details and Assumptions

Scenarios that credit successful deployment of the 10 CFR 50.54(hh)(2) measures must include assumptions about how that deployment is executed. In general, this study utilizes some of the limits associated with these capabilities that are contained in Nuclear Energy Institute (NEI) 06-12, Revision 2, "B.5.b Phase 2 & 3 Submittal Guidance," issued December 2006 (which the NRC has endorsed[8]). 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. This study does not invoke that provision because the site in question strives to deploy the equipment within 2 hours regardless of the fuel pattern and the existence of cases without successful deployment of mitigation envelops this effect.

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 delivered per minute for makeup). For PBAPS, the capacities of the available equipment are somewhat higher. The use of the 500 and 200 gpm values in this study 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.[9] As a result, no additional "penalty" is given

[8] The NRC originally endorsed this document for operating reactors by letter dated December 22, 2006 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). B.5.b refers to Section B.5.b of Order EA-02-026, dated February 25, 2002, and later made generically applicable in 10 CFR 50.54(hh)(2).

[9] 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

for inefficiencies associated with spray coverage (i.e., the spray flow rate is applied uniformly across the pool cross-sectional area without further reduction). In either spray or makeup mode, the licensee would utilize a portable diesel-driven pump to pump water from either the fire ring header, the intake canal, or the emergency cooling tower basin to the refueling floor via hoses that would be run up a reactor building stairwell.

The following set of criteria was established to model the time to diagnosis of the need to deploy 10 CFR 50.54(hh)(2) mitigative strategies:

- no ac power
- SFP level decrease by 1.5 m (5 ft), keeping in mind that 0.5 m (1.5 ft) is lost because of sloshing
- 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 SFP, though these specific criteria do not 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 SFP cooling or loss of SFP inventory do refer plant personnel to the guidelines for use of the 10 CFR 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. Note that the details of onsite response are covered more thoroughly in Chapter 8.

The above criteria could be conservative or nonconservative depending on the priorities of operators, and different criteria would clearly be more applicable to other scenarios, particularly those that did not include loss of offsite and onsite power at time zero. The assumption that pool elevation must drop 5 ft can lead to long diagnosis time periods for slowly progressing events, thus leading to a potentially conservative timeline for mitigative action. However, these same slowly developing scenarios are the ones that are least important for 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. Chapter 8 discusses the issue of diagnosis in greater detail.

Regarding the implementation mode, for cases in which the water level in the pool is greater than 0.9 m (3 ft) above the top of the racks (a surrogate for high radiation levels on the refueling floor near the edge of the SFP (see Section 5.4 of this report)) 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

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entering the pool. The regulatory implementation of 10 CFR 50.54(hh)(2) accounts for this inefficiency effect.

in Figure 2-1 of NEI 06-12, Revision 2 (NEI, 2006), regarding whether SFP leakage is excessive. In some respects it is a more complicated approach than might be used, but is arguably a more straightforward 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 OCP1/2, in which, because of 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 ft 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. Section 9.3 of this report investigates the effect of this assumption for a uniform pattern.

Whichever mode is initiated (spray versus makeup), it is assumed to be used for the duration of the event (i.e., no later switching to a different mode). For OCP 1/2 with the “moderate” leakage condition, makeup is deployed. Other, equally reasonable assumptions about mitigation deployment could result in the deployment of sprays instead (which have a potential advantage in terms of mitigation for these conditions). Section 9.3 presents a sensitivity study related to this assumption, for a uniform pattern.

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 a calculation without successful deployment of mitigation):

- Start of calculation/earthquake occurs.
- When SFP level has decreased by 1.5 m (5 ft), and 30 (diagnosis delay) plus 120 (initial deployment delay) additional minutes have transpired, then the following applies:
  - If the water level is greater than 0.9 m (3 ft) above the top of the fuel, then 500 gpm of makeup into the top of the pool commences.
  - If the water level is less than 0.9 m (3 ft) above the top of the fuel (thus indicating excessive leakage) then 200 gpm of spray at the top of the pool commences.

The above assumptions are characterized as optimistic relative to the unmitigated (pessimistic) case. However, it is important to note that aspects of these assumptions assume failures where they may not occur. For instance, the above set of assumptions only credits 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 SFP, several of which have much higher capacities than the mode selected. Table 10.3.1 of the FSAR captures these alternatives, which 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 in this study), these modes require either multiple manual alignments in the vicinity of the SFP and reactor, the availability of ac power for valve manipulations, or the use of equipment that might be involved in reactor recovery (most notably a residual heat removal pump), as well as ac power for pump operation. Finally, as mentioned previously, 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, uncertainties associated with the response to a beyond-design-basis seismic event, and the resultant effects on the SFP, make consideration of unmitigated scenarios prudent from an informed decision-making standpoint. Some specific issues relevant to the situation considered in this report include the following:

- The regulatory requirements for 10 CFR 50.54(hh)(2) equipment currently focus on the use of this equipment for responding to a loss of a 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). Note that some plants (including the reference plant) have already acquired some additional equipment in anticipation of this requirement, with full compliance scheduled for 2016.

- The large seismic event could damage onsite (and offsite) infrastructure designed to facilitate accident response, as well as cause 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. Section 5.4 of this report describes the shielding analyses that inform this aspect of the accident analysis. Chapter 8 further discusses accessibility issues in the context of human response. Note that, as part of the implementation of 10 CFR 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).[10]

- A concurrent reactor event (resulting from the loss of ac power or other damage), or an ongoing accident at the other unit's SFP, could hamper mitigative actions by reducing accessibility because of radiation fields, impeding accessibility because of other hazards such as hydrogen accumulation, or diverting resources (both personnel and equipment). Chapter 8 discusses this issue further.

- An assembly being moved within the SFP (or from the reactor to the SFP or vice versa) at the time of the event, could lead to an earlier radiological hazard for responders, if this assembly were to become uncovered earlier in the event progression, because of its higher position in the SFP. Section 5.4 of this report provides refuel floor dose rate estimates for this situation.

- Accessibility could be reduced if an inadvertent criticality event in the SFP were to occur. See Section 2.3 of this report for more information about inadvertent criticality events.

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[10] The industry's FLEX proposal, developed in response to NRC Order EA-12-049, includes a specification for a means to connect makeup to the installed SFP cooling system to overcome the potential for lack of access to the SFP deck area. This is primarily to address the potential needs for makeup in a saturated condition caused by boil off for an uncooled pool.

For these reasons, this study presents results for cases in which accident mitigation efforts are unsuccessful for some period of time.

5.4 Refueling Floor Dose Rate Analysis Using SCALE

This study included analyses to predict the radiological conditions on the refuel floor for a range of conditions associated with loss of water in the SFP. Note that the analyses described in this section only account for the radiological conditions stemming from neutron and gamma "shine" from exposed radioactive material and do not account for 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 SFP were to commence, radiation fields in the vicinity of the pool would be extremely high.

The analyses described, which Oak Ridge National Laboratory (ORNL) performed, looked at a range of conditions. This range included both the high-density and low-density loading conditions studied in this report, as well as the situation in which a single assembly is in the lifted position at the time of the event. The times following discharge that were considered are the same as those associated with the different OCPs. This portion of the analyses is plant specific for the reference plant, and utilized 2011 vintage information for representing the fuel design and characteristics in the SFP. Calculations were performed using the ORIGEN and MAVRIC modules of the SCALE code suite. MAVRIC in turn used BONAMI, CENTRM, DENOVO, and Monaco routines, along with the FW-CADIS methodology. The analysis used the flux-to-dose conversion factors in American National Standards Institute/American Nuclear Society (ANSI/ANS) 6.1.1-1977. The 200 neutron group and 47 gamma group cross sections based on the ENDF/B-VII.0 cross-section library distributed with SCALE 6.1 were used.

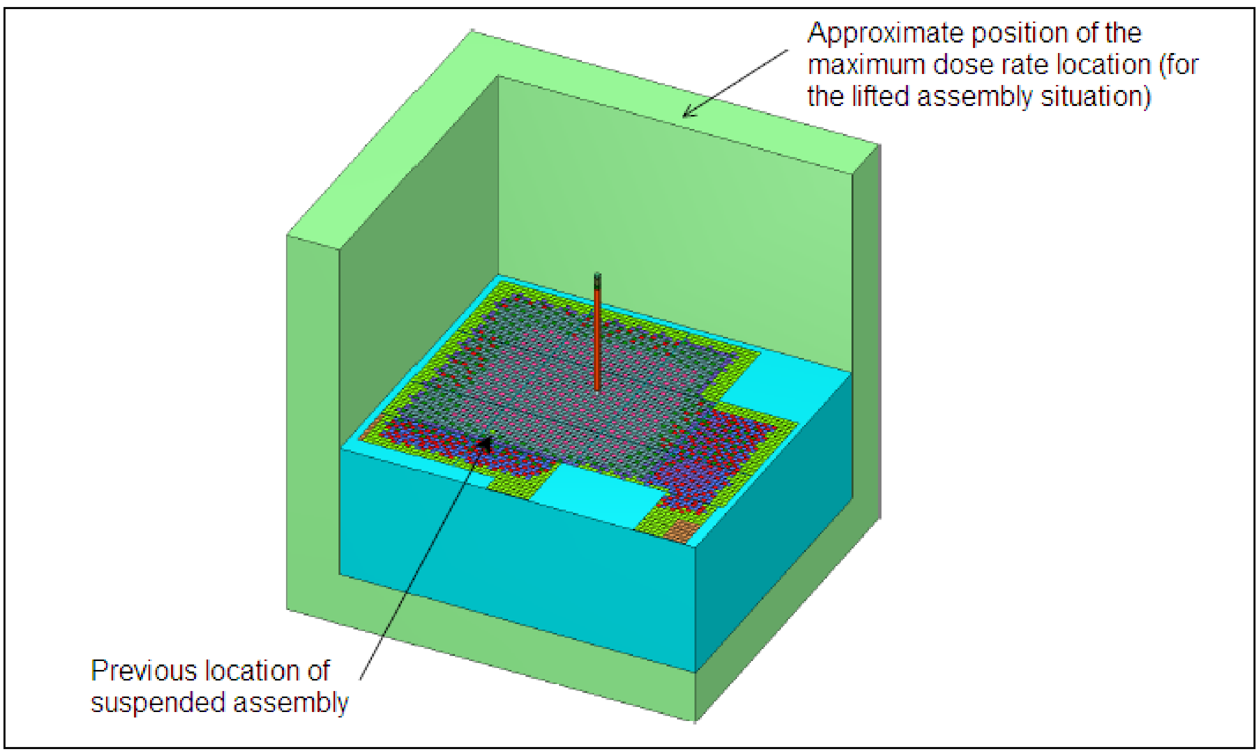

Figure 35 Cutaway depiction of a lifted assembly with water level at the top of the racks

Results of the analyses for the high-density loading situation can be summarized as follows:

- For water depths of 3 m (10 ft) above the top of the racks, projected dose rates are very, very low (less than 0.1 millirem (mrem) per hour). This is consistent with Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," which uses this water depth as a conservative measure of adequate shielding.

- Dose rates for the maximally exposed location on the refueling floor once the water level drops to the top of fuel hardware are very high (on the order of 450 to 600 rem per hour, depending on the OCP).[11]

- At a water depth of 0.6 m (2ft) above the top of the fuel, the projected dose at the maximally exposed location on the refueling floor surpasses 25 rem in one hour. 25 rem is the 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 U.S. Environmental Protection Agency (EPA) 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," issued May 1992)

- Dose rates elsewhere on the refueling floor are significantly lower than those at the maximally exposed location (e.g., see Figure 36).

[11] This range shows that, for situations in which the water level is at the top of the fuel hardware, the dose rates are somewhat sensitive to the time during the operating cycle (a 33-percent decrease in this case). For instances in which water is covering the fuel hardware, this sensitivity decreases. For example, the analogous range of values for a water level 100 cm (3.3 ft) above the fuel hardware is 1.6 to 1.7 rem per hour.

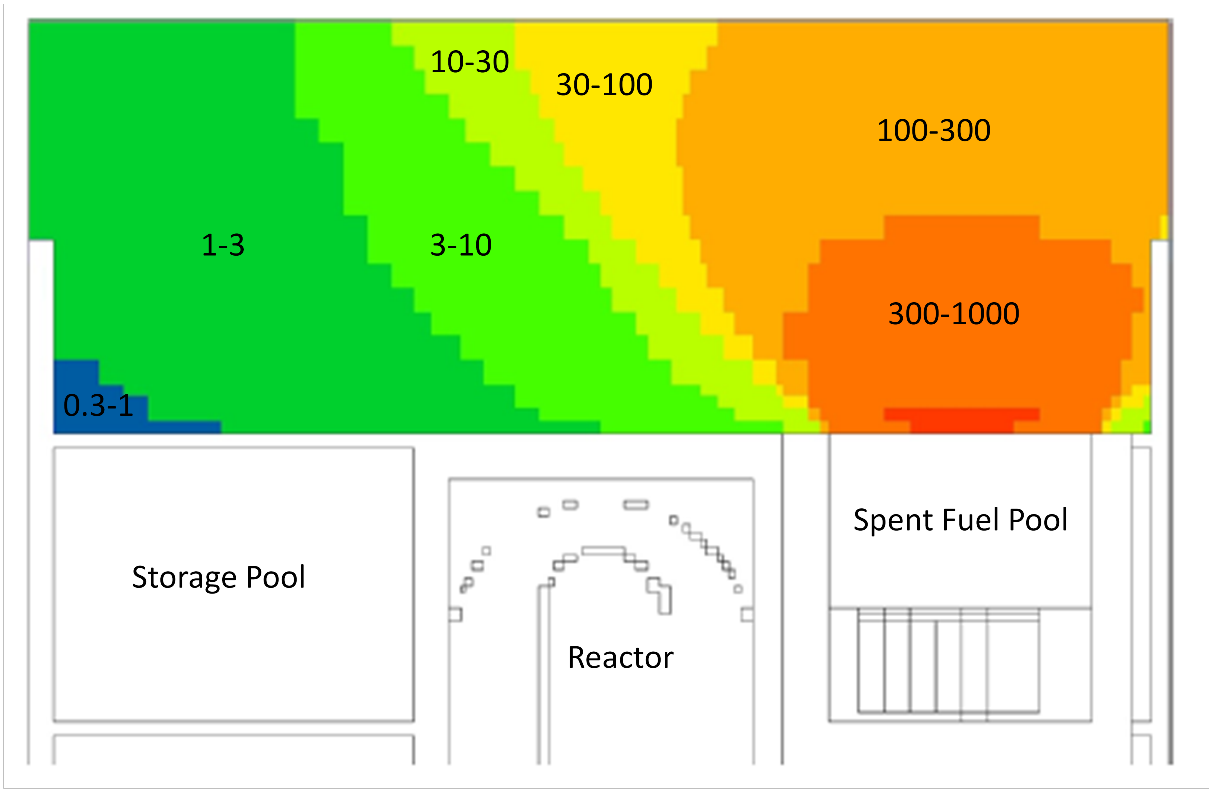

Figure 36 Approximate dose rate of elevation contours, water at top of fuel hardware, around the time of defueling (rem per hour).

Relative to the high-density loading situation, the other situations can be compared as follows:

- For low-density loading situations, dose rates for the maximally exposed location on the refueling floor once the water level drops to the top of fuel hardware are lower than the high-density loading case, but still very high (on the order of 300 to 470 rem per hour, depending on the OCP).

- For a recently discharged assembly in the lifted position, dose rates for the maximally-exposed location are on the order of 3 rem per hour when the water level is at the top of the lifted assembly and over 1,000 rem per hour when the water level is at the top of the racks (i.e., when the lifted assembly is completely exposed). These are dose rate contributions solely from the lifted assembly (i.e., they are in addition to the dose rate contributions from the assemblies in the racks).

- For an older assembly (discharged more than a decade previously) in the lifted position, dose rates for the maximally exposed location are on the order of 0.2 rem per hour when the water level is at the top of the lifted assembly and 7 rem per hour when the water level is at the top of the racks. Again, these are dose rate contributions solely from the lifted assembly.

The high dose rates associated with the single lifted assembly (particularly those for the recently discharged assembly) are sensitive to the assumed position of that assembly. This case assumes that the assembly is located somewhere near the middle of the pool (see

Figure 35), which results in direct line-of-sight from the edge of the SFP. Placement near a wall would reduce the dose rate for locations near the edge of the pool that do not have a direct line of sight to the assembly.

5.5 Discussion of Repair and Recovery

This study makes no attempt 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. 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 an SBO cross-tie line to the Conowingo Dam. The study assumes that the damage sustained by the onsite and offsite electrical distribution systems from the earthquake is enough to significantly delay these recoveries until after the 48- or 72-hour truncation times.

That being said, and as covered previously in this section, the scenarios with successful deployment of mitigation do assume that onsite and offsite resources are able to extend operation of the 10 CFR 50.54(hh)(2) equipment indefinitely, which could represent a situation in which ac power is recovered at an intermediate point and ac-dependent means of SFP makeup are brought back online.

5.6 Scenario Development

5.6.1 Identification of Key Events

The scenario development included the following major assumptions based on the structural analysis documented in Section 4 of this report or other considerations:

- All offsite and onsite ac power is lost as a direct result of the seismic event (see Section 4.2 of this report).

- Direct current power may be lost. Because of the difference from the reactor situation (in which dc power to control turbine-driven systems is important in an SBO), the availability or unavailability of dc power has a much narrower effect. For the specific set of assumptions used in the MELCOR and MACCS2 analyses, there is no effect as analyzed. Chapter 8 further discusses this issue with respect to the HRA.

- The 10 CFR 50.54(hh)(2) equipment (when credited) is available for the duration of the event, following delays associated with diagnosis and deployment (see Section 5.3.1 of this report).

- Initial water loss from “sloshing” will be 0.5 m (1.5 ft) (see Section 4.2 of this report).

- Tearing of the SFP liner is not the most probable outcome, but is possible (see Section 4.1 of this report).

- There is no failure of penetrations, including the refueling transfer canal gate (see Section 4.2 of this report).

- The overhead structures (building debris, crane) do not pose a threat to the SFP in terms of failure resulting from the initiating event (see Section 4.2 of this report).

- Inadvertent criticality, including seismic effects on the integrated poison rack material, is not treated (see Section 2.3 of this report).

5.6.2 Scenario Calculation Matrices

The following table shows how the combinations described thus far translate to the scenarios considered for each OCP.

Table 18 Scenario Breakdown per OCP

Case #	Scenario Characteristics		Radioactive Release Commences before 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		

5.6.3 Summary of Event Split Fractions

As described previously, the analysis considered the available seismic hazard information to obtain an initiating event frequency of approximately one event in 60,000 years for the reference plant.

Table 19 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	5.2×10^{-4}	Damage not expected
2	0.3 to 0.5	0.4	1 in 40,000	2.7×10^{-5}	Damage not expected
3	0.5 to 1.0	0.7	1 in 60,000	1.7×10^{-5}	Damage possible
4	> 1.0	> 1.0	1 in 200,000	4.9×10^{-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 20 Refresher on ac Fragility

Item	Relative Likelihood	Comments
Loss of normal SFP cooling	0.84	This study used the ac power fragility from NUREG-1150 of 0.84 as a surrogate for the conditional probability of normal SFP cooling and makeup not being available. This simplifying assumption was made in light of the fact that the study is not a PRA (but rather a consequence analysis with probabilistic considerations) and that this value already approximates the upper bound of 1. In reality, the availability of normal SFP cooling and makeup would be a combination of the AC fragility, the fragility of the actual equipment and its support equipment, and operator actions to recover the equipment, which could result in a conditional probability higher than the value used here.

As described previously, the structural assessment led to the SFP leakage estimates stated below.

Table 21 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 tens of hours
"Moderate" leakage	0.05	Tearing of SFP liner; damaged concrete limits outflow; drains pool within 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 percent for OCP1 to 66 percent for OCP5 (recall that the OCPs were intentionally "front loaded" because the most change in SFP conditions occurs during the outage).

Table 22 Refresher on the OCP Fractional Contributions

OCP #	Time window (Time of evaluation) (in days)	Fraction of operating cycle	Pool-reactor configuration	Spent fuel configuration for high-density loading
1	2–8 (5)	0.01	Refueling	Dispersed (except for Section 9.3)
2	8–25 (13)	0.02		
3	25–60 (37)	0.05	Unconnected	Dispersed
4	60–240 (107)	0.26		
5	240–700 and 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 do not), so as to assign scenario-specific release frequencies, scenario-specific individual risk of an LCF, or the like. 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 Spent Fuel Pools 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. MELCOR has been developed through the 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 heatup 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 comprises 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. 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. Plant systems and their response to off-normal or accident conditions include the following:

- 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, as well as 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

MELCOR modeling 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 more reasonable execution times. The input structure for MELCOR 2.1 differs completely from that of 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 SFP accidents and (2) it is able to model the accident progression, radionuclide release, and in-building transport/retention. MELCOR 1.8.6 was used in the present study, and the SFP models in both versions of the code (1.8.6 and 2.1) are functionally the same.

As part of NRC's post-9/11 security assessments, the agency developed and applied SFP modeling using detailed thermal-hydraulic and severe accident progression models integrated into the MELCOR code to assess the realistic heatup of spent fuel under various pool draining conditions. SNL performed the analyses for a reference BWR, with additional supporting analyses for separate effects and fluid flow modeling 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 an SFP) and formation and convection of stratified molten pools.

MELCOR 1.8.5 Version RP added two modeling enhancements applicable to BWR SFP modeling (also included in MELCOR 1.8.6 and 2.1): (1) a new rack component, which permits better modeling of an SFP rack and (2) a new oxidation kinetics model. The new BWR SFP 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. These new SFP features can be used to perform two types of SFP calculations: (1) a 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 ruthenium release. A partial loss-of-coolant inventory or boiloff accident could involve no or late uncovery of the bottom of the racks. Boiloff of the coolant leads to steam generation and steam oxidation of the cladding and hydrogen generation that could lead to hydrogen combustion.

Breakaway Oxidation Model

Argonne National Laboratory (ANL) has performed oxidation kinetics testing on zirconium-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 associated with 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 as follows:

$$\frac{dw^2}{dt} = K(T) \tag{1}$$

where, *w* is the reacted metal mass per unit surface area. 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 Zircaloy-4 oxidation correlations are provided below.

The steam preoxidized, wide-temperature, prebreakaway Zircaloy-4 oxidation correlation (Natesan and Soppet, 2004) is as follows:

$$\mathrm{K(T)} = 26.7 \exp\left(-17{,}490/\mathrm{T}\right) \ [\mathrm{kg^2/m^4.s}] \tag{2}$$

The steam preoxidized, wide-temperature, postbreakaway Zircaloy-4 oxidation correlation (Natesan and Soppet, 2004) is as follows:

$$\mathrm{K(T)} = 2.97\mathrm{E4} \exp(-19{,}680/\mathrm{T}) \ [\mathrm{kg^2/m^4.s}] \tag{3}$$

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:

$$\mathrm{LF} = \int_0^t \mathrm{dt'} \frac{\mathrm{t'}}{\tau(\mathrm{T})} \tag{4}$$

$$\tau(\mathrm{T}) = 10^{\mathrm{P_{LOX}}} \tag{5}$$

$$\mathrm{P_{LOX}} = -12.528 \log_{10}\mathrm{T} + 42.038 \tag{6}$$

where P_{LOX} is the MELCOR fit of the timing for the transition from prebreakaway to postbreakaway oxidation reaction kinetics for Zirlo and Zircaloy-4 in the ANL experiments.

The air oxidation model was benchmarked against experimental data from the SNL SFP facility as part of the security assessment work. 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 postbreakaway oxidation kinetics.

Hydraulic Resistance Model

The MELCOR modeling approach for flowpaths 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 flowpath 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:

$$\Delta\mathrm{P} = \frac{1}{2}\rho\, \mathrm{v}^2 \left(\mathrm{f}\frac{\mathrm{L}}{\mathrm{D}} + \mathrm{K}\right) \tag{7}$$

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:

$$f = S_{LAM}/Re \tag{8}$$

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 9x9 BWR assembly at SNL (Durbin, 2005) to obtain the required frictional and form loss coefficients, including the effects of grid spacer and partial rods. The present study used these measurements given a lack of data for a 10x10 BWR assembly.

6.1.2 Heat Transfer Modeling within Spent Fuel Pool and to Surrounding Walls

The MELCOR core models calculate the thermal response of the core.[12] 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 SFP 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 preexisting 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 intracell radiation or conduction, but the user is responsible for defining a single input parameter that captures the geometry of the heat transfer path. Figure 37 depicts the heat transfer paths within a ring and across a ring boundary. For radiation between different core rings, the user adjusts the view factors and the surface areas.

The core models radiative heat transfer from the outermost ring components (if present) to the core boundary specified as a heat structure. The SFP 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.

[12] MELCOR core models were originally designed for the reactor core. Because of the code flexibility, the same modeling approach can be used for the spent fuel pool (with the addition of the rack as a separate component). Therefore, as far as code models are concerned (e.g., heat transfer between groups of assemblies and with the fluid, and radionuclide release, transport and deposition), there is no difference between reactor assemblies and spent fuel assemblies. It is up to the user to define the propoer information in the input deck.

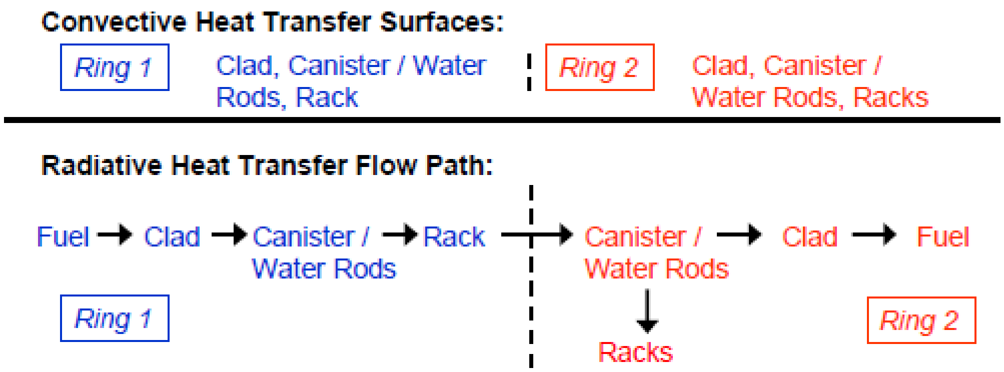

Figure 37 MELCOR modeling of heat transfer paths

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 atmosphere and includes droplet heat and mass transfer and fission product removal capabilities. All calculations used a droplet size of 1,250 microns. The spray was positioned at the top of the SFP (elevation of the refueling bay), thus allowing the droplets to be directed into the assemblies and open spaces based on their respective cross-sectional areas.

The interphase momentum model, which replicates the Wallis flooding curve, controls the penetration of the spray water into the assembly. 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 38). 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 percent (i.e., $\alpha > 0.998$). Because of 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.

MELCOR thermal-hydraulic model interprets the liquid film as a small pool at the bottom of each control volume (see Figure 38). Because of 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 which permits a better local representation of the fluid conditions and the location of the spray dryout. Parametric calculations are performed to show the impact of this modeling parameter (i.e., flow regime model active or inactive as discussed in 6.3.1).

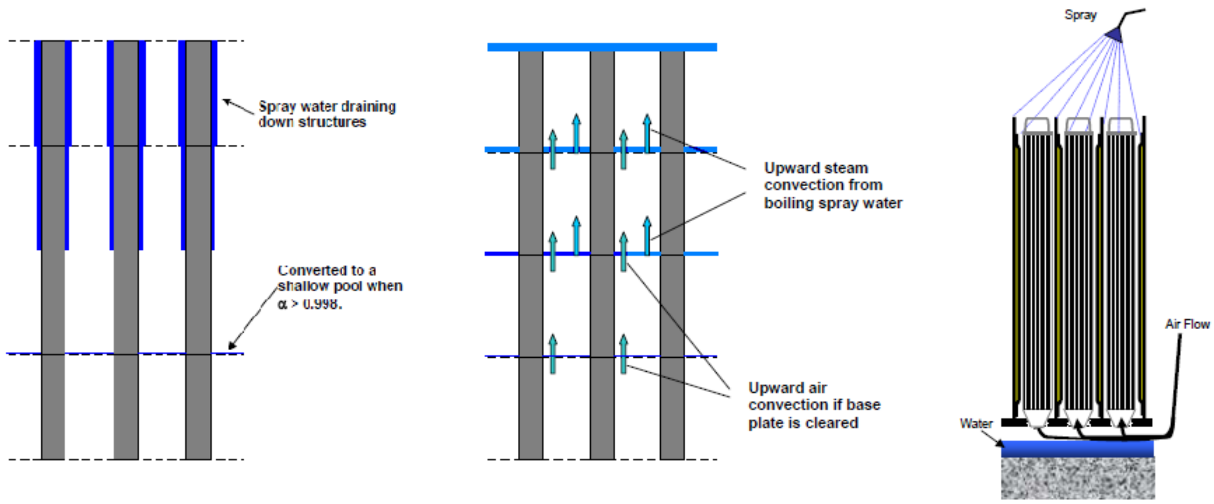

Figure 38 Spray model for SFP analysis

6.1.4 Modeling of Fuel Collapse and Baseplate Failure

Fuel collapse is based on user-defined cumulative fuel damage fraction logic, in which 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 are 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 and 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 RN package models the release and transport of fission product vapors and aerosols (referred to as radionuclides). Release of radionuclides can occur from the fuel-cladding gap by exceeding a failure temperature criterion or losing intact geometry or 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 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 nonradioactive mass to properly model the transport of fission products. The SFP MELCOR model includes 15 default material classes and two user-defined classes to model the behavior of cesium iodide (CsI) and cesium molybdate (Cs_2MoO_4), as shown in Table 23.

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 iodine with the default inventory wholly transferred to Class 16.
- Class 7—Characteristic released compound is molybdenum 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 Cs_2MoO_4. The released vapor pressure and compound mass is consistent with Cs_2MoO_4.

Gauntt (2010) proposes an approach for the estimation of increased ruthenium release under air-oxidation conditions. Ruthenium (Class 6) has the lowest of vapor pressures in the default MELCOR model that prevents prediction of large releases.[13] 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 hyperstoichiometric fuel ($UO_{2.15}$) and release of ruthenium dioxide (RuO_2). The default vapor pressure parameters in MELCOR are adjusted for the ruthenium class to match RuO_2 vapor pressure at 2200 K.[14] The new ruthenium release model is applied only to scenarios involving rapid draindown (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.

Table 23 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 Molybdate	Cs_2MoO_4	Cs_2MoO_4

The gap inventory specified in Table 24 is based on NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” issued February 1995 (NRC, 1995). However, NUREG-1465 states that, for accidents in which long-term cooling is maintained (e.g., postulated spent fuel handling accident), the gap release could be as low as 3 percent, and, in the unmitigated scenarios in this study, the fuel experiences prolonged high temperatures (and even failure in some instances). Therefore, the present work assumes that 5 percent applies to all scenarios.

13 There is a mass transfer limitation to the release from the fuel.

14 The rationale for an increased ruthenium class release is based only on increased vapor pressure and requires further experimental validation.

The decay heat calculation was based on security assessment analyses 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 burnup and power history. The utility provided the program and the appropriate input files for the SFP configuration after its last offload (i.e., September 2001) to perform decay heat calculations. Consequently, the decay heat power of every assembly was calculated as a function of time from reactor shutdown.[15] 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.

Table 24 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 burnup. The effective operating power was calculated as the total burnup of all assemblies in the SFP (gigawatt days per metric tons of uranium) divided by the average assembly metric tons of uranium and the total number of days of criticality. Based on the effective operating power, MELCOR calculates the specific time-dependent decay heat and mass inventory for each element. 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 (kilogram (kg)) times the specific element decay heat (watts per kilogram) 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 postprocessing routine is implemented that uses the MELCOR predicted release fractions along with actual inventories calculated for each batch.

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15 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 percent). Since the number of old assemblies was increased by 60 (3,055 total in the pool), the decay heat for these assemblies was assumed to be an average of the older assemblies.

To accommodate consequence calculations using MACCS2, an extensive control system was written in the MELCOR input file that tracks the fission product releases from each ring[16] and the subsequent release to the environment. Time-dependent, nondimensional 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 MACCS2. MELCOR activity release for each isotope (e.g., m = Cs-137, Cs-134, Cs-136 for Class 2) is given by the following:

$$A_m(t) = \sum_{r=1}^{6}[RF_{m,r}(t) \times A^0_{m,r}] \tag{9}$$

MACCS activity release is given by the following:

$$\tilde{A}_m(t) = \widetilde{RF}(t) \times \sum_{r=1}^{6} A^0_{m,r} \tag{10}$$

where is defined as:

$$\widetilde{RF}(t) \times A^0_1 + \ldots + \widetilde{RF}(t) \times A^0_M = \sum_{m=1}^{M} A_m(t) \tag{11}$$

$$\widetilde{RF}(t) = \frac{\sum_{m=1}^{M} A_m(t)}{\sum_{m=1}^{M} A^o{}_m} = \frac{\sum_{m=1}^{M} A_m(t)}{\sum_{m=1}^{M}\sum_{r=1}^{6} A^0_{m,r}} \tag{12}$$

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 MACCS2)
A = released activity (Becquerel (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 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 identification, design type, initial enrichment, discharge burnup, and discharge date. The analysis basis for the high-density SFP inventory was 3,055 assemblies, a number based on the pool capacity of 3,819 assemblies, reduced by 764 assemblies to accommodate a full core offload capability. Information on

[16] A ring is a collection of assemblies in the MELCOR radial nodalization.

assemblies discharged before 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 et al., 2011), maintained within the SCALE nuclear safety analysis code system (Rearden et al., 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 7x7, 8x8, 9x9, and 10x10 General Electric assembly designs (Ilas et al., 2006) used in the reference plant 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 and Gauld, 2008) to be accurate within plus or minus 2 percent.

For the burnup analysis, the irradiation and decay history for each of the 3,055 assemblies in the pool was simulated 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, 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 (1,683 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 5 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.[17] Therefore, the actual inventories from batches were scaled appropriately to correspond to the rings in the 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 SFP results were compiled for each assembly group and all decay times and included activities (Bq) for 69 radionuclides and decay heat.[18]

Results from the present analysis were compared with those generated previously for the reference plant pool using assembly data provided by the utility through 2001 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

[17] The nodalization was based on the security assessment work. The additional data on later cycles were received after the MELCOR model had been developed and the calculations were started.

[18] The decay heat in the present analysis is based on the past security assessment work.

average discharge burnup increased significantly after Cycle 12. The burnup values used in the present analysis are significantly higher and therefore more representative of modern SFP inventories than earlier analyses. Previous analyses using the 2001 data are representative of discharged fuel up to about 1995.

Other differences are attributed to the specific power of the assemblies which influences the decay heat power and activities of short-lived fission products in the analysis time range. The utility did not provide information on the specific power. Notwithstanding power uprates for the reference plant reactor, the most recent occurring in 2002, the specific power used to calculate inventories for the assemblies in the present analysis was lower than that 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 percent larger than the current results, likely the result of 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 percent over all cooling times, with present results slightly larger than utility values, most likely because of the increase in discharge burnup since 2001.

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 an SFP calculation, ignition is assumed to occur in the reactor building when the hydrogen concentration exceeds 10 percent by volume. In addition, MELCOR checks to determine whether there is sufficient oxygen. The minimum oxygen mole fraction for ignition is 5 percent. The maximum diluents mole fraction for ignition (mole fraction of steam plus mole fraction of carbon dioxide) is 55 percent. If all of these conditions are satisfied, a burn is initiated. Some uncertainty may exist regarding the combustion of hydrogen, especially with regard to the timing of a spontaneous ignition. 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 in which combustion was very likely or very unlikely. Consequently, the SFPS presents the results of cases with and without combustion. However, some cases have conditions in which the occurrence or timing of a combustion event has more uncertainty; these cases were assumed to ignite or not ignite according to the default spontaneous combustion criteria in MELCOR (see Section 9.1). Once a burn is initiated, it can propagate to other control volumes using the default hydrogen concentrations of 4 percent, 6 percent, and 9 percent for upward, horizontal, and downward propagations, respectively.

6.2 Description of MELCOR Models

The SFP, 40 ft (12.2 m) wide by 35.3 ft (10.8 m) long by 38.75 ft (11.8 m) 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-in.- (0.63-cm-) thick stainless steel. The walls and the floor of the

SFP are approximately 6 ft (1.83 m) thick. In the northeast corner of the SFP is a cask area that is 10 square feet (ft^2) (0.93 square meter (m^2)).

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 (7 m) of water for radiation shielding. The SFP racks are freestanding, full length, and top entry and are designed to maintain the spent fuel in a spaced geometry that 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 3,819 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 baseplate, and base support assemblies. Each cell is composed of (1) a full-length enclosure constructed of 0.075-in.- (0.2-cm-) thick stainless steel, (2) sections of Bisco Boraflex, which is a neutron-absorbing material, and (3) wrapper plates constructed of 0.020-in.- (0.05-cm-) thick stainless steel. The inside square dimension of a cell enclosure is 6.07 in. (0.15 m). The cell pitch is 6.28 in. (0.16 m). The baseplate is made from 0.5-in.- (1.27-cm-) thick stainless steel with 3.8 in. (0.1 m) 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 (see Figure 39). 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-in. 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 39 shows the layout of the rack cells. There is the potential for lateral cell-to-cell flow between connected rack cells.

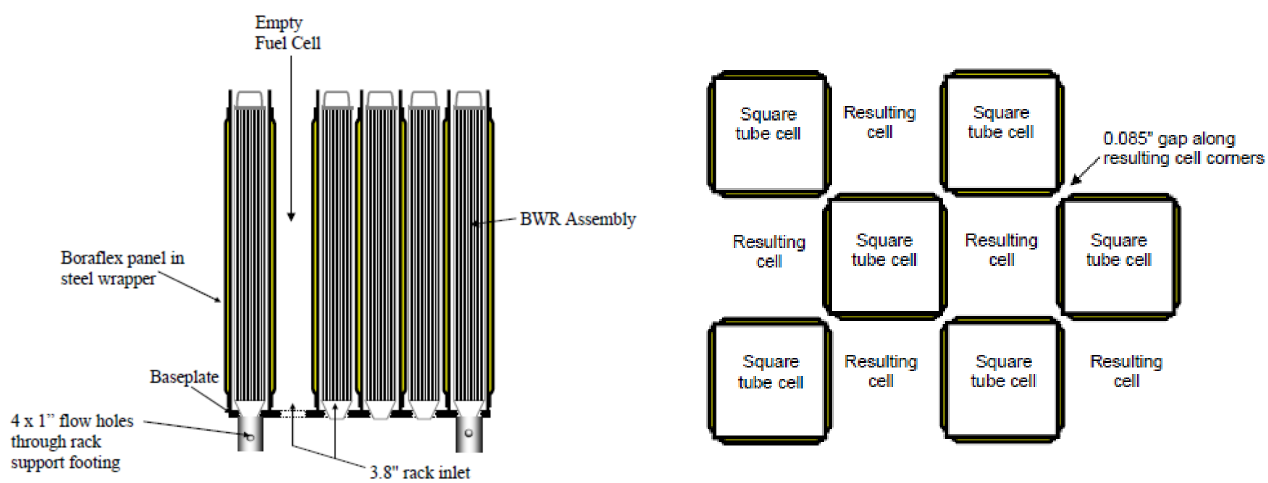

Figure 39 Typical SFP rack cut away cross sections

Figure 40 shows the control volume nodalization of the SFP region of the whole pool model. The bottom of the pool was divided into eight regions. CV299 represents all open regions in the SFP around the racks, including the cask area. The racks are subdivided into the other seven regions. Ring 7 (CV170 and CV171) represents the empty rack cells on the periphery of the SFP. All of the assemblies in the SFP are located in Rings 1 through 6. Each ring with assemblies is further subdivided into 19 control volumes—one control volume below the racks, nine control volumes inside the canister, and nine control volumes 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 41). Similarly, 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 control volume nodalization does not contain a bypass region (between the channel box and rack) as shown on the right side of Figure 41.

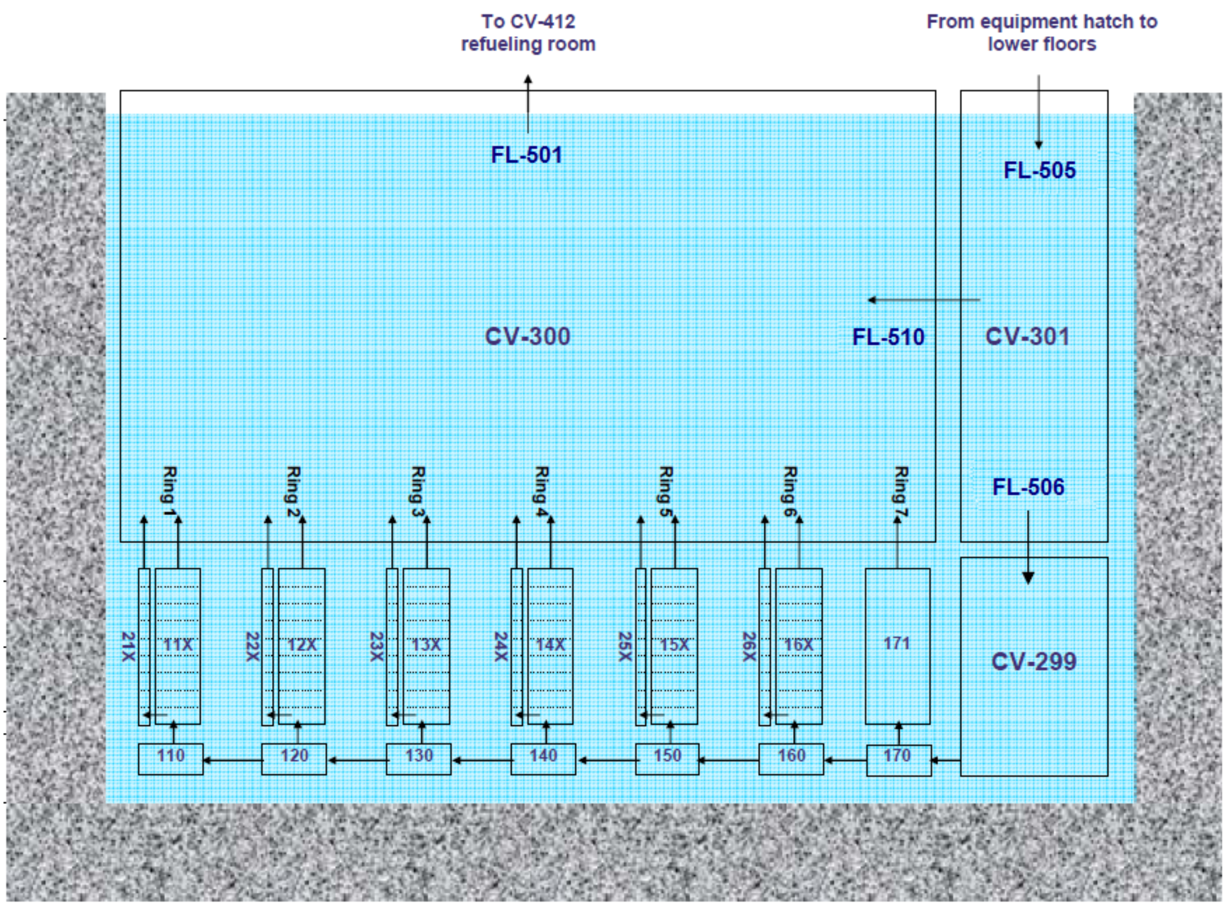

Figure 40 MELCOR nodalization of the whole pool high density model

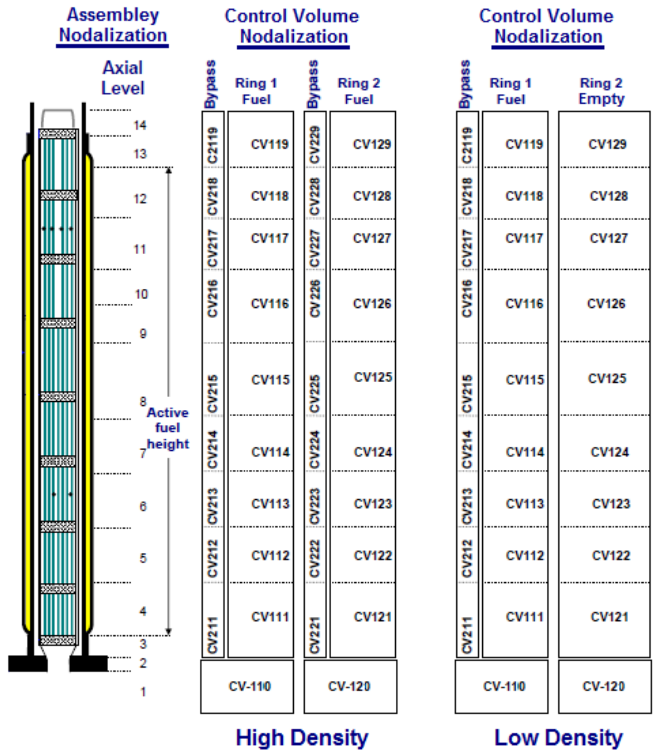

Figure 41 MELCOR nodalization of the assemblies (only two rings shown)

The hydraulic resistance was specified using the results from the SNL experimental test program (Durbin, 2005).[19] For example, for the flowpath connecting CV113 and CV114 in the fully populated region, the MELCOR input values included a form loss coefficient of 3.8, and a friction factor (S_{LAM}) of 31.3 (equal to 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 flowpaths. 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 flowpaths between the bypass region and the assembly. Initially, the canister wall precludes flow. However, if the canister fails, a radial flowpath 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.

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19 In the present study, the assembly nodalization is based on the GE14C 10x10 configuration (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 percent compared to the 9x9 design. The hydraulic resistance data are assumed to apply. The frictional loss coefficient for a 10x10 array could be somewhat different since it is a function of hydraulic diameter and grid spaces design.

A complete reactor building has been developed for the reference plant (NRC, 2012d). However, the bulk of the reactor building does not play a significant role in SFP accidents, given that the study does not explicitly model (1) the effect of the SFP accident on reactor systems or (2) specific obstacles to deploying mitigation (e.g., presence of steam on lower elevations). Consequently, the reactor building model was simplified to only model the refueling room (i.e., within the red dashed line in Figure 42).

A single control volume models the refueling bay. An open hatch in the southeast quadrant connects (via a flowpath) 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 overpressure failure flowpaths within the reactor building. The simplified refueling floor model included the two most important flowpaths—(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 overpressure greater than 1,720 pascal (Pa) (0.25 pounds per square inch gauge (psig)). If the reactor building pressure rises above 3,450 Pa (0.5 psig), failure of the roof decking 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 when 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 overpredict the amount of thermal mixing within the building. Based on insights from the computational fluid dynamics calculations for the security assessment work, 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 40) comes directly from the hatch region (see left side of Figure 42). 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 the computational fluid dynamics calculations.

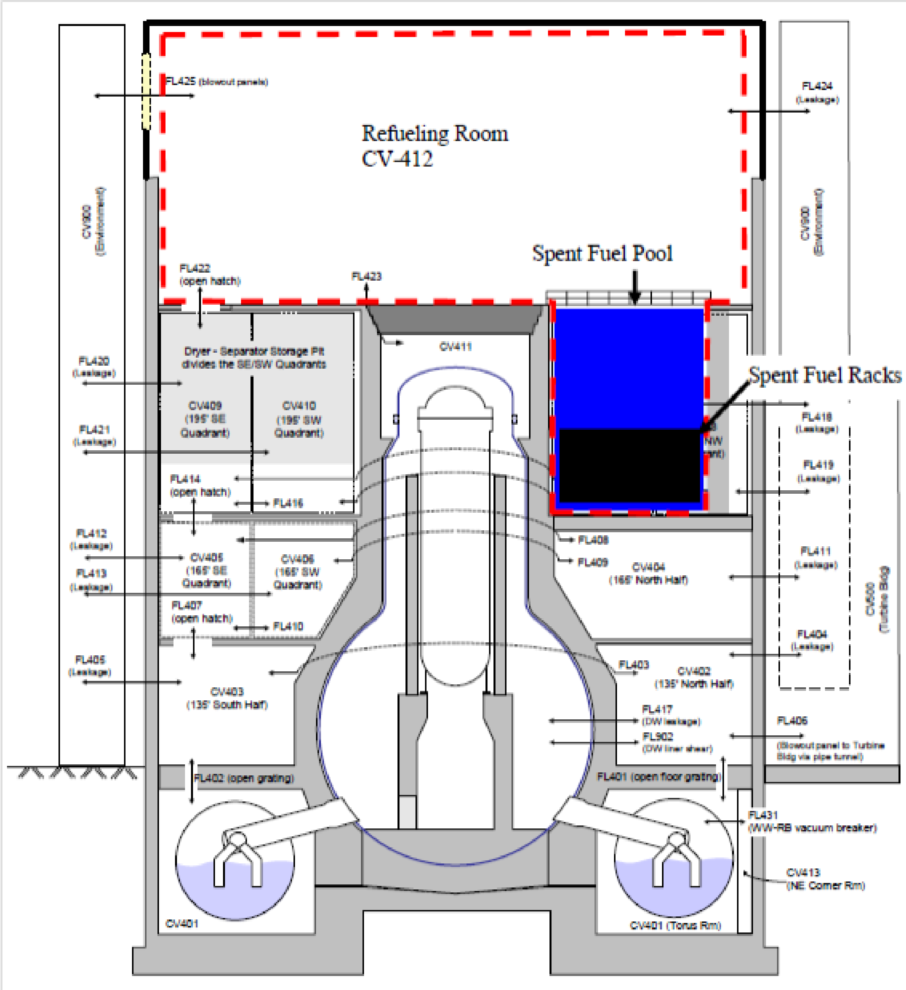

Figure 42 MELCOR reactor building model

6.2.1 High-Density Loading during Outage

During an outage in which the SFP and reactor are hydraulically connected, a single control volume is used to represent both the reactor well and separator/dryer pool, as shown in Figure 43. The total volume of pool in CV601 is about 1,900 m^3 (neglecting the dead-end pool volume of 243 m^3 below the separator/dryer gate elevation). CV601 is hydraulically connected to CV300 (see Figure 40) using two flowpaths until the water level reaches the SFP gate and no more water can flow into 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 1,400 m^3.

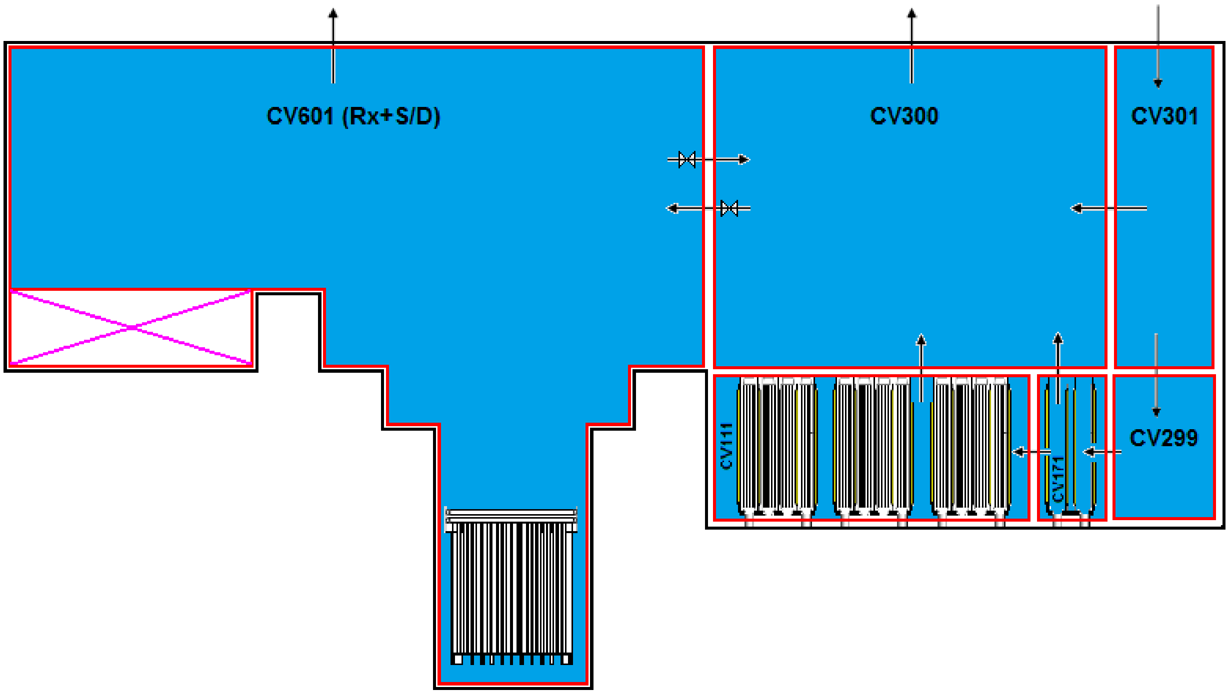

Figure 43 SFP and reactor connection model during outage

For both OCP1 (at 4 days) and OCP2 (at 13 days), CV601 is connected to the detailed model of the SFP (
Figure 40). Figure 44 shows the assembly layout for OCP1 in a 1x4 pattern in which the assemblies are grouped into six 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.[20] Ring 3 is empty during the outage where the assemblies still reside in the reactor.[21] Ring 5 contains the last offload (284 assemblies) with an additional 31 assemblies from previous offloads. Rings 2, 4, and 6 have a total of 2,456 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 41) without the bypass control volume. This will ensure a better representation of flow through the assemblies and modeling of heat transfer between components in various rings.

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20 All of the old assemblies are smeared in MELCOR Rings 2, 4, and 6 (i.e., decay power per assembly is the same).

21 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.

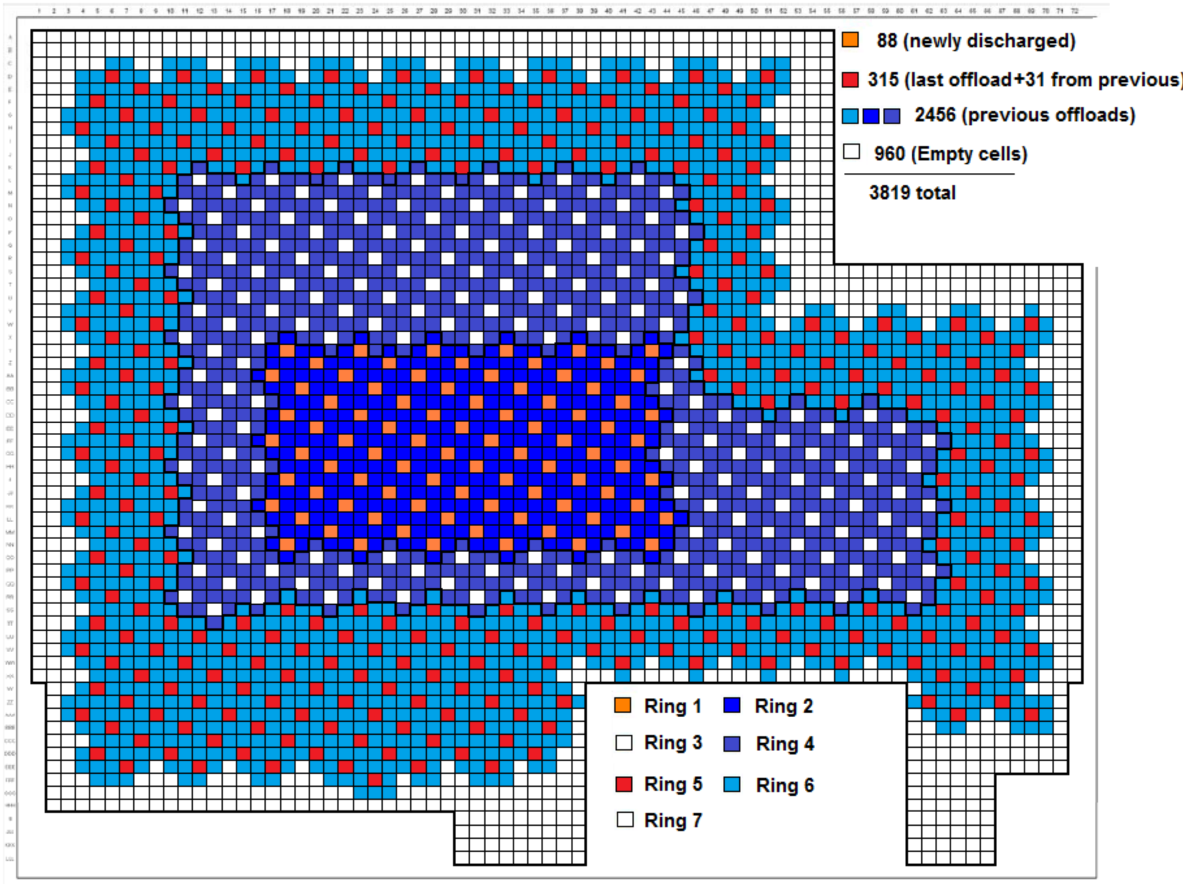

Figure 44 Layout of assemblies for OCP1 high density (1x4) model

Figure 45 shows the cell-wall radiation view factors between the various rings.[22] 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 44, 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, Rings 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.[23] In OCP2, the 196 assemblies have been moved to Ring 3, as shown in Figure 46, and the radial thermal coupling is preserved as in Figure 45.

22 MELCOR models intracell radiation between concentric rings by default. To disable the radiation model for Rings 2 to 3 and 4 to 5, the radial view factor area is set to zero.

23 The view factor is assumed to be unity. It should be noted that there is a temperature gradient within each ring, and MELCOR attempts to model a multidimensional geometry with a simplified two-surface radiation model.

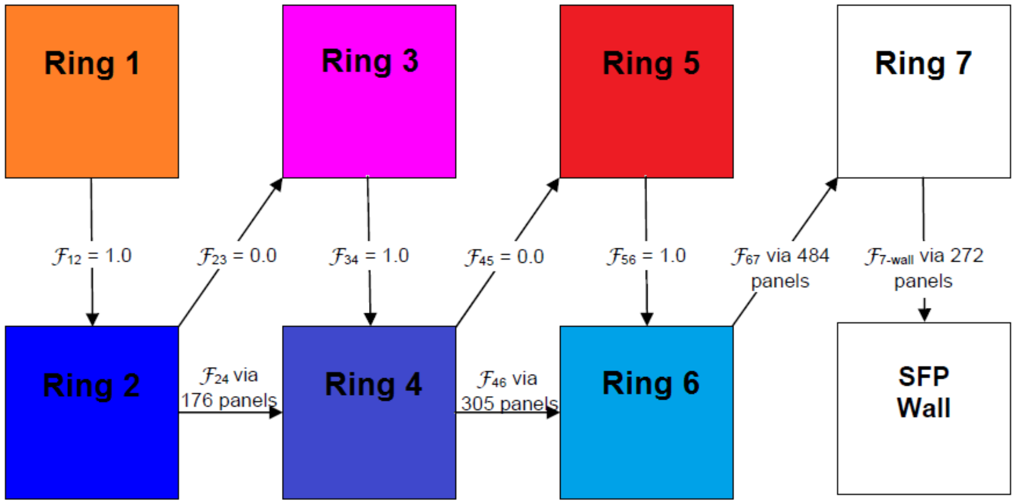

Figure 45 MELCOR radial radiative coupling scheme

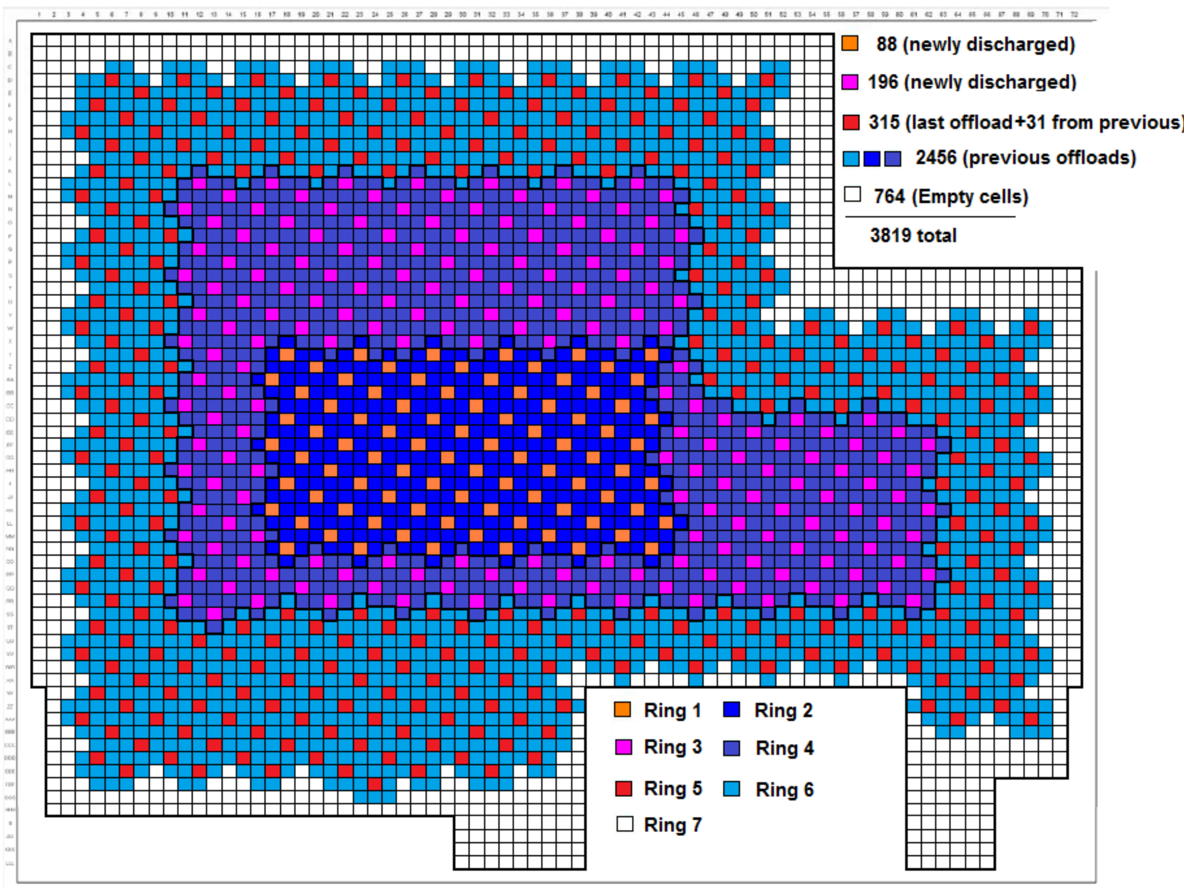

Figure 46 Layout of assemblies for OCP2 high-density (1x4) model

The methodology described in Section 6.1.5 was used to estimate the decay heat power as a function of time for different OCPs. Table 25 shows the results of this analysis. The reactor power was based on the decay power for all assemblies residing in the reference plant reactor (NRC, 2012d) by subtracting the power associated with assemblies that have already been moved to the SFP. For example, for OCP1, the analysis assumed that 88 assemblies are already in the SFP.

Table 25 Distribution of Decay Heat in the Reactor and SFP for High Density Loading

	Reactor (kW)	Spent Fuel Pool (kW)							
		Days	Ring 1 (88)[1]	Ring 3 (0)	Ring 5 (315)	Ring 2 (352)	Ring 4 (784)	Ring 6 (1,320)	Total (2,859)
OCP1	10,216	3.6	1,927	0	465	80	179	301	2,951
	9,915	3.9	1,867	0	452	80	179	301	2,878
	9,006	5.0	1,690	0	417	80	178	300	2,666
	7,406	8.0	1,403	0	358	80	178	300	2,320
	6,710	10.0	1,282	0	334	80	178	300	2,174
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (315)	Ring 2 (352)	Ring 4 (784)	Ring 6 (1,320)	Total (3,055)
OCP2	4,395	13.1	1,144	1,533	332	80	178	300	3,567
	4,117	15.0	1,077	1,444	330	80	178	299	3,409
	3,530	20.0	957	1,294	318	79	176	296	3,120
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (315)	Ring 2 (352)	Ring 4 (784)	Ring 6 (1,320)	Total (3,055)
OCP3		37	720	973	324	79	177	298	2,571
OCP4		107	422	602	301	78	173	292	1,868
OCP5		383	191	315	230	73	162	273	1,245

1. The numbers in parentheses are the number of assemblies.

6.2.2 High-Density Loading Postoutage

The layout for the postoutage high-density loading is similar to OCP2 (see Figure 46). In postoutage, the assemblies are assumed to be in a 1x4 pattern, which applies to OCP3, OCP4, and OCP5. The assembly layout remained constant for these OCPs. However, the decay heat decreased from OCP3 to OCP5 as the aging time since reactor shutdown increased. Table 25 summarizes the decay heat power in each ring.

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 (equal to 284 × 3). 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 47 shows the layout of assemblies in the SFP for OCP1, and Figure 48 shows the layout for OCP2. For both configurations, all of the old fuel has been removed from the pool, and the current offload is in a 1x4 pattern with empties. Because of space limitations, the last two offloads are placed in a checkerboard pattern.[24] For the axial nodalization, Ring 1 contains both the channel (inside the

[24] There is not enough room to place all of the fuel in a 1x4 pattern. The current offload eventually requires 1,420 cells (284 for assemblies and 284 × 4 for empties surrounding them), which would leave only 1,635 cells (excluding Ring 7). The 568 assemblies would require 2,840 cells for storage in a 1x4 pattern.

canister) and the bypass (outside between canister and rack) control volumes, while both volumes are combined for Ring 2 (see Figure 41). The basic radial thermal coupling from Figure 45 still applies, but the boundary area from Ring 6 to Ring 7 is 472 panels. For modeling convenience, Rings 2, 4, and 6 from the high-density layout are still present, but the cells contain only the rack component.

Table 26 provides the distribution of decay heat in the pool. A comparison with the high-density decay heat shows that the total decay heat in the pool for the low-density case is reduced by less than 20 percent. 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 zirconium fire because of reduced mass in the empty assemblies.

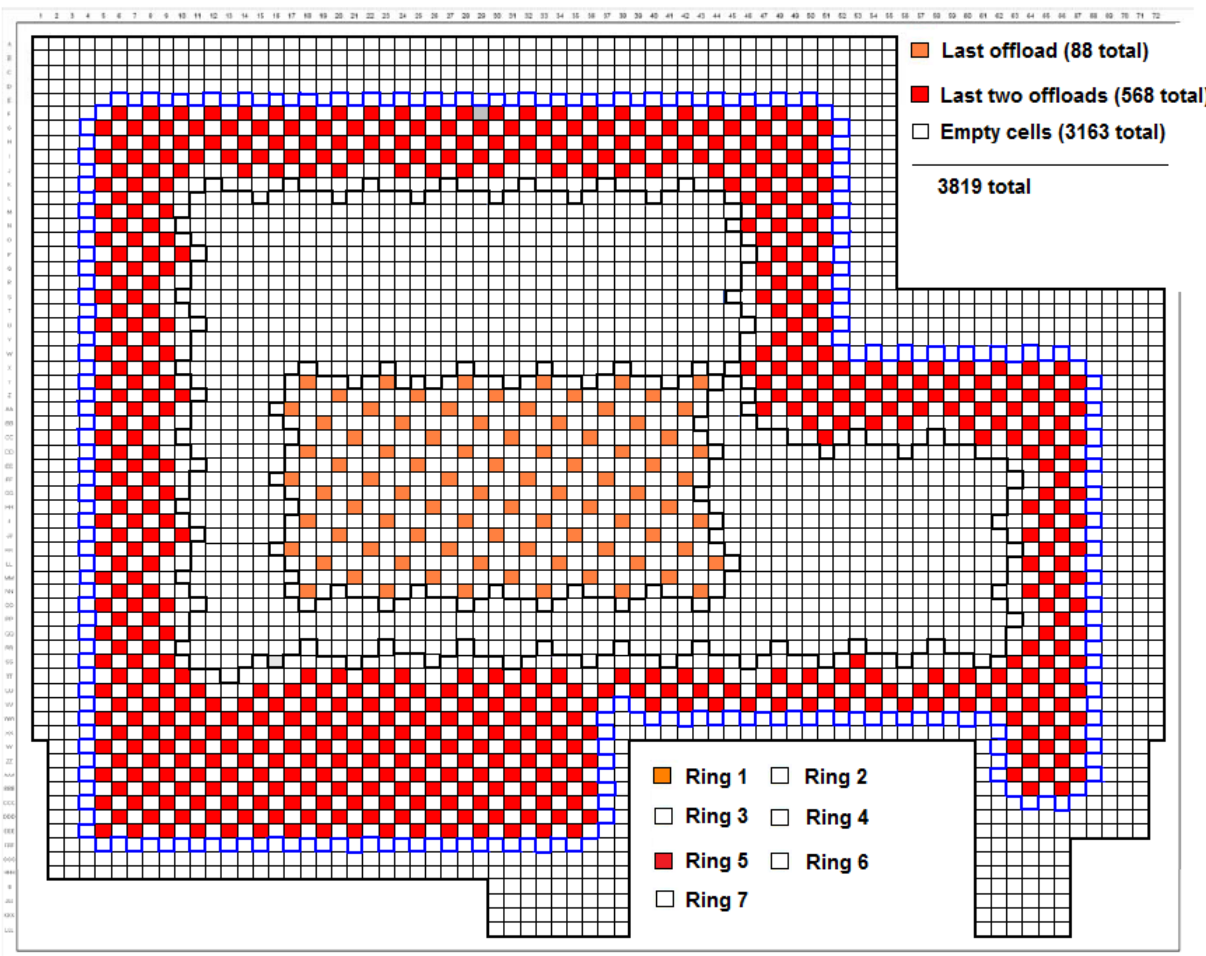

Figure 47 Layout of assemblies for OCP1 low-density model

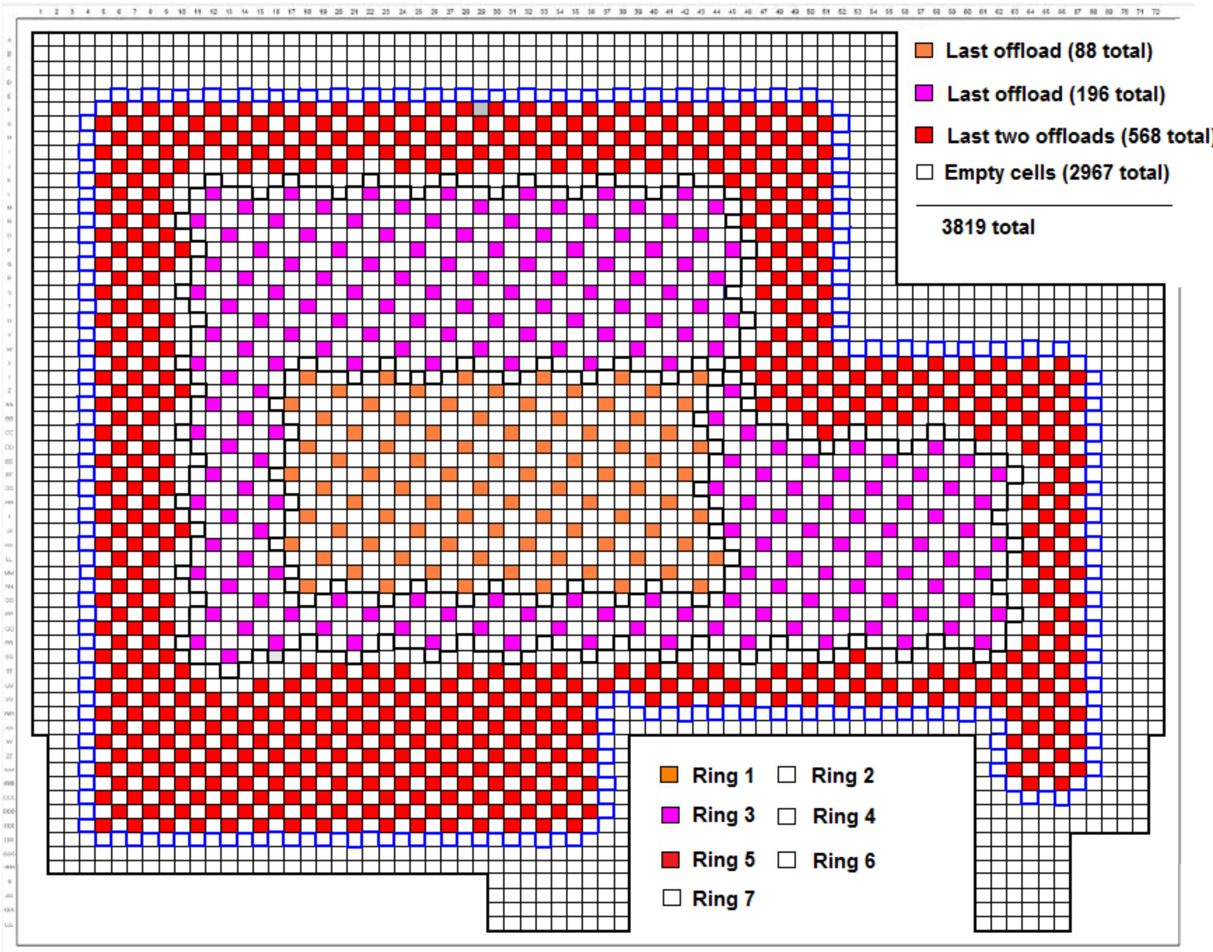

Figure 48 Layout of assemblies for OCP2 low-density model

6.2.4 Low-Density Loading Postoutage

The postoutage low-density layout for OCP3, OCP4, and OCP5 is identical to OCP2 (see Figure 48), and the pool decay heat is provided in Table 26.

Table 26 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)
OCP1	10,216	3.6	1,927	0	599	0	0	0	2,526
	9,915	3.9	1,867	0	587	0	0	0	2,454
	9,006	5.0	1,690	0	551	0	0	0	2,241
	7,406	8.0	1,403	0	492	0	0	0	1,895
	6,710	10.0	1,282	0	468	0	0	0	1,750
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (568)	Ring 2 (0)	Ring 4 (0)	Ring 6 (0)	Total (852)
OCP2	4,395	13.1	1,144	1,533	466	0	0	0	3,143
	4,117	15.0	1,077	1,444	464	0	0	0	2,985
	3,530	20.0	957	1,294	448	0	0	0	2,699
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (568)	Ring 2 (0)	Ring 4 (0)	Ring 6 (0)	Total (852)
OCP3		37	720	973	455	0	0	0	2,149
OCP4		107	422	602	427	0	0	0	1,451
OCP5		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, the following four classes of scenarios do not result in a release from the fuel:

1. boiloff scenarios with no SFP leaks
2. mitigated scenarios for small leaks
3. unmitigated scenarios in late phases (OCP4, OCP5)
4. mitigated moderate leak scenarios in OCP2, OCP3, OCP4, and OCP5

Boiloff

For the boiloff scenarios, a simplified model was used to estimate the pool heatup and water level drop. Figure 43 shows this model in which all of 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 25 for high-density cases or Table 26 for low-density cases. The top of the pool is connected to the reactor building (see Figure 42) in the same manner as in the detailed model. This simplified model is used as a screening tool to determine whether more detailed analysis is needed. Figure 49 shows the water level as a function of time for both high- and low-density cases for

OCP1, OCP2, OCP3, and OCP4.[25] Figure 49 also identifies the time required to reach pool saturation. For cases in the same OCP, the high-density cases become saturated sooner since there is less water volume and more decay heat. In the late OCPs following refueling, the difference in the timing directly correlates to the decay heat power. While there are differences in postsaturation water level for OCP3 and OCP4, the water level for OCP1 and OCP2 is similar as a result of mixing assumed between the reactor well water and the SFP water (see Figure 43). For the OCP4 low-density and OCP5 cases, the SFP never becomes saturated in 72 hours. The slight water level increase during the sensible heating period results from the change in pool density as the water heats up. The analysis shows that there is 4.6 m (15 ft) of water above the top of racks in OCP1 at 72 hours.

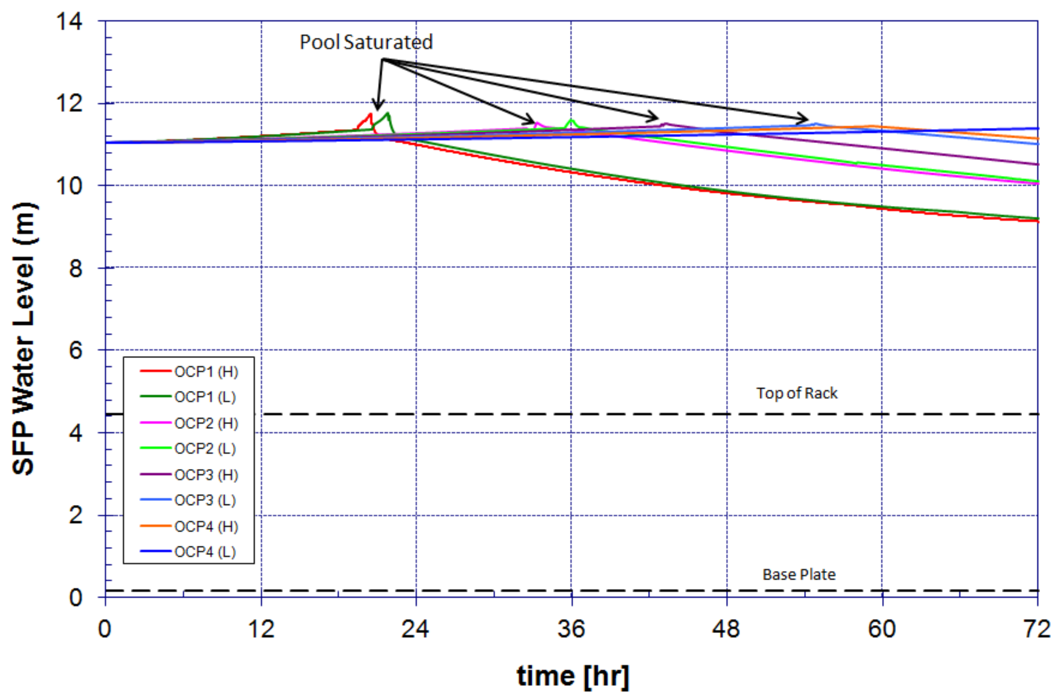

Figure 49 Water level for boiloff scenarios

Mitigated Scenarios (Small Leaks)

The small leak is modeled in MELCOR with a 4.4 cm (1.75-in.) 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). Figure 50 and Figure 51, respectively, show the water level and the injection and leak mass flow rates for the low-density OCP1 case. 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 case, mitigation is direct makeup to the pool

25 The initial water level is assumed to be 11 m. The initial water temperature is 82 degrees F (28 degrees C). Both these initial conditions are applied to all accident scenarios in this report. Based on a teleconference with the licensee held on April 24, 2012, this is the postoutage water temperature under steady-state conditions where the heat exchangers are working (prior to postulated accident). During an outage (OCP1 and OCP2), the water temperature could vary between approximately 80 degrees F and 100 degrees F. The higher temperature affects the sensible heating of the pool and is not expected to change the overall conclusion of boiloff scenarios given the significant margin observed.

(injection) since the water level at the time of deployment is more than 1 m above the top of the racks. For this small leak, the initial water flow rate is about 250 gpm (0.016 cubic meters per second (m^3/s)), 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 leak 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 configuration as long as the water level remains above the top of the racks.

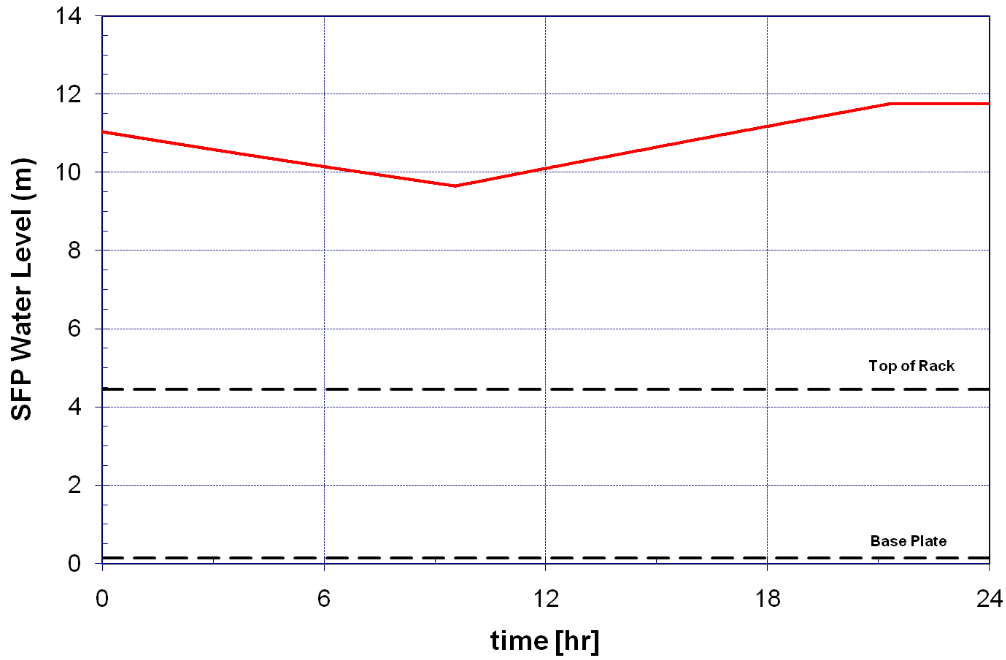

Figure 50 Water level for mitigated low-density OCP1 (small leak) scenario

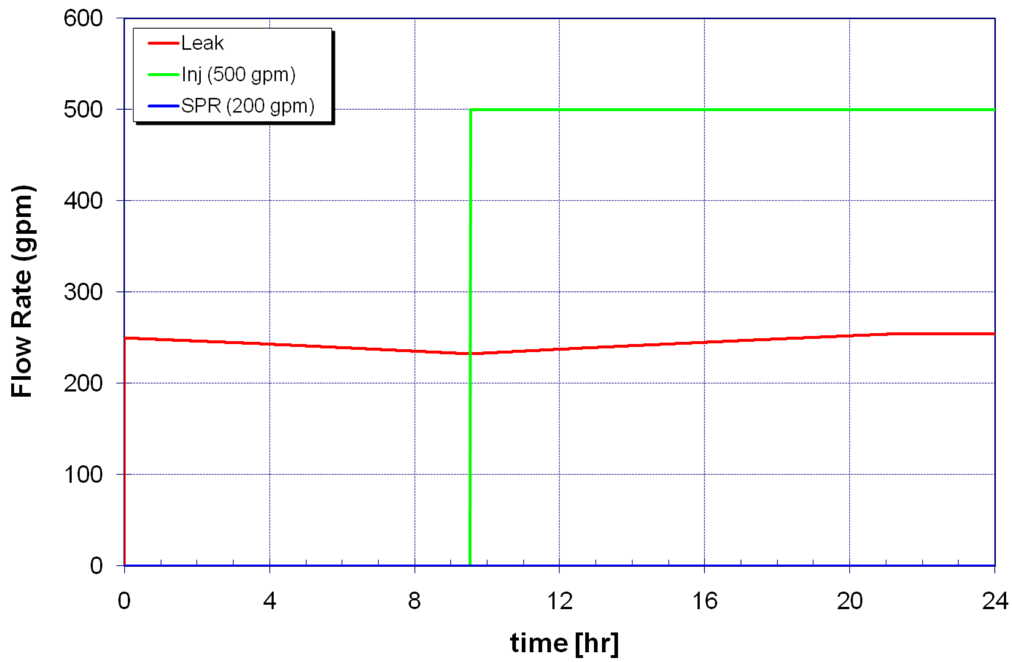

Figure 51 Flow rates for mitigated low-density OCP1 (small leak) scenario

Unsuccessful Deployment of Mitigation for OCP4 and OCP5 Scenarios

For OCP4, the decay heat is between 37 percent to 48 percent lower than for OCP3. None of the scenarios in OCP4 or OCP5 leads to a release from the fuel.[26] Figure 52 through Figure 55 illustrate the thermal-hydraulic response of the high-density pool to a small leak and a moderate leak. It takes less than 6 hours to clear the rack baseplate and initiate airflow for the moderate leak, while for the small leak case, the rack baseplate does not clear until about 39 hours. In both cases, there is a heatup 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 heatup is slower since 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 56 or Figure 57). 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 the total decay heat in the pool for low-density case is only 77 percent of the high-density case, the decay heat in Ring 1 is identical in both cases.

---

26 The start of the release of radionuclides from the fuel is modeled based on a temperature of 900 degrees C (1,173 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.

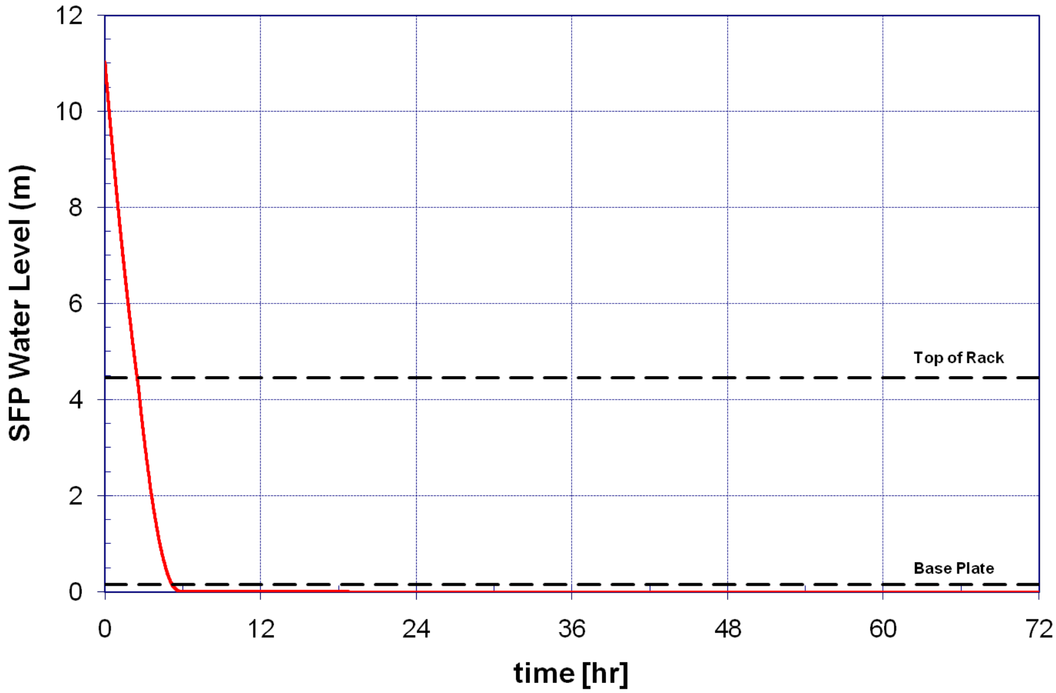

Figure 52 Water level for unmitigated high-density moderate leak (OCP4)

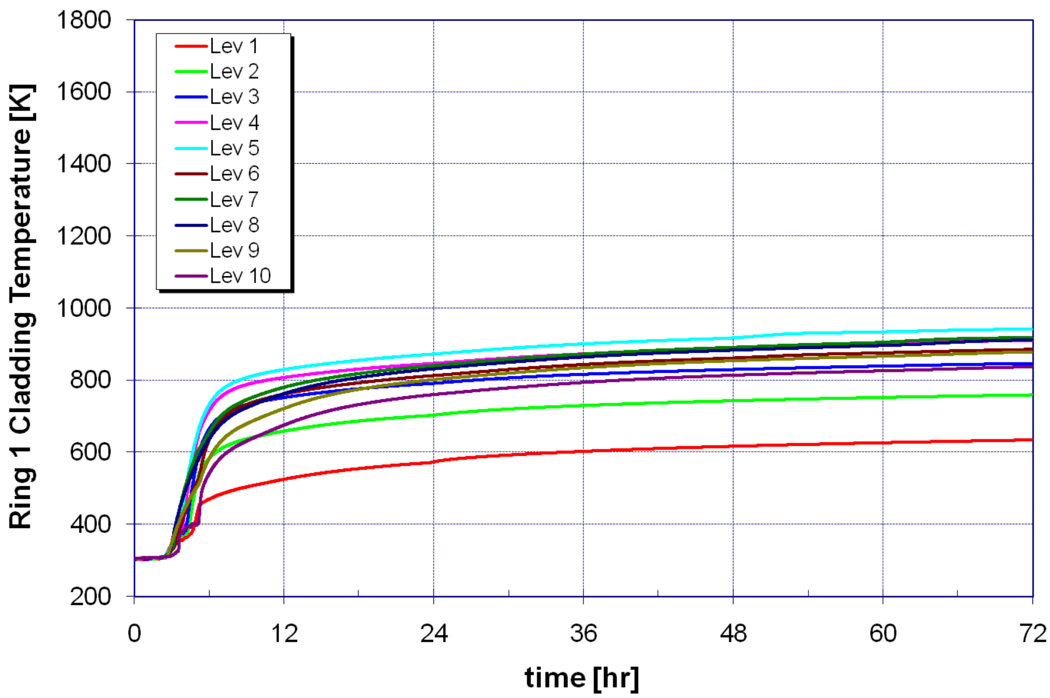

Figure 53 Ring 1 temperature for unmitigated high-density moderate leak (OCP4)

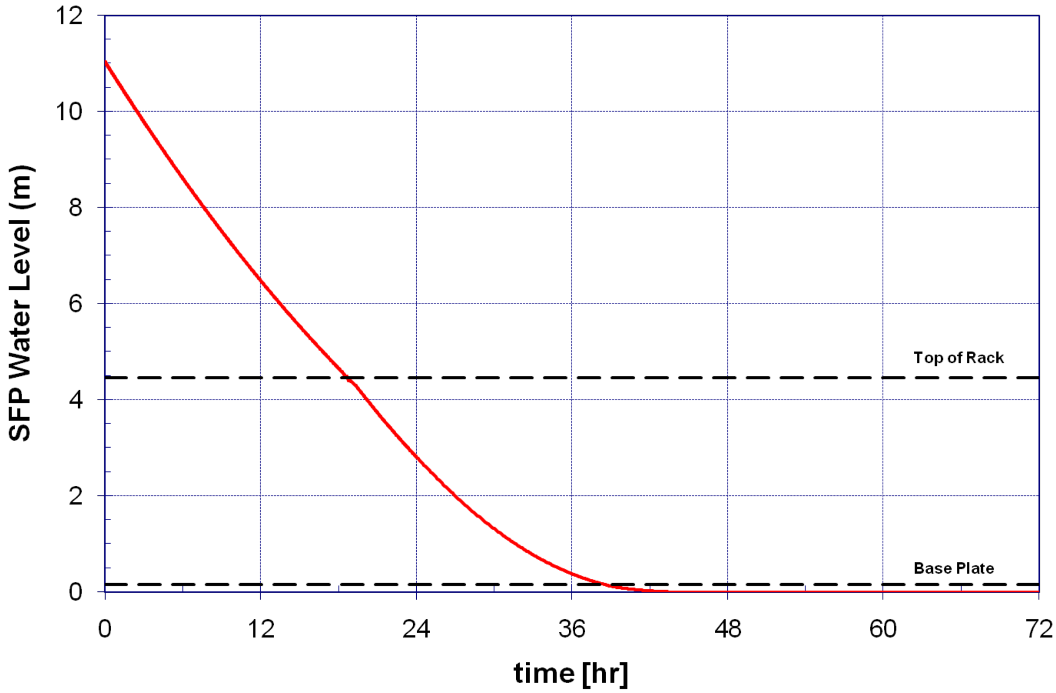

Figure 54 Water level for unmitigated high-density small leak (OCP4)

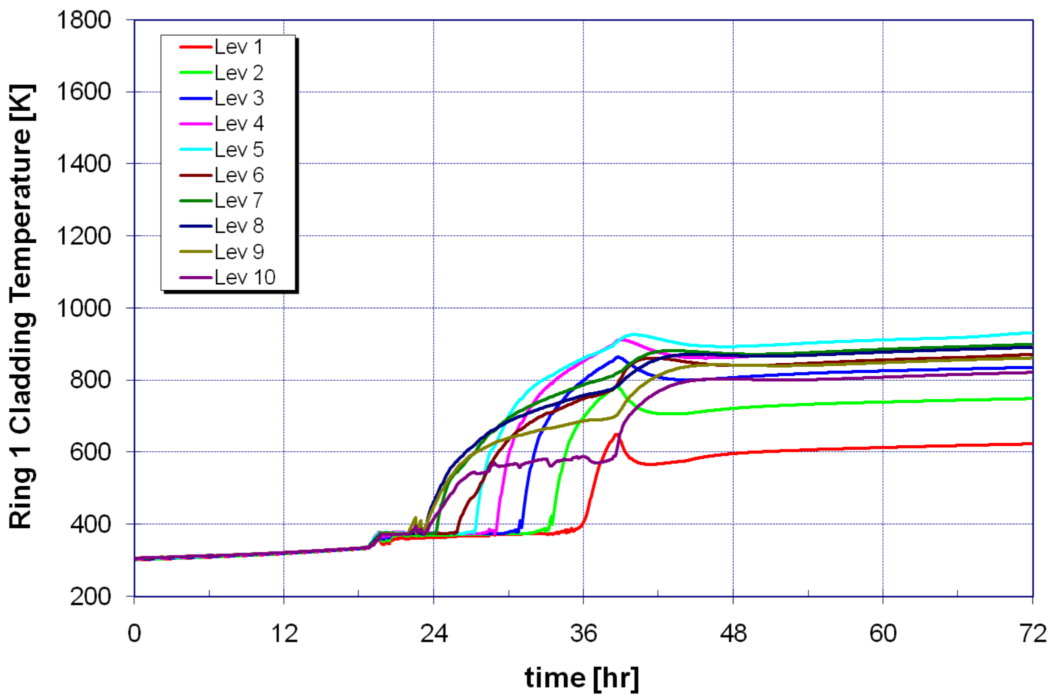

Figure 55 Ring 1 temperature for unmitigated high-density small leak (OCP4)

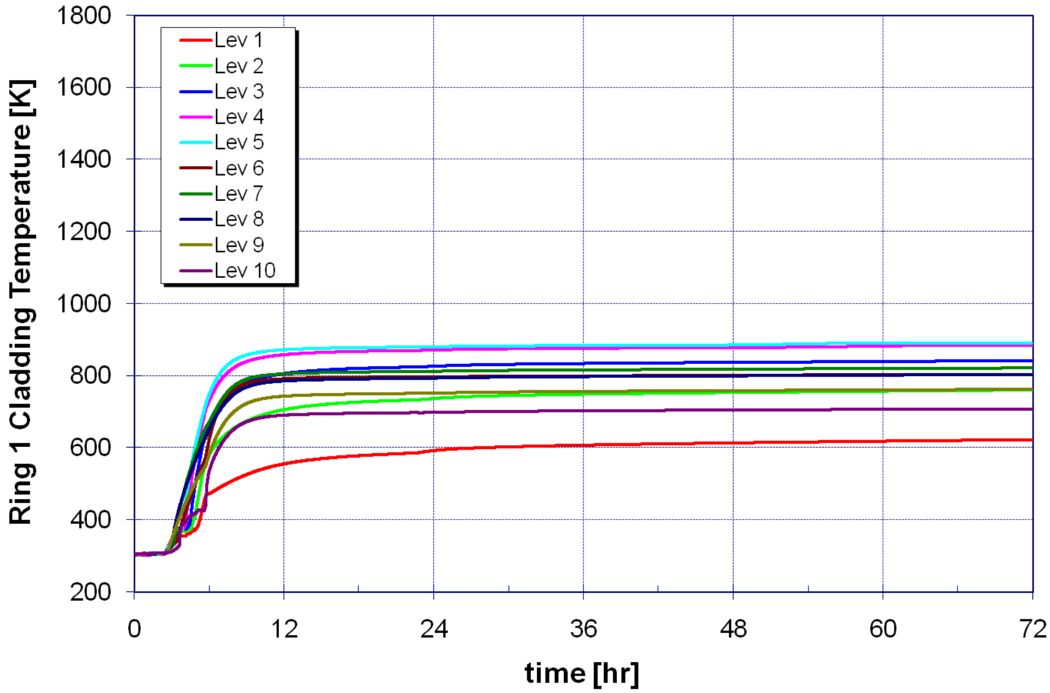

Figure 56 Ring 1 temperature for unmitigated low-density moderate leak (OCP4)

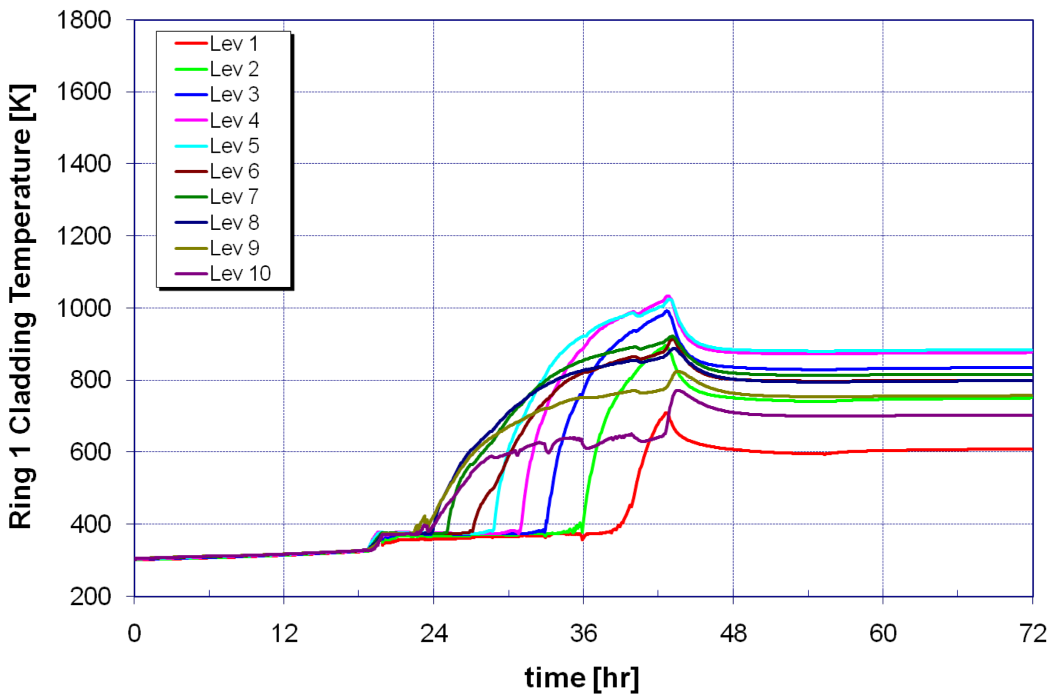

Figure 57 Ring 1 temperature for unmitigated low-density small leak (OCP4)

Mitigated Moderate Leak Scenarios in OCP2, OCP3, OCP4, and OCP5

Mitigation for moderate leak cases involves actuation of the sprays for the postoutage scenarios (OCP3, OCP4, and OCP5) and direct injection in OCP1 and OCP2. The moderate leak is modeled in MELCOR with a 11.4-cm- (4.5-in.-) 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). Section 6.1.3 of this report discussed the MELCOR modeling of the sprays and presented two modeling options (i.e., simple flow regime model on or off). Only high-density OCP3 results[28] are presented since the unmitigated scenarios in later phases do not lead to release, and the moderate leak size is large enough to avoid the baseplate blockage resulting from quasi-steady water level at the bottom of the pool in response to the 200-gpm (0.013-m^3/s) spray water. The results of the OCP2 calculation showed no release from the fuel resulting from various heat transfer mechanisms (see also discussion for OCP1 in Section 6.3.2 of this report).

Figure 58 shows the water level for the moderate leak, high-density OCP3 scenario. Because of the spray activation at 3 hours (see Figure 59), the bottom of the racks clears for natural circulation airflow more than 1 hour later compared to an unmitigated case (see Figure 52). 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 less than 200 gpm (0.013 m^3/s) in Figure 59 because of heat transfer from spray droplets to the atmosphere and fuel rods.[29] Figure 60 shows the response of the clad in Ring 1 for the case in which the simple flow regime model is active. As expected, the top cells experience more cooling as there is more water coverage. The temperatures reach a quasi-steady state by about 10 hours[30] and the maximum clad temperature is about 850 K. Figure 61 shows the clad temperatures for the case in which 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 the 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 differ, the maximum clad temperatures are almost the same and well below the gap release criterion. This is partially because of the importance of the heat removal by natural circulation of air through the racks. If there was no natural circulation of air through the racks, the cooling of the fuel by the spray flow (i.e., modeled with the simple flow regime map) would be the only effective cooling mechanism, and therefore 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 an additional 3-hour delay in the actuation of the spray as shown in Figure 62.[31] Both Figure 63 and Figure 64 show that (for OCP3), following the initial heatup 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.

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28 The low-density case is similar to the high-density case, 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.

30 The calculation fails shortly after 10 hours from numerical problems.

31 These cases were actually run based on an earlier logic for spray actuation that assumed a 3-hour additional delay at the end of deployment.

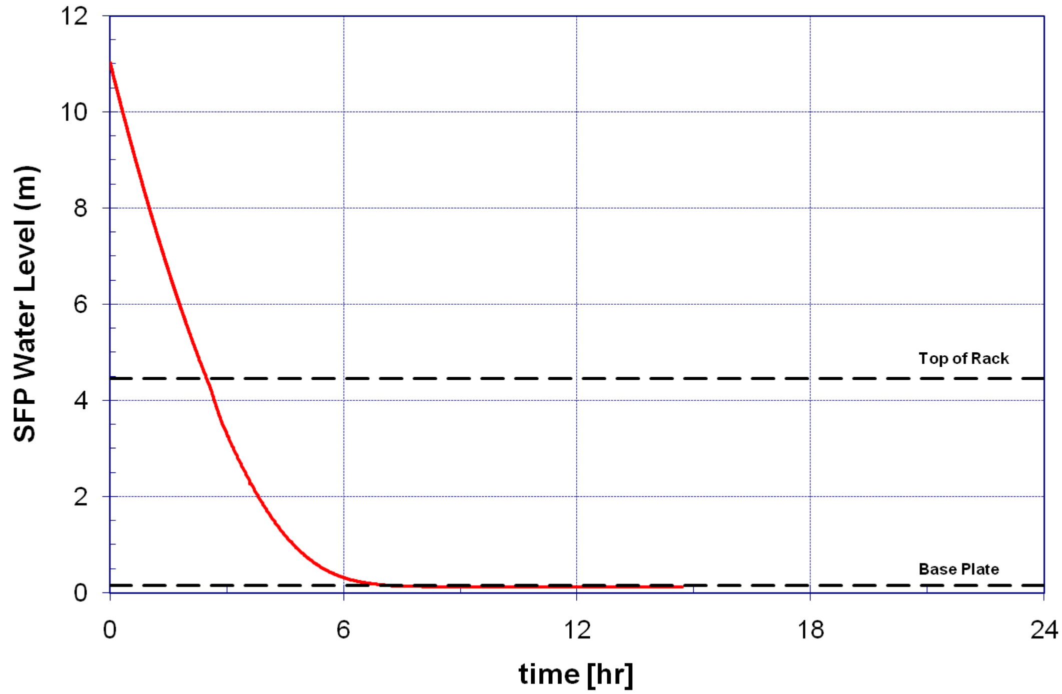

Figure 58 Water level for mitigated high-density moderate leak (OCP3)

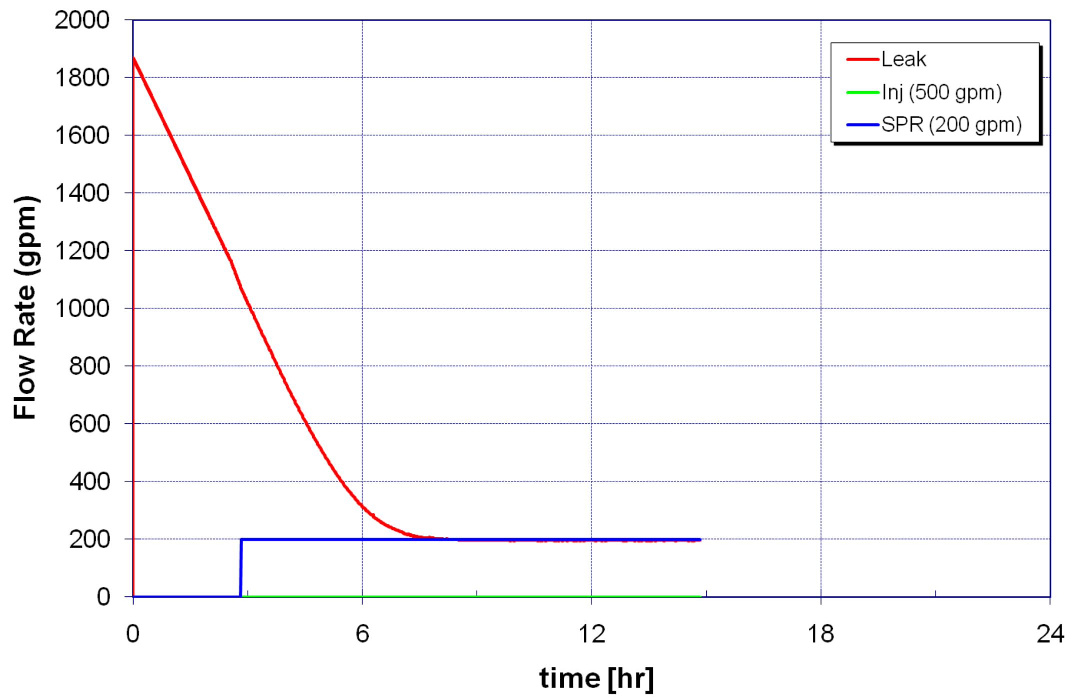

Figure 59 Water flow rates for mitigated high-density moderate leak (OCP3)

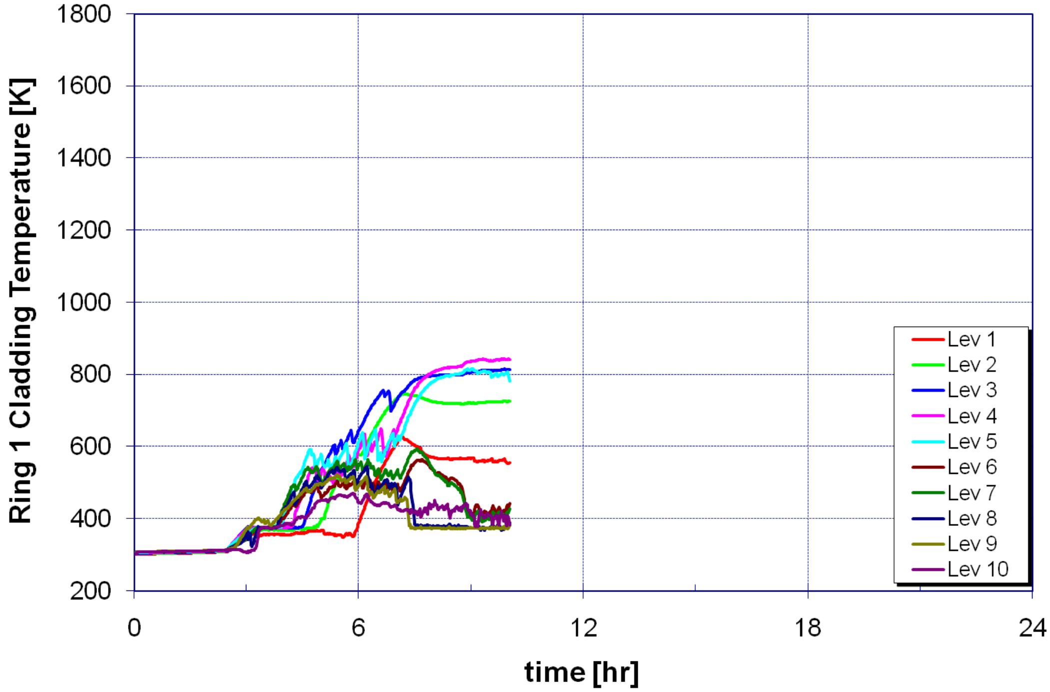

Figure 60 Ring 1 clad temperatures for mitigated (simple flow regime active) high-density moderate leak (OCP3)

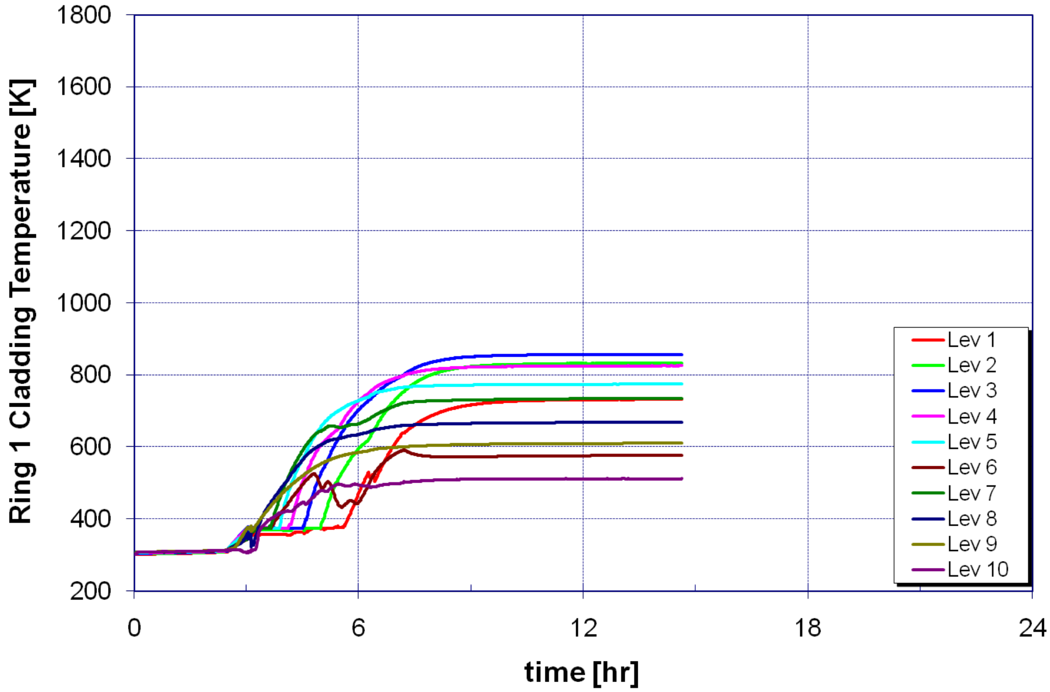

Figure 61 Ring 1 clad temperatures for mitigated (simple flow regime inactive) high-density moderate leak (OCP3)

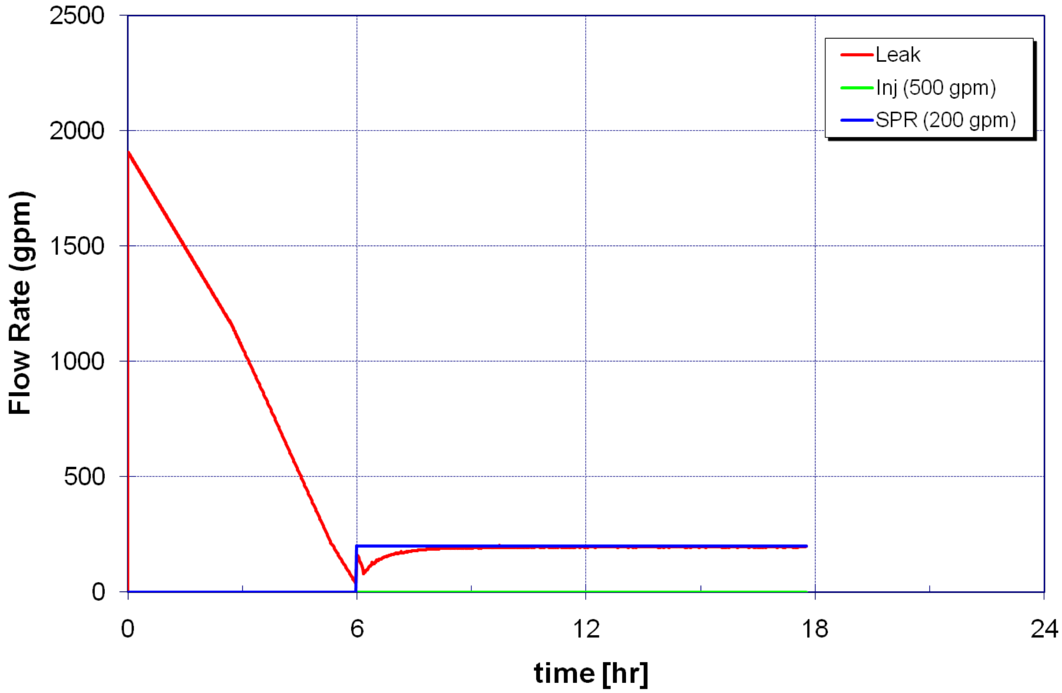

Figure 62 Flow rates for mitigated high-density moderate leak (OCP3) with late actuation of sprays

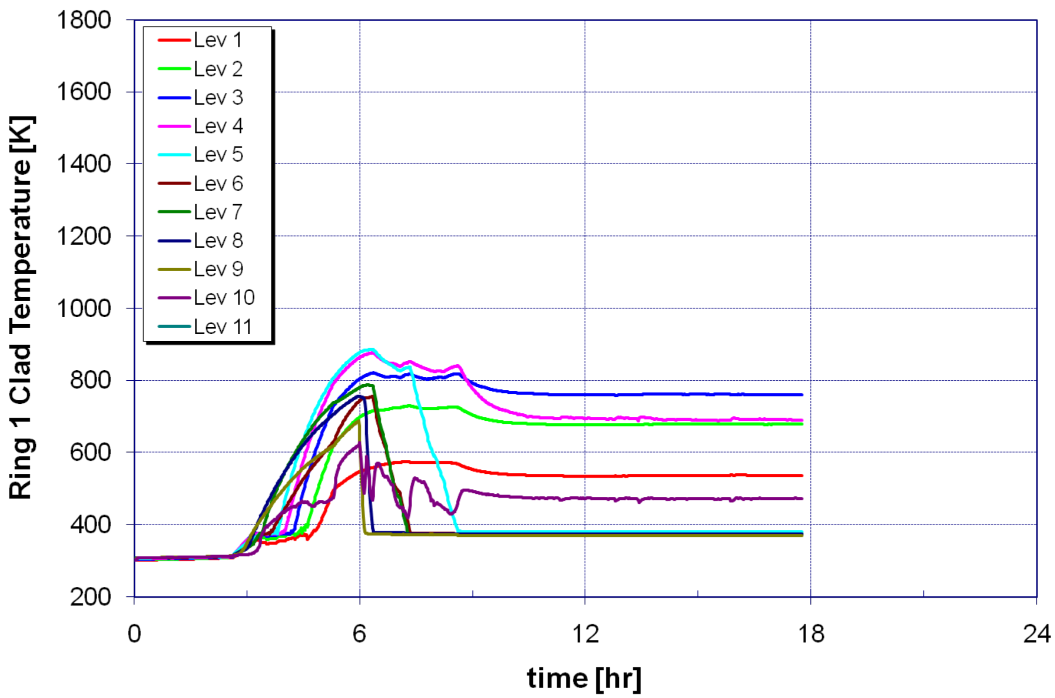

Figure 63 Ring 1 clad temperatures for mitigated (simple flow regime active) high-density moderate leak (OCP3) with late actuation of sprays

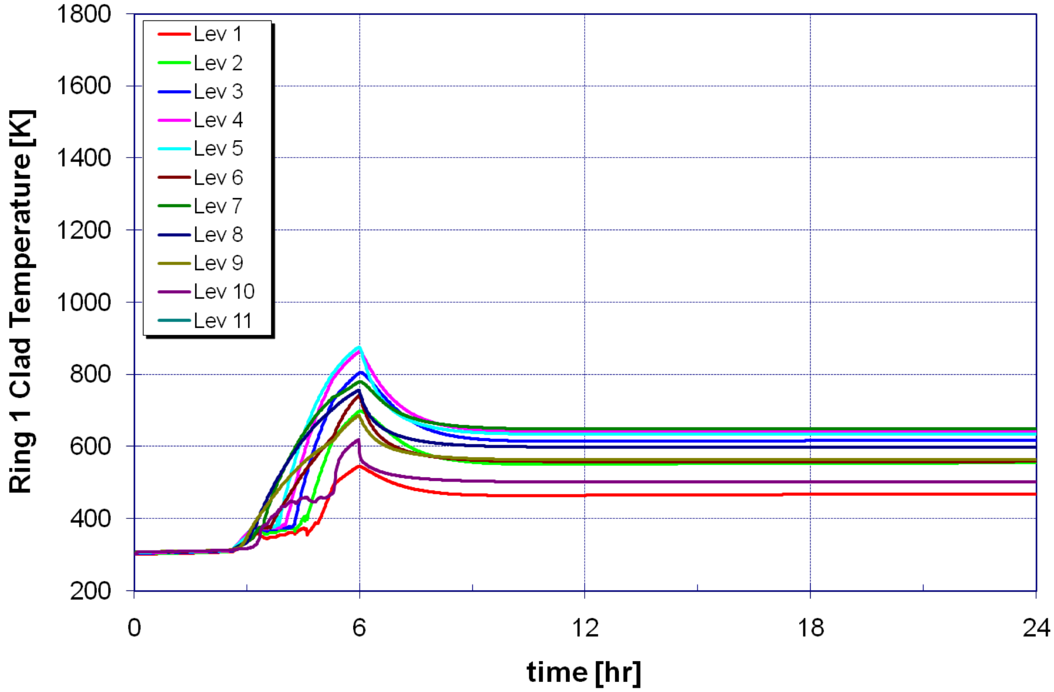

Figure 64 Ring 1 clad temperatures for mitigated (simple flow regime inactive) high-density moderate leak (OCP3) with late actuation of sprays

6.3.2 Sequences That Do Lead to a Release

All the sequences in OCP1, OCP2, and OCP3 lead to release without successful deployment of mitigation. This section will only discuss representative scenarios 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 42), 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-percent concentration if there is adequate oxygen (oxygen concentration is greater than or equal to 5 percent) and no steam inerting (steam concentration is less than or equal to 55 percent). The analysis considered the sensitivity of the ignition assumptions and potential for reactor building refueling bay failure on a case-by-case basis (see Section 9.1 of this report).

<u>Unsuccessful Deployment of Mitigation for Moderate Leak (OCP1) Scenario</u>

The water level for the high-density scenario (Figure 65) 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 postoutage scenarios (see Figure 52) because of the additional water in the reactor well connected to the SFP. The reactor power (Figure 66) 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 67) 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 zirconium fire that starts near the top of the full rod region (see Figure 41) and propagates downward. The heatup in Ring 1 (fuel, cladding, canister, and racks) propagates to Ring 2 assemblies (Figure 68) 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 and 14.2 hours, the channel boxes in Rings 1 and 2 fail, which 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 heat up 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 zirconium fire initiates at the top of the fuel in Ring 4 at about 42 hours in the upper levels which then propagates downward. The second heatup is more intense and involves the other rings as indicated by both the oxidation power (Figure 66) and the clad temperatures in the outer rings (Figure 68).

OCP1 had a relatively rapid draindown in which an air natural circulation flow developed through the racks before 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 concentration in the refueling floor was only 5 percent, 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 refueling bay, which remained 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 42). The reactor building DF is shown in Figure 71.[34] Aerosols also begin to deposit inside the building and the DF for cesium and iodine aerosols remains between 3 to 4 for much of the accident. The DF 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 DF to arrive at the environmental release for each ring. MELCOR mechanistically models all deposition mechanisms; however, because of the mixing within the reactor building, only an overall DF can be defined for all rings.

Figure 72 depicts the cesium environmental release fraction for individual rings. 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 68. The later releases result from the second heatup and involve all of the outer rings (Ring 3 is empty for OCP1). The total release fraction is the input to the MACCS2 code for consequence analysis and is defined by Equation

32 The initial heatup of the liner is caused by 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 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 preclude steam oxidation.

34 The integral DF 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.

11.[35] The DF is a dynamic quantity as the outer rings start to release (see the fluctuations in Figure 71); therefore, care is taken to allow the earlier releases from inner rings preserve their release history so that the total release fraction does not decrease at any time as the release progresses.

Figure 73 illustrates the results of the low-density case. Comparing the heatup with the high-density case (see Figure 67), 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, which results in slower heatup of Ring 5, as shown in Figure 74. Even though the racks fail in this low-density case, the canisters remain intact and the zirconium fire moves down initially and then upwards, as shown in Figure 73. The cesium environmental release fraction for Ring 1 shown in Figure 75 is comparable to the high-density case (Figure 72), but since no release occurs from the older assemblies, the total release fraction for the low-density case is lower.

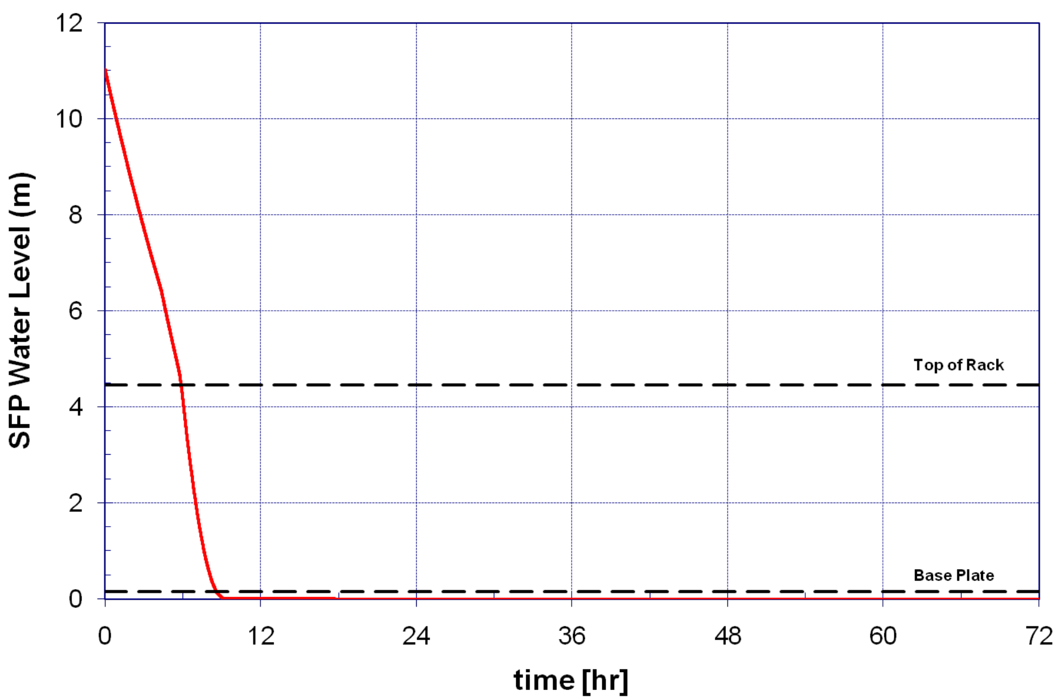

Figure 65 Water level for unmitigated high-density moderate leak (OCP1)

[35] This activity-weighted release 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 is smaller compared to Ring 1.

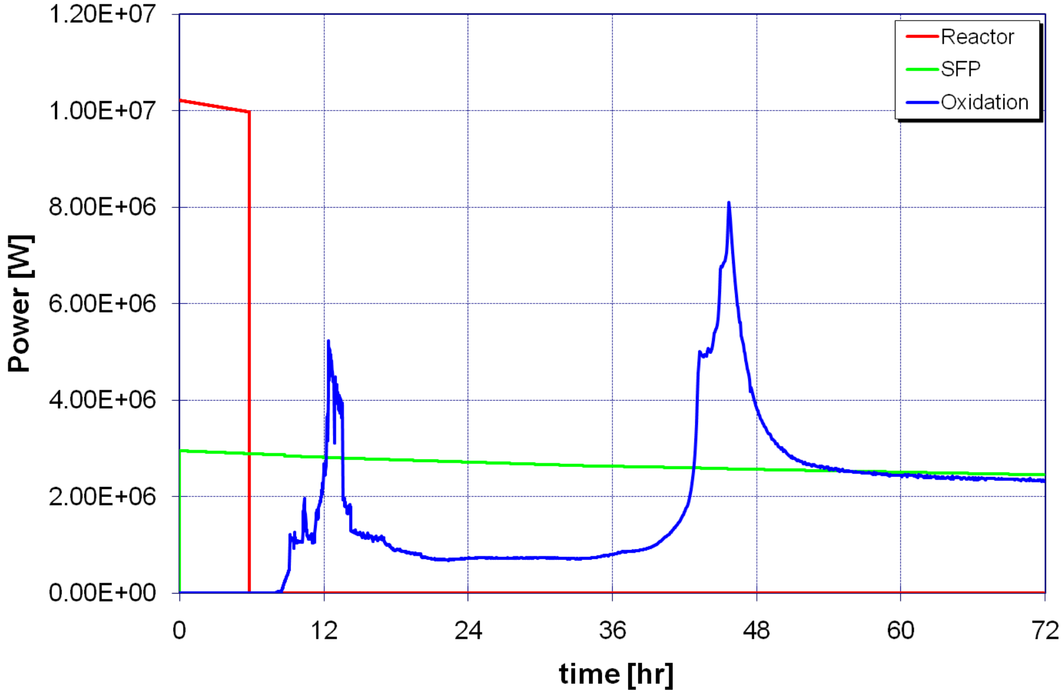

Figure 66 SFP power for unmitigated high-density moderate leak (OCP1)

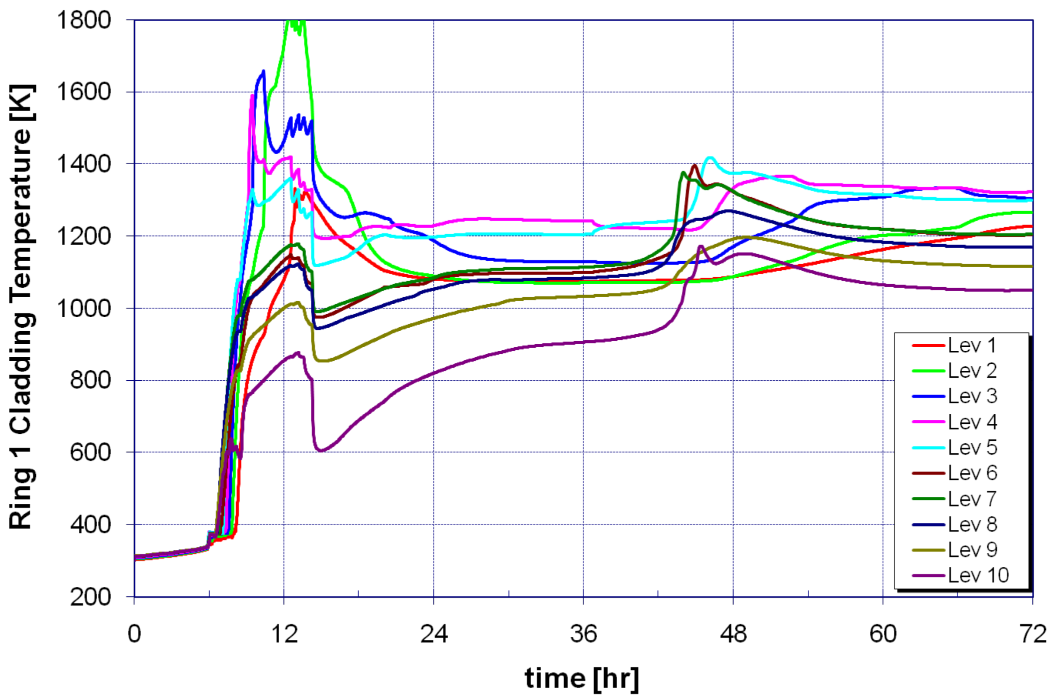

Figure 67 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP1)

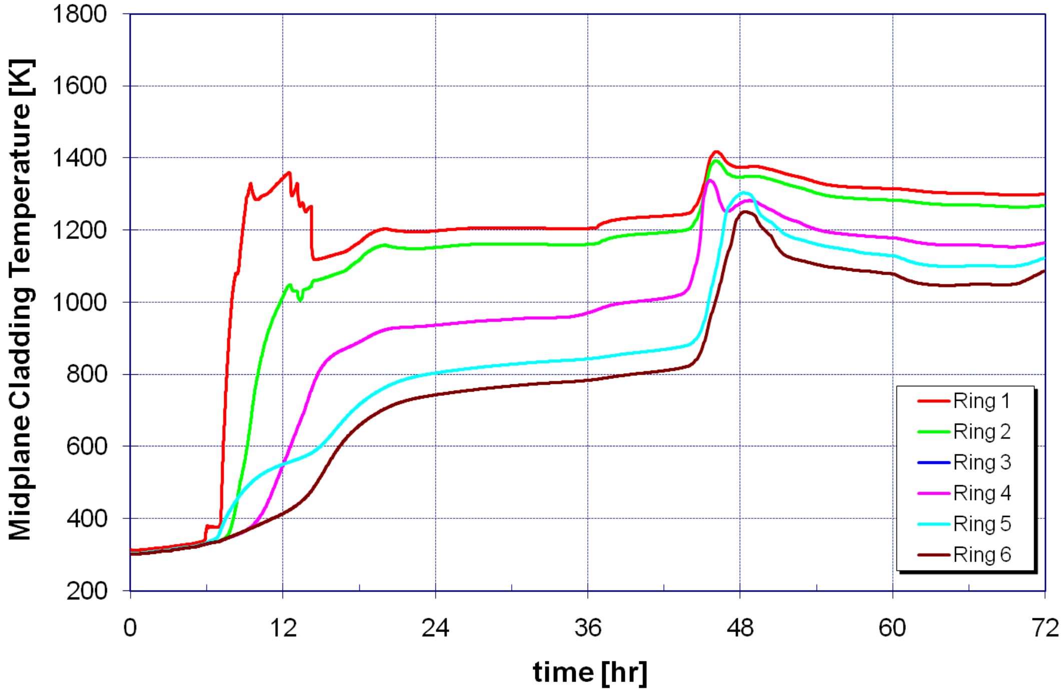

Figure 68 Midplane clad temperature for unmitigated high-density moderate leak (OCP1)

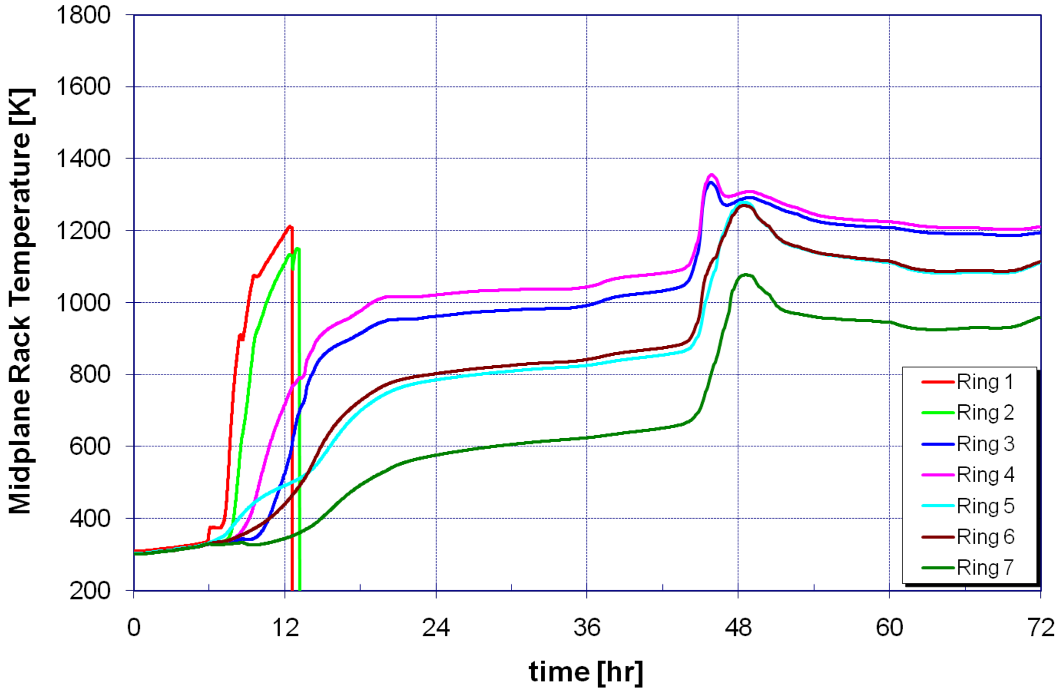

Figure 69 Midplane rack temperature for unmitigated high-density moderate leak (OCP1)

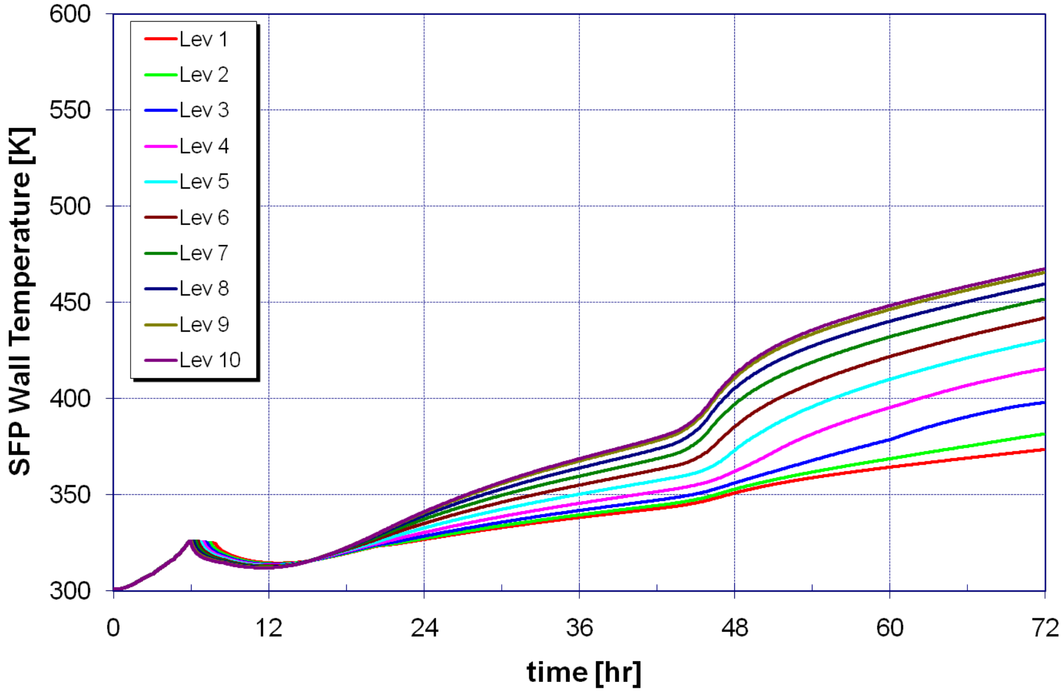

Figure 70 SFP wall liner temperature for unmitigated high-density moderate leak (OCP1)

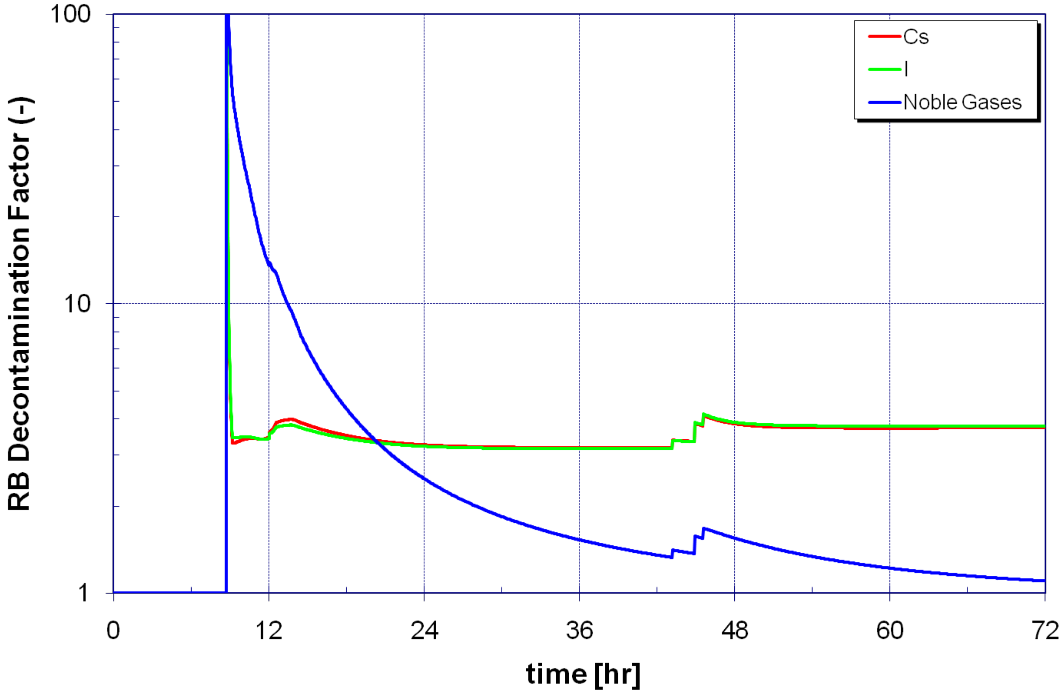

Figure 71 Reactor building DF for unmitigated high-density moderate leak (OCP1)

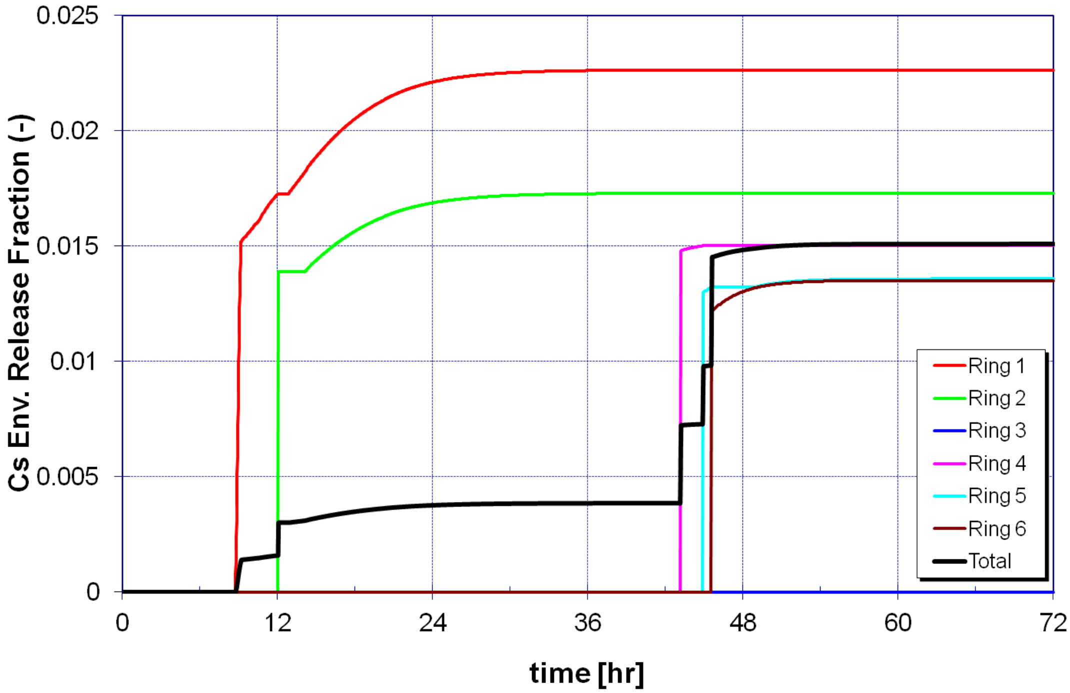

Figure 72 Cesium environmental release fraction for unmitigated high-density moderate leak (OCP1)

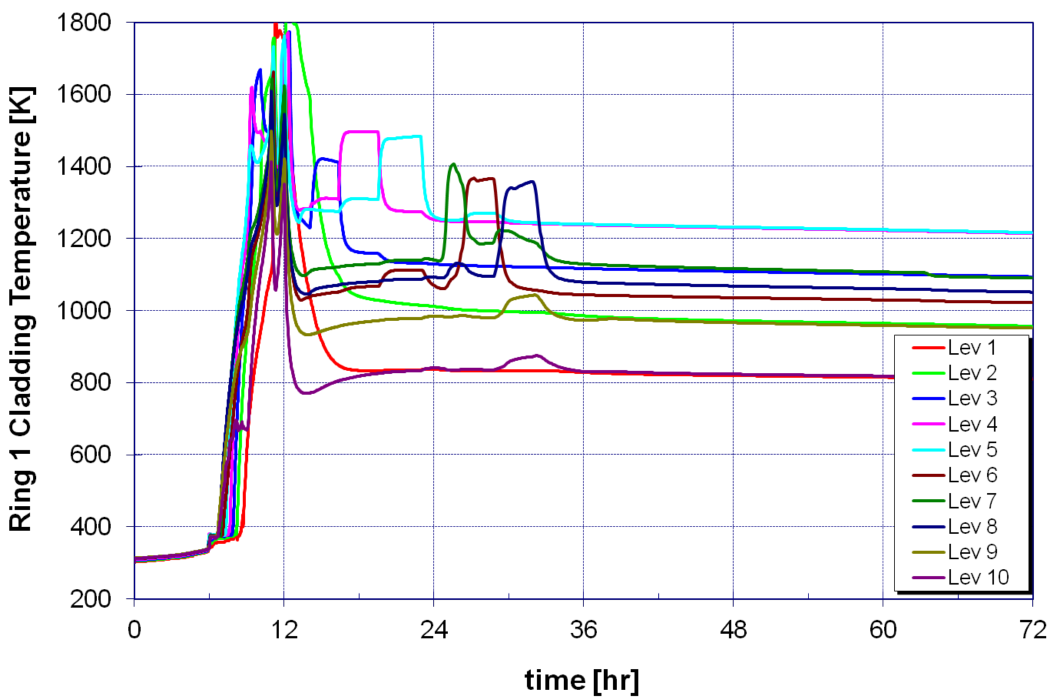

Figure 73 Ring 1 clad temperature for unmitigated low-density moderate leak (OCP1)

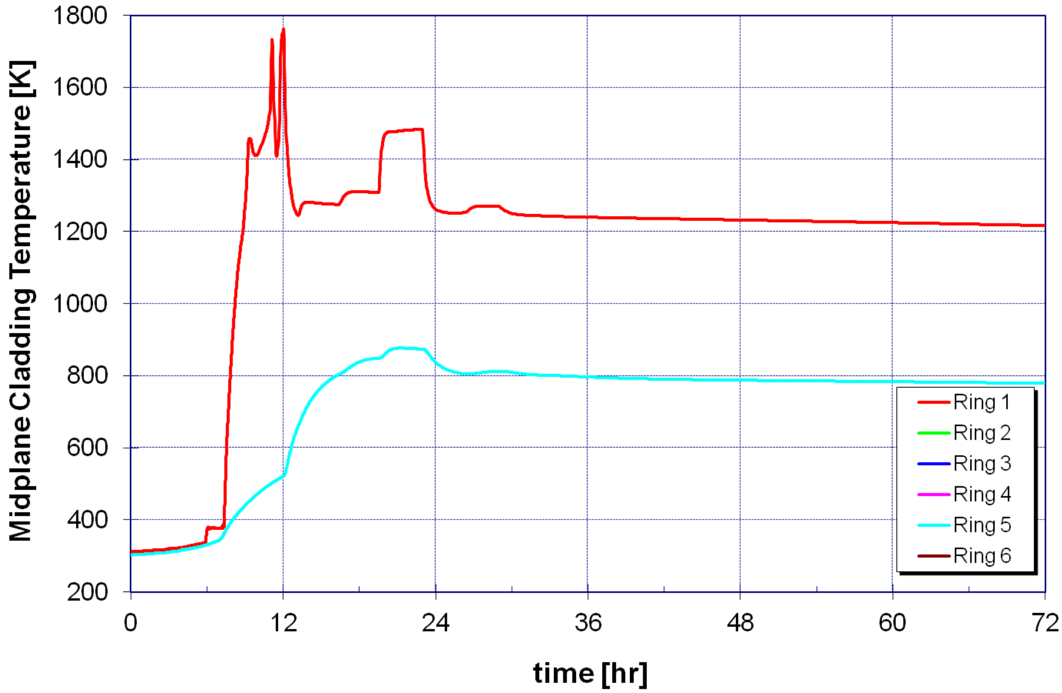

Figure 74 Midplane clad temperature for unmitigated low-density moderate leak (OCP1)

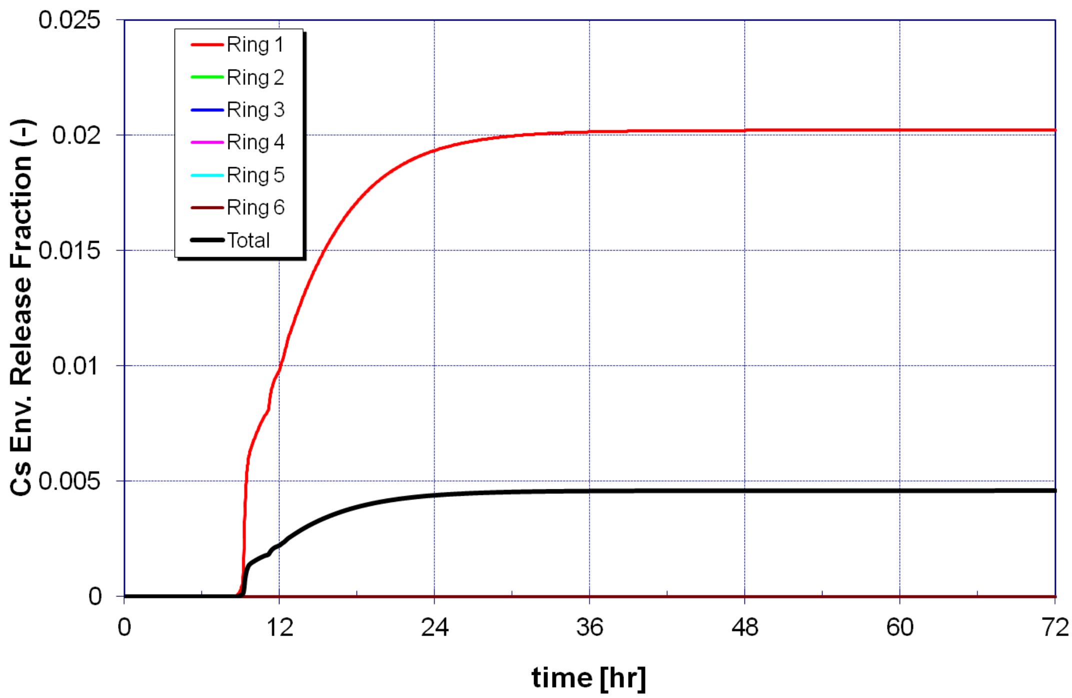

Figure 75 Cesium environmental release fraction for unmitigated low-density moderate leak (OCP1)

Mitigated Moderate Leak (OCP1) Scenario

Figure 76 illustrates the response of the pool to the mitigated scenario. The connectivity between the reactor and the SFP, as well as the additional volume of water, results in a relatively slow draindown. Thus, 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 heatup of the fuel occurs (see Figure 77), there is no indication of a zirconium 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 discharged, high-temperature fuel to adjacent fuel assemblies; and steam cooling from boiling in the bottom of the assemblies between cells keep the fuel temperature near 1200 K. Only Ring 1 had cladding failure and subsequent releases of the gap inventory, as shown in Figure 78. All other fuel was below the threshold for cladding failure and fission product releases.

Figure 79 shows the clad temperature in Ring 1 for the low-density case. The heatup rate for the low-density case is more extreme than the high-density case, as was observed for the 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 heatup. 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. Higher fuel temperatures during the initial oxidation transient led to slightly more release in the low-density case.

[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 the 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.

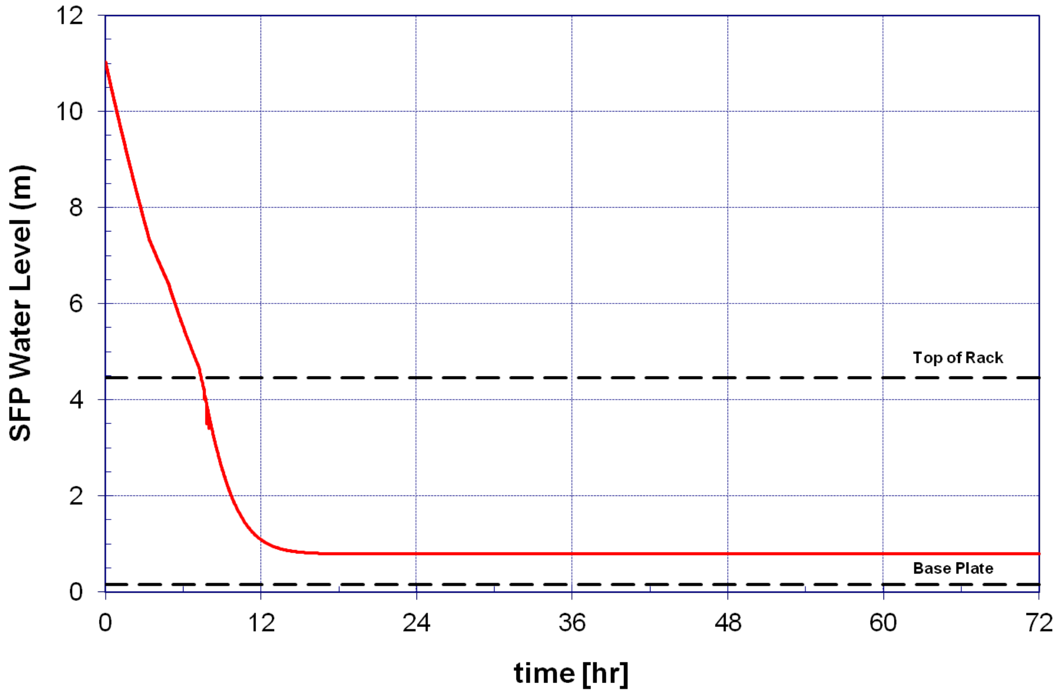

Figure 76 Water level for mitigated high-density moderate leak (OCP1)

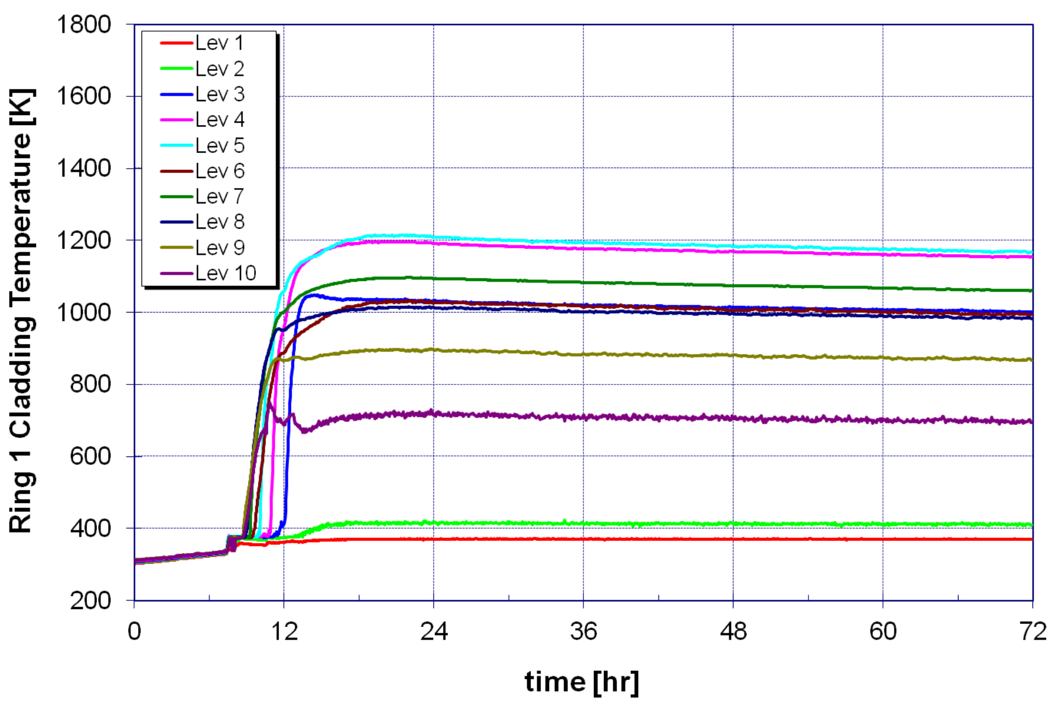

Figure 77 Ring 1 clad temperature for mitigated high density moderate leak (OCP1)

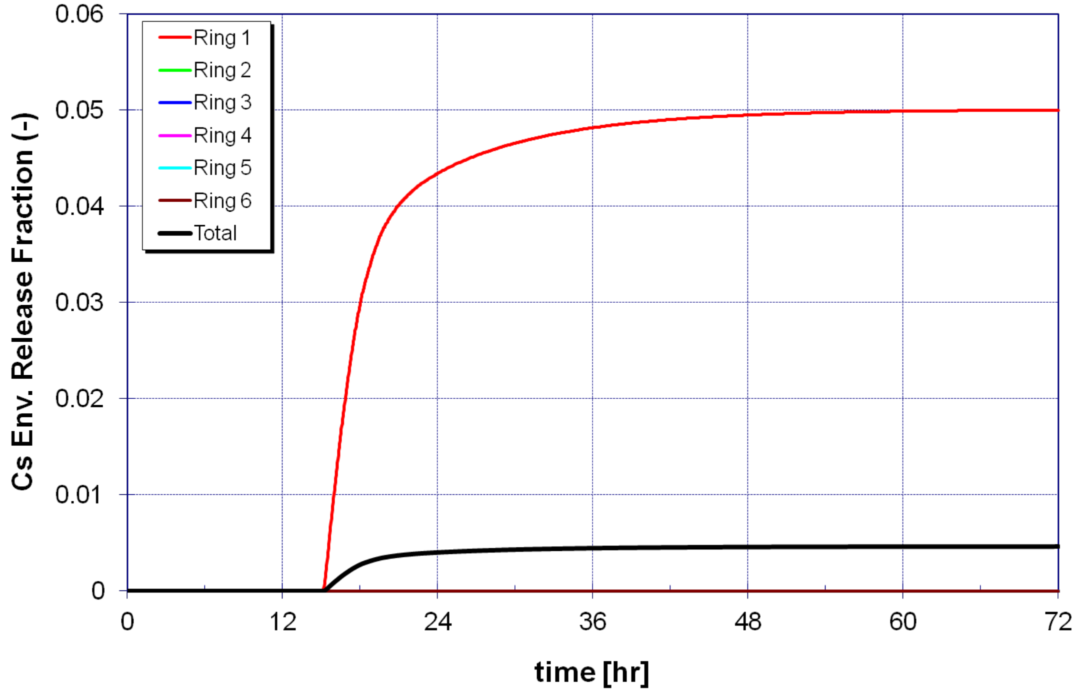

Figure 78 Cesium environmental release fraction for mitigated high density moderate leak (OCP1)

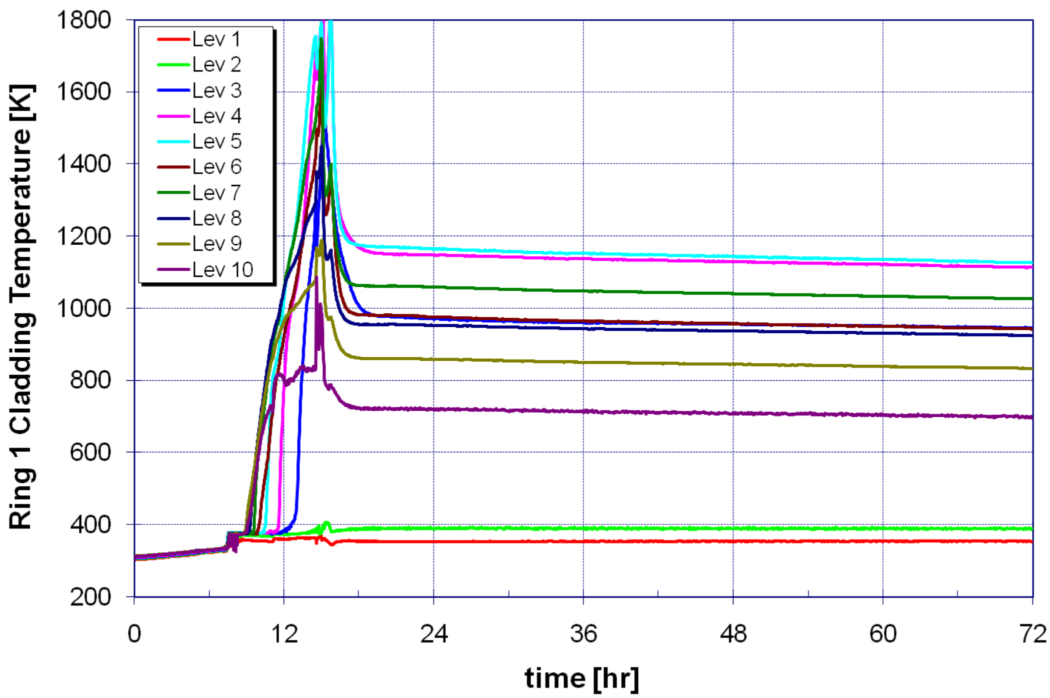

Figure 79 Ring 1 clad temperature for mitigated low density moderate leak (OCP1)

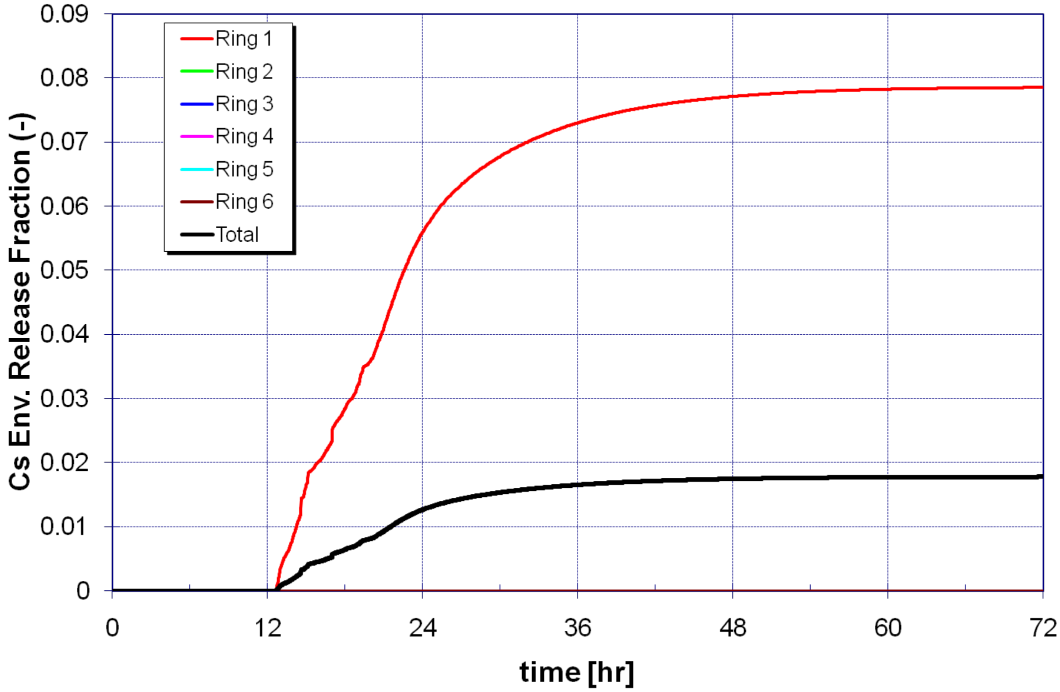

Figure 80 Cesium environmental release fraction for mitigated low density moderate leak (OCP1)

Unsuccessful Deployment of Mitigation for Small Leak (OCP2) Scenario

This scenario involves a hydrogen combustion that caused a late rapid air oxidation. Figure 81 shows the slow draindown of the pool exposing the top of the racks at 42.6 hours compared to 18.7 hours for postoutage scenarios (see Figure 54). Figure 82 illustrates the decay power and the oxidation power. The air oxidation power reaches an order of magnitude higher than the decay heat during the oxidation transient after 60 hours. The fuel heatup begins after the water level reaches about the fuel midplane (see Ring 1 response in Figure 83). The high-temperature fuel in Ring 1 heats the surrounding low decay heat fuel in Ring 2, as shown in Figure 84.

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 structures. The hydrogen concentration builds up until it reaches 10 percent at 65 hours and combusts. At the time of combustion, all the necessary conditions are satisfied; the hydrogen concentration is 10 percent, the oxygen is 10 percent, and the steam is less than the 55-percent 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 Ring 1 and then in Ring 2. The reactor building decontamination factor approaches unity (Figure 86) resulting in about a 17-percent cesium release to the environment (Figure 87).

The response for the low-density case was similar to, 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. Figure 88 shows the response of the highest decay heat assemblies in Ring 1. The peak fuel temperatures were less than 1,400 K. As shown in Figure 89, the fuel in Ring 3 had a similar response, but the fuel in Ring 5 was substantially lower. Fewer fuel assemblies and lower peak temperatures resulted in 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 percent) than in the high-density case.

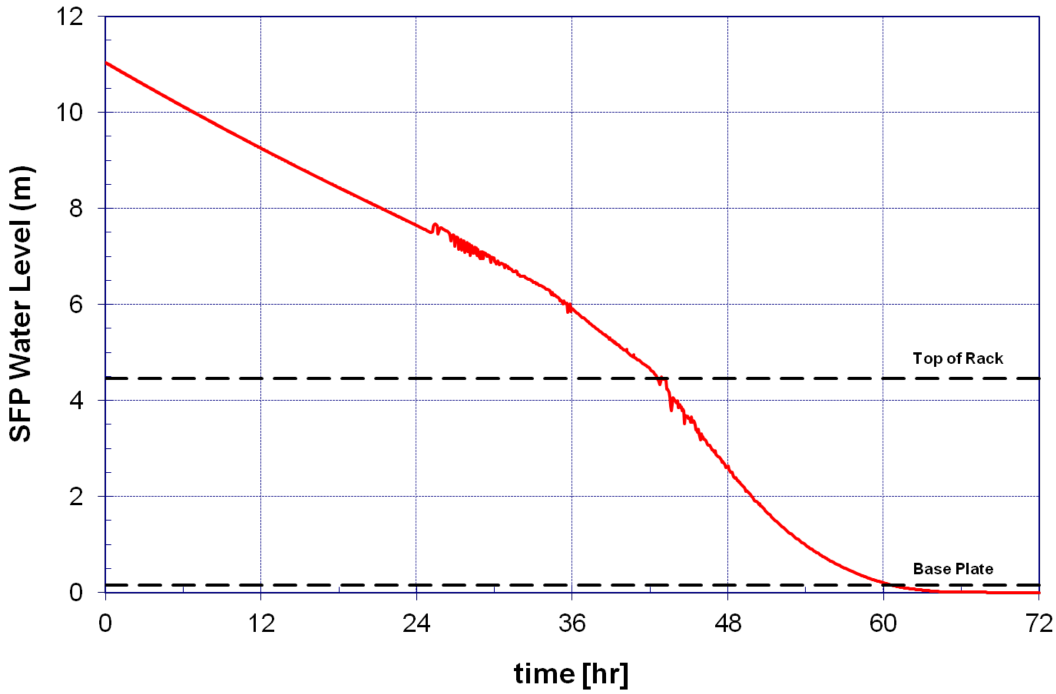

Figure 81 Water level for unmitigated high-density small leak (OCP2)

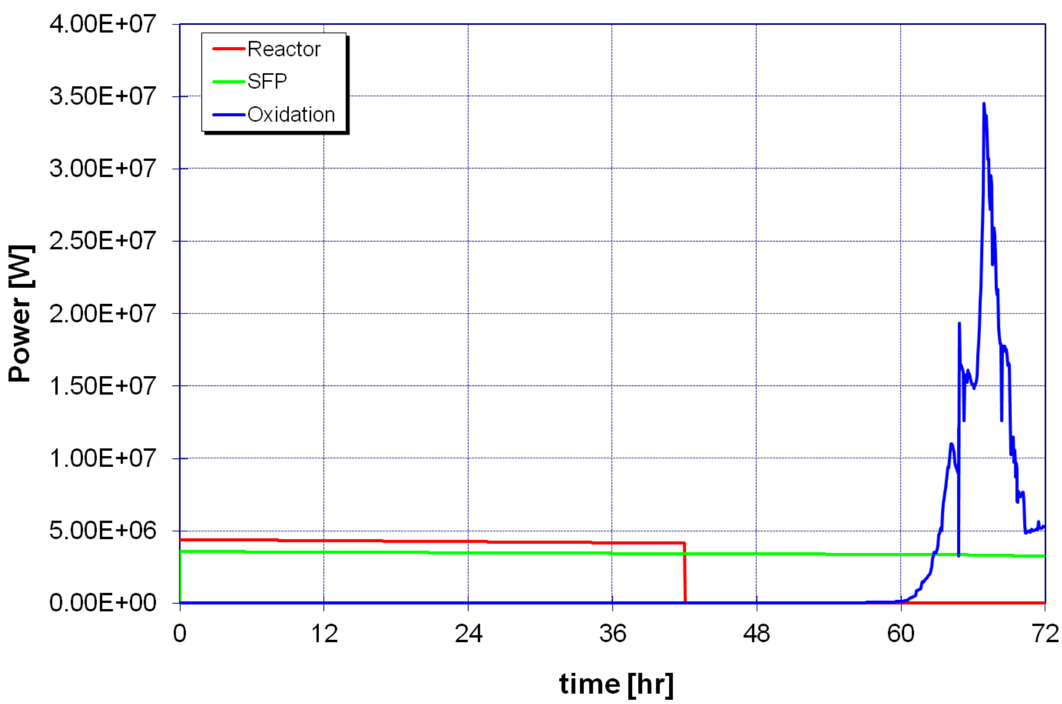

Figure 82 SFP power for unmitigated high-density small leak (OCP2)

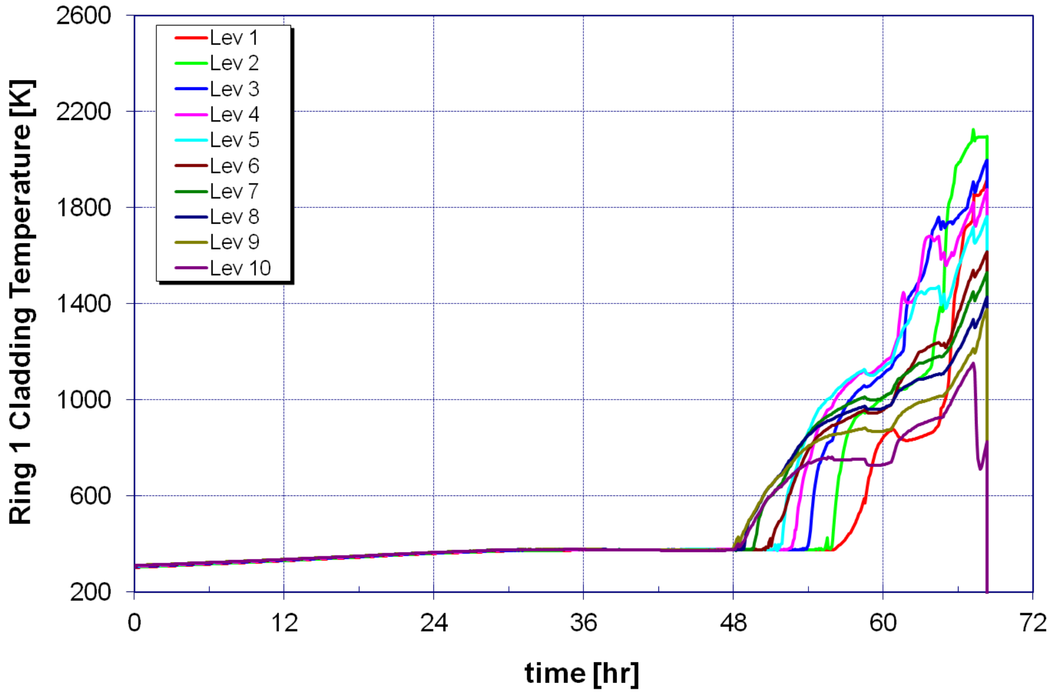

Figure 83 Ring 1 clad temperature for unmitigated high-density small leak (OCP2)

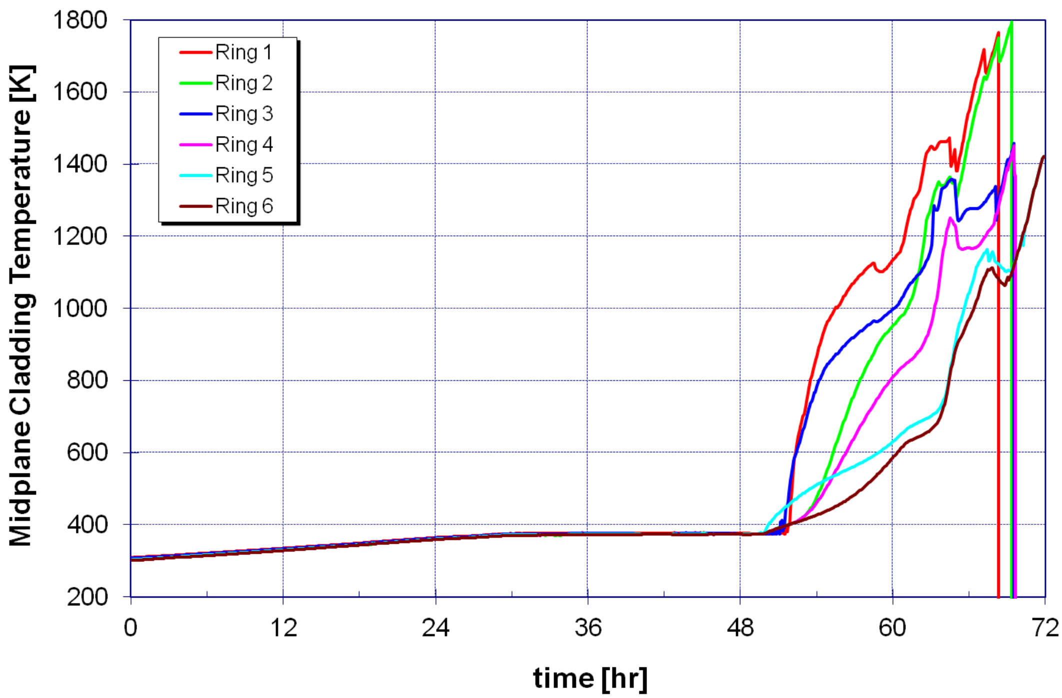

Figure 84 Midplane clad temperature for unmitigated high-density small leak (OCP2)

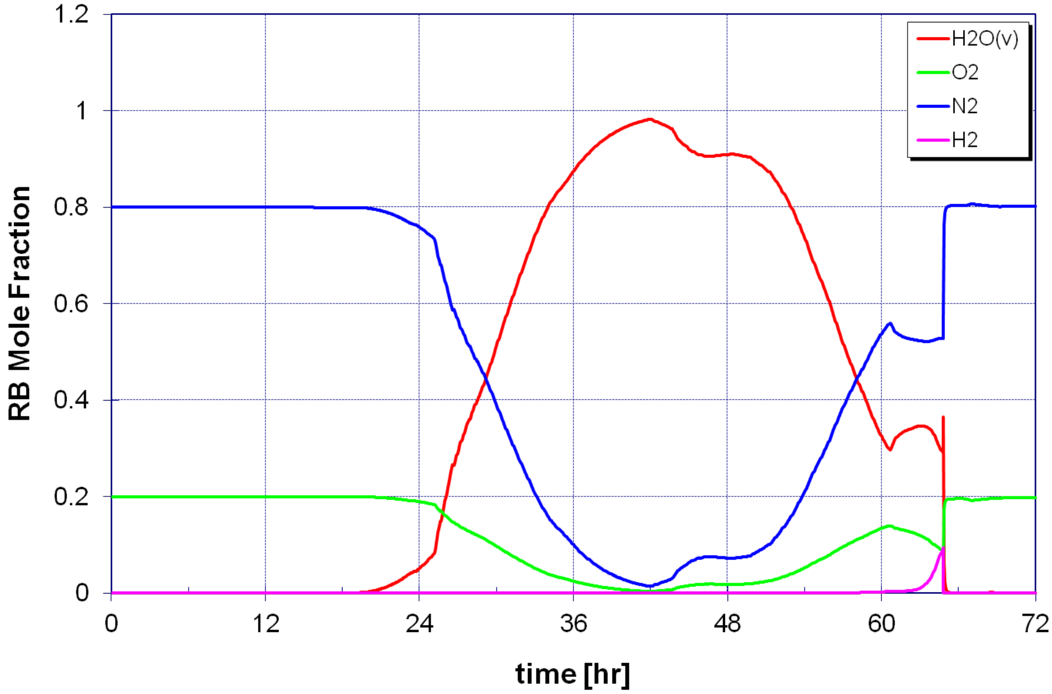

Figure 85 Reactor building mole fraction for unmitigated high-density small leak (OCP2)

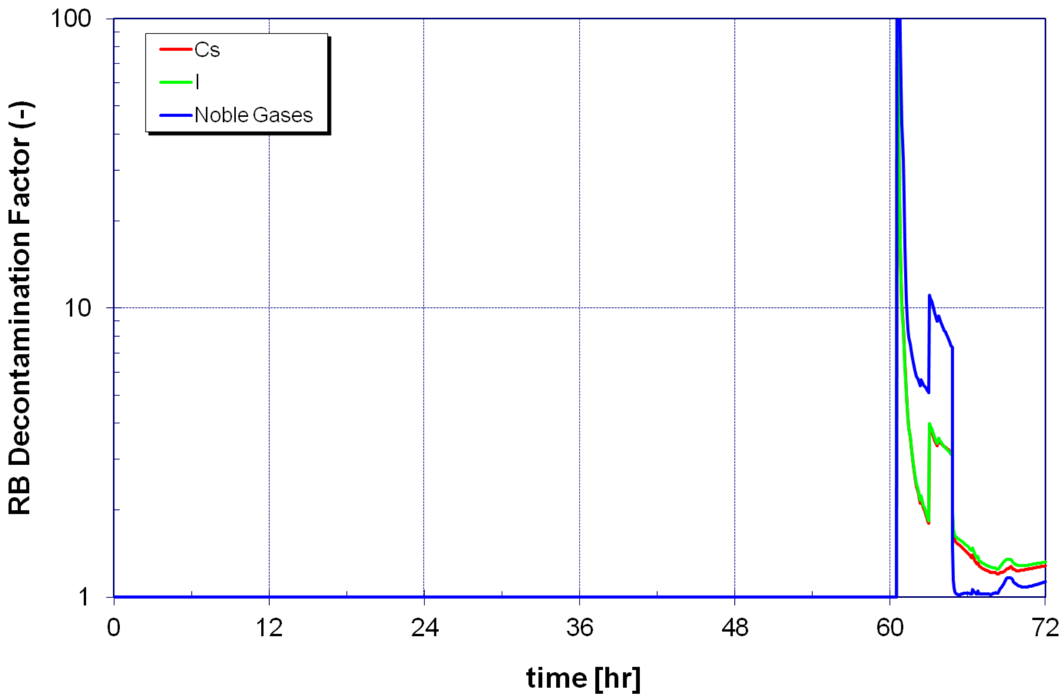

Figure 86 Reactor building DF for unmitigated high-density small leak (OCP2)

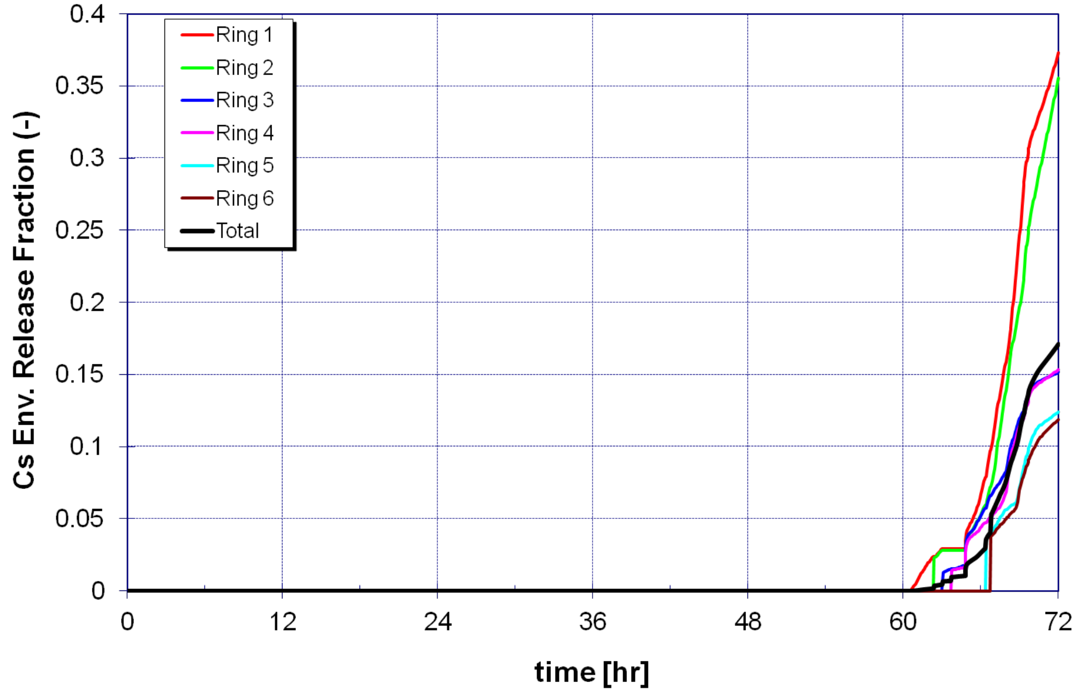

Figure 87 Cesium environmental release fraction for unmitigated high-density small leak (OCP2)

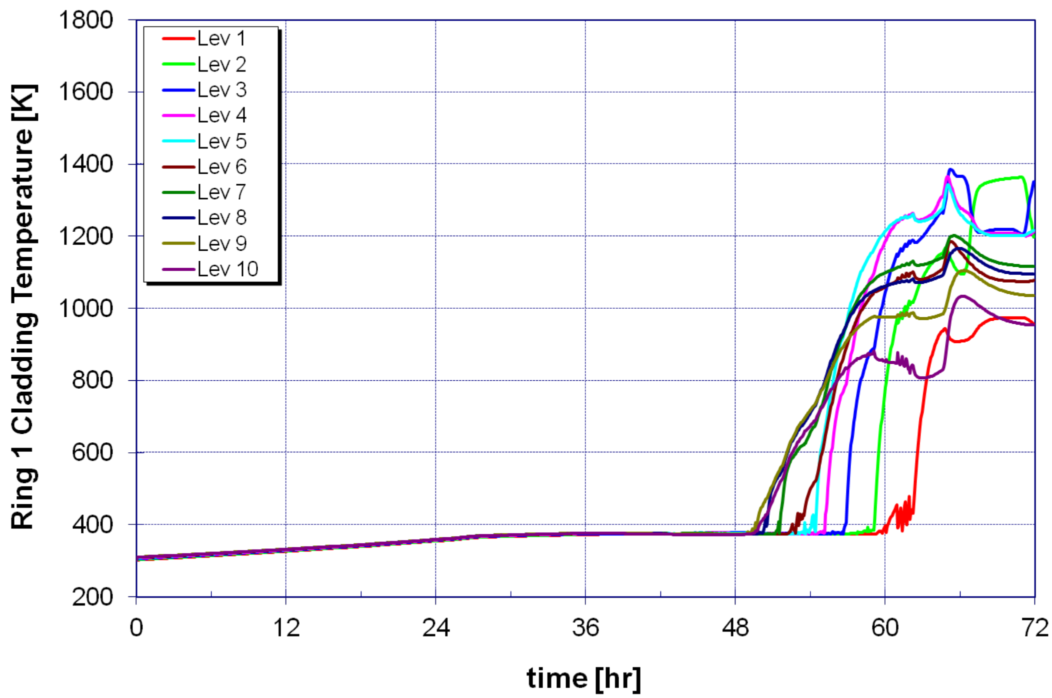

Figure 88 Ring 1 clad temperature for unmitigated low-density small leak (OCP2)

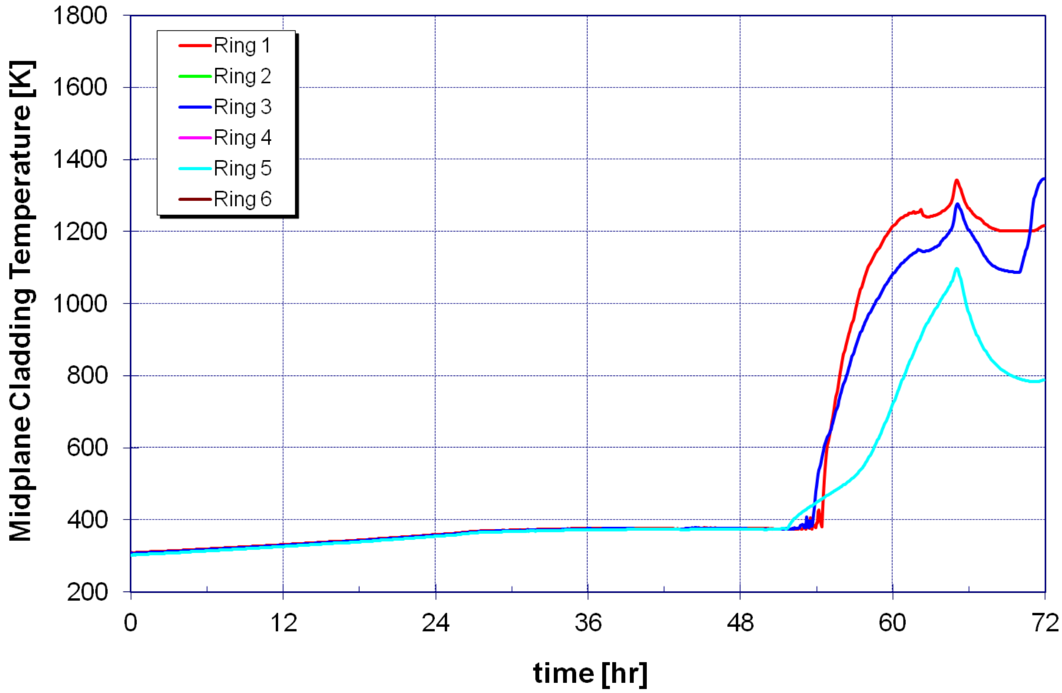

Figure 89 Midplane clad temperature for unmitigated low-density small leak (OCP2)

Unsuccessful Deployment of Mitigation for Moderate Leak (OCP3) Scenario

Figure 90 (compare to Figure 67 for OCP1) depicts the response of the fuel temperature in Ring 1. The heatup starts sooner because there is less water to drain and the approach to a zirconium fire is more gradual because of lower decay heat (i.e., by a factor of 2.5—see Table 25) and the natural circulation of air through the assemblies. However, once the zirconium fire is started, the maximum temperatures are comparable in both cases. As shown in Figure 90, the zirconium fire starts at Level 5 but then moves slowly to Level 4, Level 3, and Level 2. After the peak temperature at Level 4, the peak temperature in the zirconium fire front decreases with each successive level. Radial heat transfer from the fuel racks to the SFP wall (Figure 91), the buildup of the oxide layer on the fuel, and the depletion of the oxygen in the reactor building (Figure 92) cause the clad temperature to decrease. After 24 hours, the fuel temperatures in Ring 1 are relatively stable. There was no hydrogen combustion in this calculation. When the hydrogen concentration peaks at 8 percent, the oxygen concentration is only 3 percent (well below an amount sufficient for combustion as shown in Figure 92).

Figure 93 shows the temperature profiles for the low-density case.. The low-density temperatures are about 400 K lower than the high-density case, with the total cesium release being about 0.1 percent compared to 0.7 percent 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 analysis was performed to examine the effect of higher vapor pressure for the air-oxidizing ruthenium releases. Figure 94 (the default ruthenium release model) and Figure 95 (the enhanced ruthenium release model used in the present study) show that the ruthenium release differs by an order of magnitude.[37] All the calculations with moderate leaks were based on the enhanced ruthenium release model.

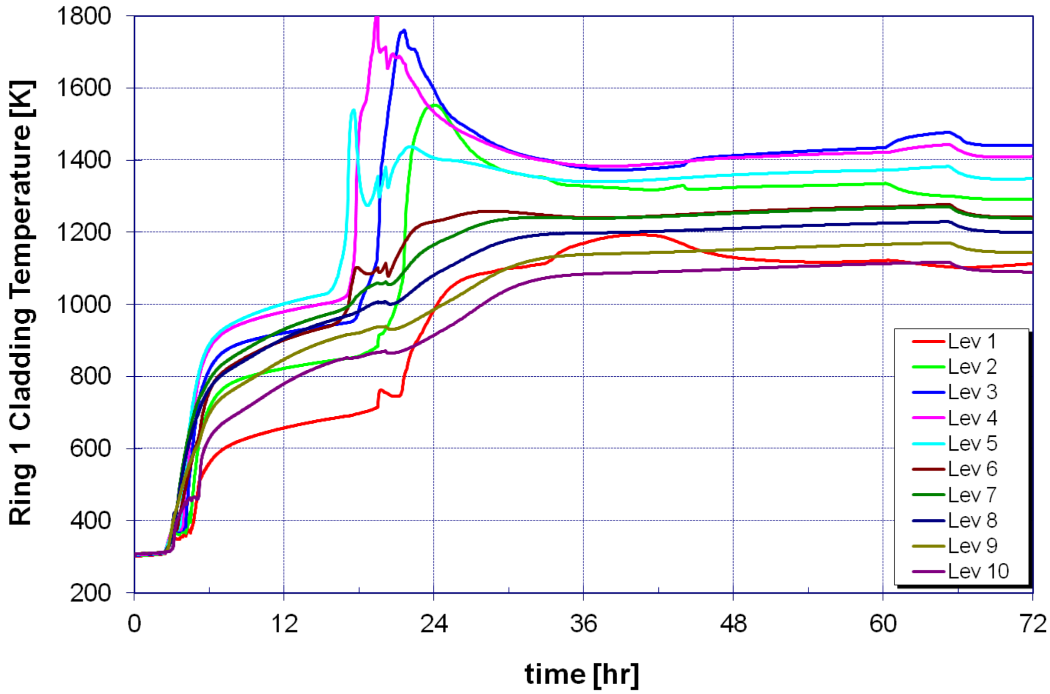

Figure 90 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP3)

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[37] However, ruthenium release differences could be higher for scenarios in OCP1 and OCP2.

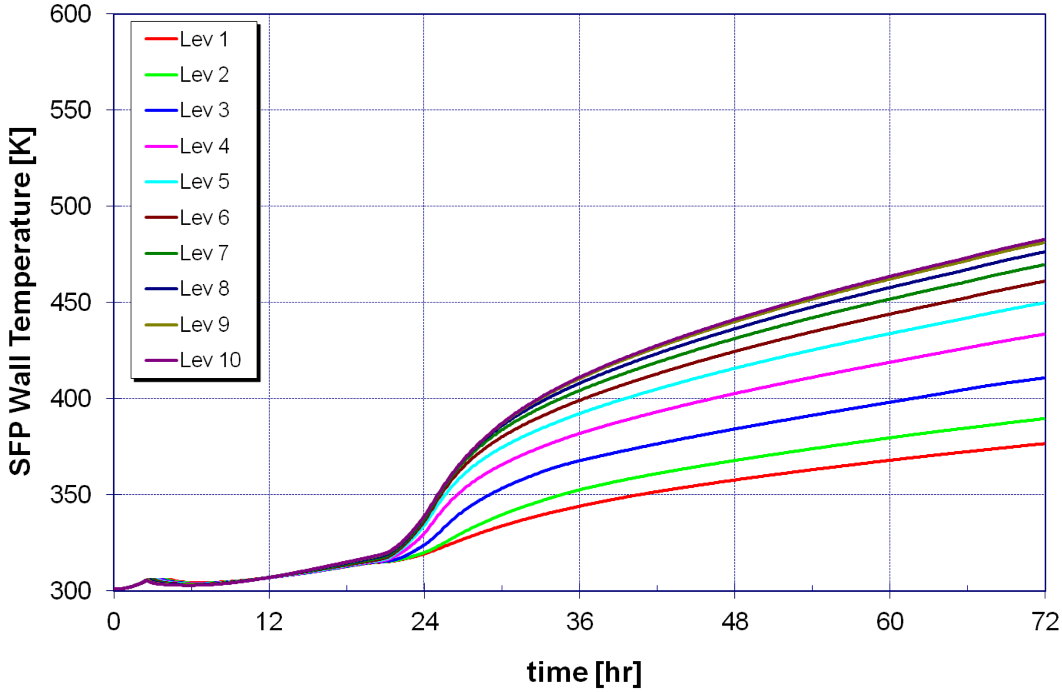


Figure 91 SFP wall liner temperature for unmitigated high-density moderate leak (OCP3)

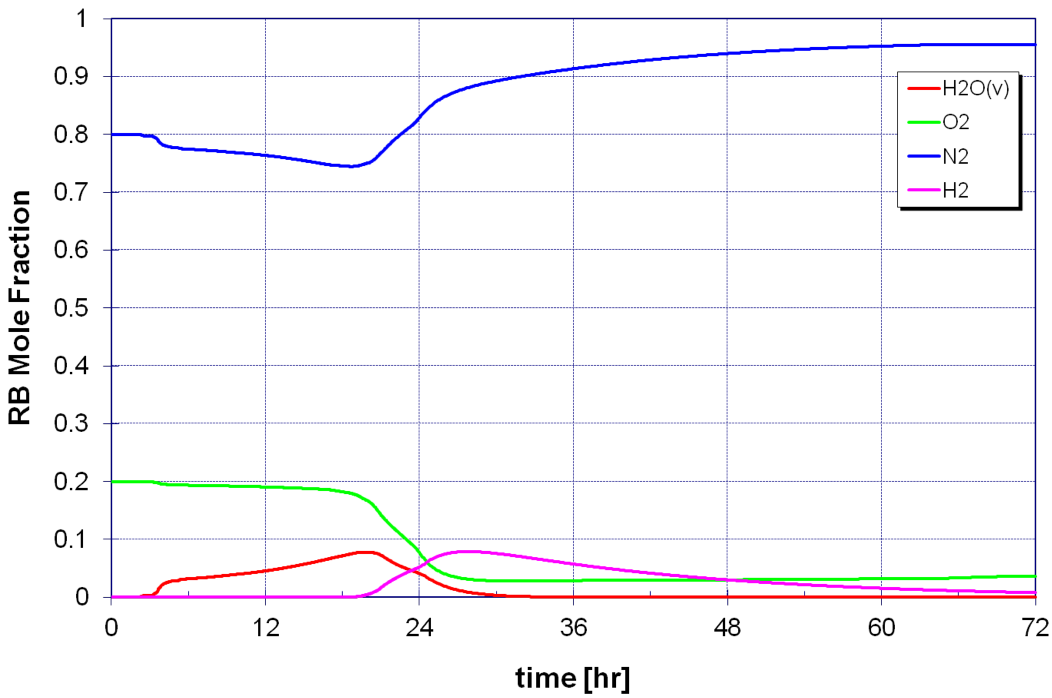


Figure 92 Reactor building mole fractions for unmitigated high-density moderate leak (OCP3)

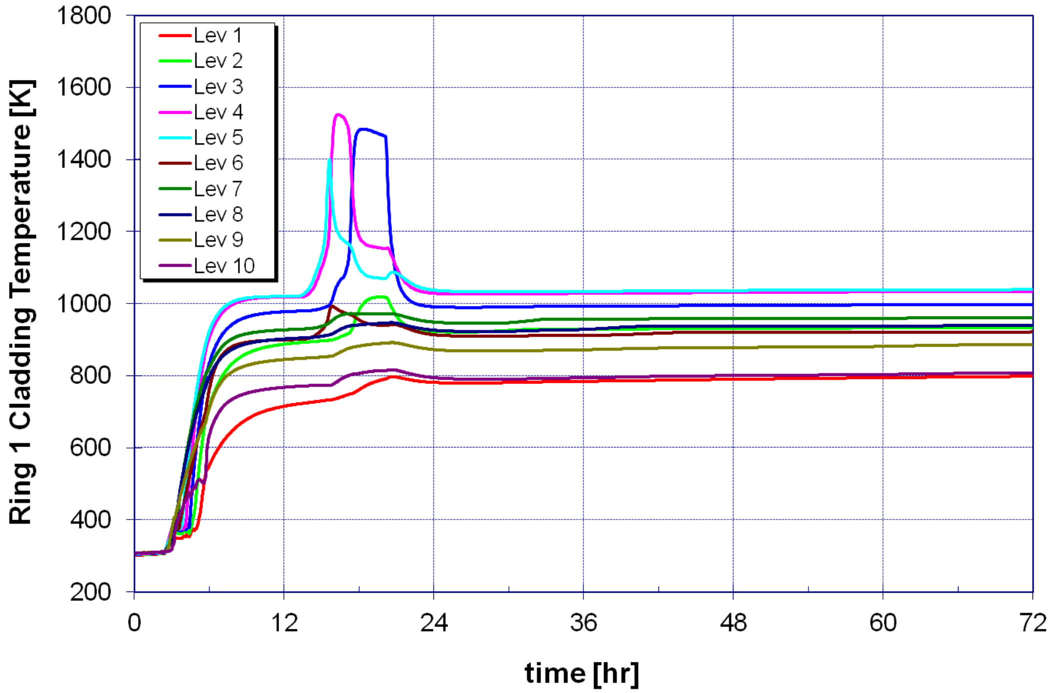


Figure 93 Ring 1 clad temperature for unmitigated low-density moderate leak (OCP3)

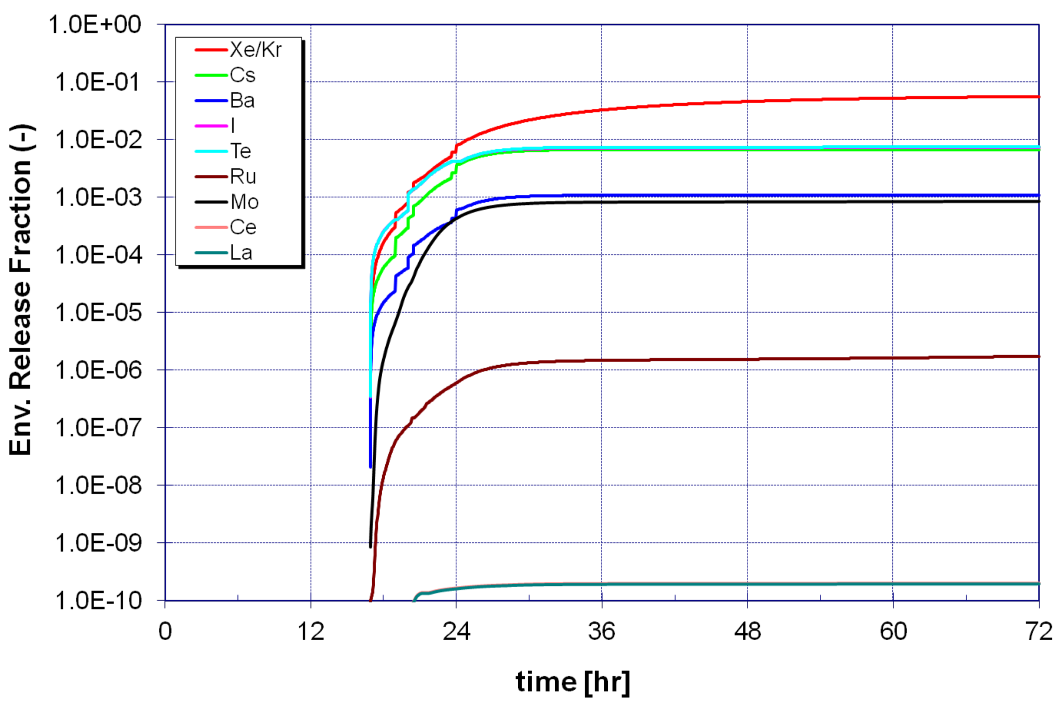


Figure 94 MELCOR default ruthenium release for unmitigated high-density moderate leak (OCP3)

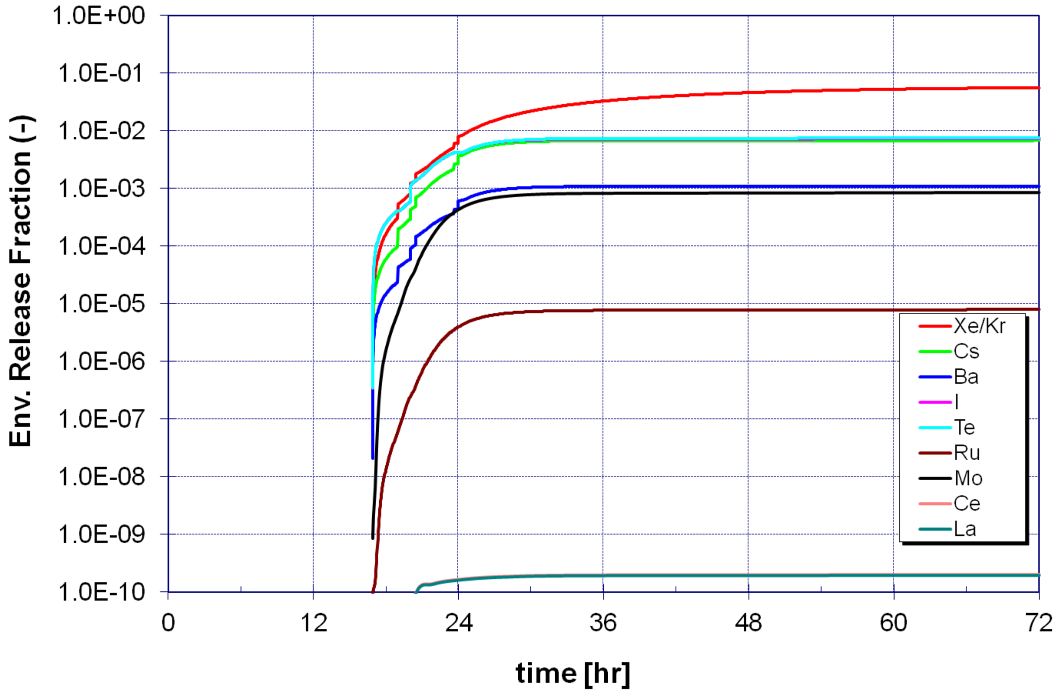


Figure 95 MELCOR enhanced ruthenium release under air oxidizing conditions for unmitigated high density moderate leak (OCP3)

6.3.3 Source Terms for Offsite Consequence Analysis

Table 27 summarizes the release characteristics and key events for the high-density scenarios, and Table 28 summarizes these factors for the low-density scenarios. Previous sections of this report provided a more detailed discussion of key phenomena for selected sequences. The releases are binned for offsite consequence analysis, which Section 7 describes.

For the high-density loading, all of the mitigated scenarios (except OCP1) have no release, either because the makeup exceeds the leak rate, as in the small leak cases, or the mitigation is successful in limiting the fuel heatup and avoiding 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 OCP1 cases which had higher release fractions than the high-density cases in some instances. This difference resulted from more rapid heatup of the fuel in Ring 1 because of less efficient heat transfer to the outer assemblies. As shown above, 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 because of a larger inventory of water (assemblies removed) resulting in longer times for clearing the baseplate. The gap release first occurs in Ring 1 (hot assemblies), which has the same decay heat in both high-density and low-density configurations.

6.3.4 Accumulation of Water Elsewhere in the Reactor Building

There are approximately 50 floor drains on the refueling floor, both at floor level and in the lower, recessed areas of the floor. The two stair towers are fully enclosed and will not be subjected to condensation. The doors to the stair towers are secondary containment doors, and so they have air seals (weather stripping, but not watertight seals). The open crane hatch in the refueling floor has a surrounding 6-in. (0.15-m) curb, so condensation on the floor will run to the floor drains and not the hatch. However, condensation forming directly over the hatch, which is 17 ft 0 in. (5.2 m) by 21 ft 9 in. (6.6 m), will fall to Elevation 135 ft (41.2 m). There is a 4-in. (0.1-m) floor drain directly under the hatchway at Elevation 135 ft (41.2 m), with no equipment in the footprint of the hatch.

If the stainless steel liner plate and 6-ft- (1.83-m-) thick reinforced concrete slab of the SFP leak through, the water will fall onto Elevation 165 ft (50.3 m) of the reactor building. Directly beneath the SFP on Elevation 165 ft (50.3 m) are the three fuel pool cooling pumps, the three fuel pool cooling heat exchangers, and the three fuel pool service water booster pumps. There are several floor drains in this area. Equipment adjacent to this area that could be affected by a large volume of water includes the fuel pool equipment panel and the reactor level and pressure instrument racks. If the floor drains on Elevation 165 ft (50.3 m) cannot keep up with the flow, then the alternate flow paths would be the crane hatch or the door to each of the two stair towers, having the same configuration as described for the refueling floor above. A significant flow rate could also affect the emergency auxiliary load centers on Elevation 165 ft (50.3 m). Water flowing over the curb of the crane hatch would reach Elevation 135 ft (41.2 m), where it would either enter the floor drains, flow through the grating to the torus room floor, or exit the building under the doors of the equipment access lock. Water reaching the stair towers would travel to the bottom of the stair tower. Water in the W stair tower would reach the residual heat removal pump room which has its own floor drains (procedurally controlled, normally closed) and sump pump. Water in the east stair tower would reach the core spray pump room or elevator shaft bottom which have floor drains (procedurally controlled, normally closed) that run to the reactor building main sump.

The reactor building MELCOR model is simplified (see Figure 42). Therefore, all water leakages corresponding to the SFP damage and draindown and overflow from water accumulation from condensation are directed to the environment. The model does not track the flow of the water and accumulation in other parts of the reactor building.

Table 27 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 release at 72 hours	Cs-137 (MCi) Released	I release at 72 hours	I-131 (MCi) Released
OCP1	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
OCP2	None	Yes							
	None	No							
	Small	Yes							
	Small	No	42.6	60.5	64.8	17.1%	7.90	17.1%	1.91
	Moderate	Yes							
	Moderate	No	5.9	11.6	No	1.6%	0.73	2.0%	0.22
OCP3	None	Yes							
	None	No							
	Small	Yes							
	Small	No	18.7	40.6	47.3	42.0%	24.20	51.2%	0.73
	Moderate	Yes							
	Moderate	No	2.5	16.9	No	0.7%	0.39	0.7%	0.01

Table 28 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
OCP1	None	Yes							
	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	0.5%	0.05	1.7%	0.13
OCP2	None	Yes							
	None	No							
	Small	Yes							
	Small	No	43.1	59.2	No	1.7%	0.28	3.3%	0.37
	Moderate	Yes							
	Moderate	No	5.9	10.5	No	0.4%	0.07	0.7%	0.08
OCP3	None	Yes							
	None	No							
	Small	Yes							
	Small	No	18.8	41.6	No	0.6%	0.10	1.2%	0.02
	Moderate	Yes							
	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 SFP (as detailed in the previous sections), a release of radioactive material from the nuclear power plant site into the atmosphere could occur. Such a release of radioactive material is expected to 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 MELCOR Accident Consequence Code System, version 2 (MACCS2) code is used to model offsite release and consequences of radioactive material. MACCS2 (SNL, 1997) has been developed by SNL for the NRC 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 can 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. Along with MACCS2, SNL has also developed WinMACCS for the NRC. WinMACCS is a user friendly graphical interface to MACCS2 that facilitates selection of input parameters and sampling of uncertain input parameters and performs post processing of results.

MACCS2 rev. 3.7.0 was used for the offsite consequence analysis in this study. In addition, many of the input values for offsite release and consequence modeling are based on approaches developed in the “State-of-the-Art Reactor Consequence Analyses” research project (NUREG-1935). These approaches are documented in greater detail in NUREG/CR-7009, “MACCS2 Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses Project”, (expected to be published in 2013), The modeling for NUREG-1935 was, in turn, based on previous studies such as NUREG-1150, an expert elicitation of the NRC/Commission of the European Communities (CEC) to update certain transport and dose parameters in NUREG/CR-7161 (NRC, 2013), and an update of the dose coefficients and dose-response modeling to be consistent with the latest Federal guidance report (FGR) at the time (FGR-13, “Cancer Risk Coefficients for Environmental Exposures to Radionuclides,” issued in 2002 (EPA, 2002)). Differences between the approaches used in NUREG-1935 and the approaches used in this study are documented below.

7.1 Offsite Consequence Modeling

7.1.1 Radiological Source Term

A source term definition for MACCS2 was created for each accident consequence calculation. The activity levels of different radionuclides from the fuel in the pool were supplied by ORIGEN calculations. The physical state of the plume, including information on the chemical group release rates, aerosol size distributions, density, and mass flow rate was supplied by MELCOR. The MELCOR analyses provided a release rate for each chemical group. Because the amount of release may differ for different sections of the pool, a new methodology was developed for this study to account for the distribution of radionuclides in the pool as well as radionuclide-specific release magnitudes. For instance, recently discharged fuel, which has more short-lived radionuclides, is more likely to release before and to greater magnitudes than older fuel. This process is described in more detail in Section 6.1.5.

Because explicit modeling with MACCS2 of all release sequences generated by MELCOR analyses is computationally expensive, the MELCOR sequences were binned by their Cs-137 and I-131 release activities (see Table 29). The first criterion used to bin the sequences was

Cs-137 release, because Cs-137 is the most significant contributor to long-term consequences. I-131 was also chosen as a criterion 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. The tally into each of these bins can be seen in Table 30.

Table 29 Release Category Types

<table>
<tr><td colspan="2" rowspan="2">Release Category Binning</td><td colspan="3">Cesium-137 Release Activity (MCi)</td></tr>
<tr><td>0 to 0.5</td><td>0.5 to 5</td><td>5+</td></tr>
<tr><td rowspan="3">Iodine-131 Release Activity (MCi)</td><td>0 to 0.25</td><td>RC11</td><td>RC21</td><td>RC31</td></tr>
<tr><td>0.25 to 0.55</td><td>RC12</td><td>RC22</td><td>RC32</td></tr>
<tr><td>0.55+</td><td>RC13</td><td>RC23</td><td>RC33</td></tr>
</table>

Table 30 Release Category Tally

Release Category	RC11	RC12	RC13	RC21	RC22	RC23	RC31	RC32	RC33	Total
Sequence Tally	5	5	0	2	0	0	0	0	2	14

One sequence was chosen from each bin (not including bins with no contributing accident sequences) to represent the entire release category, and the offsite consequences of these sequences were analyzed. The study considered a number of different factors to determine which sequence should represent each bin, including the release frequency, the relative Cs-137 and I-131 release for the bin, the start time of release, the SFP loading configuration, and the availability of the source term data (some accident progression calculations were still ongoing at the time the selection was made). In addition, because of the significant differences in release category 33 relative to the other bins, both of these sequences were analyzed, as identified in Table 31. Then, based on their conditional probabilities, all the main MELCOR sequences and their associated consequences were applied to the scenarios reported in the results, which are high-density and low-density loading both with and without successfully deployed 10 CFR 50.54(hh)(2) equipment. Sequences with no release were not included, as they do not have offsite consequences. Section 6.3 contains more information regarding which sequences do and do not have a release. For all sequences, successful deployment of 10 CFR 50.54(hh)(2) equipment prevents release of radioactive material, except for a moderate size leak during OCP1 (as defined in Section 5.2), which is when newly discharged fuel is first loaded from the reactor. Without successful deployment of 10 CFR 50.54(hh)(2) equipment, the predicted scenario-specific release frequency is 10^{-7} per year.

Table 31 Listing of Scenario-specific Release Sequences

High Density (1x4) Fuel Loading							
Unsuccessful mitigation				Deployed 50.54(hh)(2)			
Sequence		Release Frequency (/yr)[1]	Release Category	Sequence		Release Frequency (/yr)	Release Category
OCP1	small leak	6E-09[2]	RC12*	OCP1	mod leak	6E-09	RC12
OCP1	mod leak	6E-09	RC21	No Release			
OCP2	small leak	2E-08	RC33*				
OCP2	mod leak	2E-08	RC21*				
OCP3	small leak	4E-08	RC33*				
OCP3	mod leak	4E-08	RC11				
	Total	1E-07			Total	6E-09	
Low Density Fuel Loading							
Unsuccessful mitigation				Deployed 50.54(hh)(2)			
Sequence		Release Frequency (/yr)	Release Category	Sequence		Release Frequency (/yr)	Release Category
OCP1	small leak	6E-09	RC12	OCP1	mod leak	6E-09	RC12
OCP1	mod leak	6E-09	RC11	No Release			
OCP2	small leak	2E-08	RC12				
OCP2	mod leak	2E-08	RC11				
OCP3	small leak	4E-08	RC11*				
OCP3	mod leak	4E-08	RC11				
	Total	1E-07			Total	6E-09	

[1] Release frequency = initiating event frequency * ac power fragility * OCP probability * liner fragility for the specified leak size (see Section 5.6.3 for conditional probabilities)
[2] Example calculation: 1.7×10^{-5}/yr · 0.84 · 0.0086 · 0.05 = 6×10^{-9}/yr
* Sequences marked with an (*) were used in MACCS2 analysis

7.1.2 Atmospheric Modeling and Meteorology

The atmospheric transport and dispersion model in MACCS2 is a straight-line Gaussian plume segment dispersion model. For this study, the atmospheric release of radionuclides is discretized into (at longest) 1-hour plume segments. This accounts for variations in the release rate, as well as for changes in wind direction. More plume segments increase the resolution of the dispersion modeling to the point the resolution corresponds to the time resolution of the weather data, because each segment can travel in a compass direction representative of the actual weather data at the time the plume segment is released.

A set of aerosol deposition velocities, combined with the aerosol size distribution from MELCOR, determines the rates aerosols are deposited from the plume to the ground. 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 microns in diameter, which corresponds to a deposition velocity of about 4 or 5 millimeters per second. Dry deposition velocities have been updated to account for a more typical surface roughness of 60 cm for the reference plant site. (A surface roughness of 10 cm was used for NUREG-1935 and a 60-cm surface roughness was considered in a sensitivity calculation.) The relative aerosol deposition velocities, as well as much of the non-site-specific data for acute health effects, are developed from NUREG/CR-7161 (NRC, 2013).

Because the exact weather conditions that would apply in the case of a potential accident in the future cannot be known in advance, MACCS2 accounts for weather variability by analyzing a statistically significant set of weather trials. Thus, the modeled results are ensemble averages of weather that represent of the full spectrum of meteorological conditions. The weather-sampling strategy for this study uses a nonuniform weather-binning approach. Weather binning is an approach used in MACCS2 to categorize similar sets of weather data based on windspeed, stability class, and the intensity and timing of precipitation. This sampling strategy was chosen to represent the statistical variations of the weather. Further discussion on this approach can be found in NUREG/CR-7009.

Meteorological data used for this project consisted of one year of hourly meteorological data (8,760 data points for each meteorological parameter). The data are from onsite meteorological tower observations are the same as those used in NUREG-1935. The site selected for the reference plant provided two 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 meteorological data chosen for the reference plant was 2006, which was based on data recovery (greater than 99 percent being desirable) as documented in NUREG/CR-7009. Different trends (e.g., wind rose pattern and hours of precipitation) between the years were estimated to have a relatively minor (less than 25 percent) effect on the final NUREG-1935 results. More specific details of the weather data can be found in NUREG/CR-7009.

7.1.3 Exposure, Dosimetry, and Health Effects Modeling

MACCS2 considers groundshine, cloudshine, inhalation, and ingestion exposure pathways. The principal exposure pathway to members of the public occupying land contaminated by atmospheric deposition of radioactive materials is expected to be exposure of the whole body to external gamma radiation. Although it is normally expected to be of lesser importance, the inhalation pathway contributes additional doses to internal organs (EPA, 1992), especially during the emergency phase of the accident. The MACCS2 outputs for health effects and population dose include doses from exposure via the ingestion pathway. However, the MACCS2 code does not represent these consequences in the individual LCF risk results[38]. Food ingestion parameters were chosen to be consistent with Sample Problem A, as documented in NUREG/CR-6613, Vol. 1 (Chanin and Young, 1998). Sample Problem A is based on NUREG-1150, with the exception that newer options not included in the older MACCS model were used to demonstrate new capabilities in MACCS2 (e.g., that the food ingestion model is updated to use the newer COMIDA2 rather than the original MACCS food-chain model). NUREG-1935 did not include exposure to contaminated food because staff judged it not to be a significant contributor to individual risk.

NUREG/CR-7009 reviews the shielding factors applied to evacuation, normal activity, and sheltering for each dose pathway (e.g. groundshine) used in NUREG-1150 (NRC, 1990) and NUREG/CR 6953, Volume 1, “Review of NUREG-0654, Supplement 3, ‘Criteria for Protective Action Recommendations for Severe Accidents’—Focus Group and Telephone Survey,” issued

[38] Including the ingestion pathway is predicted to increase health effect risks in this study by about 5% with an LNT dose response model, depending on the scenario.

October 2008 (NRC, 2008c). This study uses the same shielding factors as updated in NUREG/CR-7009.

The site file containing population and economic data was created for 16 compass sectors and then interpolated onto a 64 compass-sector grid for better spatial resolution for the consequence analysis. Site population data have been projected to the target year 2011 using the latest version of the computer code SECPOP2000 (SNL, 2003). SECPOP2000 uses 2000 census data and applies a multiplier to account for population growth and an economic multiplier to account for the value of the dollar to create site data for MACCS2. A multiplier value of 1.1051 from the U.S. Census Bureau was used to account for the average population growth in the United States from 2000 to 2011. Consistent with the approach used in NUREG-1935, the economic values from the database in SECPOP2000 (which uses an economic database based on the year 2002) were scaled to account for price escalation between the years 2002 and 2011. A scaling factor of 1.250 was derived based on the Consumer Price Index (CPI).

Consistent with NUREG-1935, the dose and risk coefficients and relative biological effectiveness values used in this study are based on FGR-13 (EPA, 2002). The dose coefficients allow organ-specific doses to be calculated from exposure to radiation. The risk factors in FGR-13 are based on the risk coefficients for the U.S. population detailed in the BEIR V report (NAS, 1990). As implemented in MACCS2, these factors include seven organ-specific cancers plus residual cancers not accounted for directly. The inhalation factors in FGR-13 were processed to account for a distribution of particle sizes. An activity median aerodynamic diameter of 1 micron was assumed with a log-normal form for the distribution and with a geometric standard deviation of about 2.5. Parameters that relate to acute health effects in this study, as well as much of the nonsite-specific data used for consequence analysis were taken from NUREG/CR-7161 (NRC 2013). All of the input parameters extracted from the expert elicitation are median values.

The FGR-13 coefficients, as implemented in MACCS2, include a dose and dose rate effectiveness factor (DDREF), which has been incorporated in the dose-response modeling for the long-term phase of the offsite consequences and to the dose-response modeling for the early-phase (i.e. the first week) for doses less than 20 rem. This factor accounts for the fact that protracted, low doses are estimated to be less effective in causing cancer than more acute doses. The DDREFs for all cancer types, except for breast, were 2.0; the DDREF for the breast was 1.0, as in NUREG/CR-7009.

To provide perspective on uncertain low-dose health effects, the results also include dose truncations that limit the quantified health effects to those arising from higher doses. Dose truncation values used here include 620 mrem/year (representative background radiation including average annual medical exposures), and 5 rem/year with a 10-rem lifetime cap (based on the Health Physics Society's position that there is a dose below which, because of uncertainties, a quantified risk should not be assigned). This approach is consistent with the approach used in NUREG-1935.

7.1.4 Emergency Response Modeling

The MACCS2 models were set up to calculate exposures in two distinct phases: the emergency phase and the long-term phase. The emergency-phase models calculate the dose and associated health effects to the public, as well as the effects of emergency preparedness actions that protect the public. The emergency phase is defined as the seven day period following the start of the release.

The staff modeled offsite response organization (ORO) decision making based upon the accident sequences, timing, radiological release, and knowledge of response activities and the availability of emergency response technical support. Since actions beyond the emergency planning zone (EPZ)[39] would be 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 SFPS 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.

For each of the accident sequences, staff determined that a General Emergency 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. A release from a SFP with a moderate leak begins earlier than a damage state with a small leak, but these still do not begin until evacuation is well underway or completed within the EPZ.

A number of different protective actions can be modeled in MACCS2. The residents are modeled as groups (known as cohorts) and have different types of protective actions and associated response timings. The actions that can be taken include:

<u>Shelter-in-place (SIP):</u> For certain areas where dose may be reduced below the PAG through sheltering, SIP is modeled as an expected protective action consistent with the emergency plans. In other areas, sheltering can occur prior to evacuation.

<u>General Public Evacuation:</u> Residents evacuate the affected area when the official order to evacuate is received.

<u>Early Evacuation:</u> Residents evacuate after the earthquake, but before the official order to evacuate is received.

<u>Shadow Evacuation</u>: Residents evacuate from areas that are not under an official evacuation order. A shadow evacuation typically begins when a large scale evacuation is ordered (NRC, 2005b). 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, 2008c). In the SFP project, the initiating event is an earthquake that would be felt by residents of the EPZ. The event would be followed with media information related to an accident at the nuclear power plant, widespread loss of power and damage to some buildings. It was assumed that these factors would increase the shadow evacuation to 30 percent of the public in the environs of the plant.

[39] EPZ in this study refers to the plume exposure pathway EPZ with a radius of about 10 miles from the reactor site. This should not be confused with the ingestion exposure pathway EPZ with a radius of about 50 miles from the reactor site.

<u>Hotspot and Normal Relocation:</u> Models are included in the MACCS2 code to reflect emergency relocation of people from areas that were not included in the evacuation order where the dose exceeds emergency-phase PAGs. Within the MACCS2 calculation, individuals who would be relocated because their projected total committed dose is projected to exceed the protective action criteria are assumed to not receive any additional dose following relocation for the duration of the emergency phase. This relocation dose criterion is applied at a specified time after plume arrival within the affected area and is applied to the entire population within the analysis area, including the nonevacuating cohort (0.5% of the population) within the EPZ. The hotspot and normal dose and time values were developed for each evacuation model. They were established with the assumption that relocation begins after the evacuation is substantially complete which depends on the timing of the first plume for each sequence. For the larger release sequences which affect areas beyond 30 miles, the normal relocation time was assumed to be 12 hours after the hotspot relocation time. This assumption provides time for offsite response organizations to address the higher priority hotspot areas first.

The detailed emergency plans developed for the EPZ provide a substantial basis for expansion of response efforts if necessary (NRC, 1980a). This study identified many potential accident sequences and performed preliminary consequence modeling to establish baseline dose projections as a function of distance. This information was used to develop the appropriate emergency response parameters for the release being modeled. The distance to which the PAG may be exceeded assisted in determining the extent of offsite protective actions and the type of protective actions (sheltering or evacuation) that would be implemented. In the event of model predictions of elevated doses at distances beyond the plume exposure pathway EPZ, a review of the State emergency response plans was performed to determine the types of protective actions that would be implemented in these areas. The results of the dose projections were binned based on the EPA's PAGs (EPA, 1992) to support an efficient use of detailed consequence modeling to determine the potential effects of such accidents. For this analysis, the PAG was considered to be exceeded if the four day projected dose is expected to exceed one rem for a member of the public. Using the dose projections, the three evacuation models presented in the table below were developed for analysis. Detailed information on the implementation of these evacuation models is provided in APPENDIX A: .

Table 32 Summary of Evacuation Models

Evacuation Model	4-Day Dose Projection	EPZ	Area beyond EPZ
1	Small: Does not exceed PAG beyond EPZ.	General public evacuation, including early evacuation of 30% of the public.	Shadow evacuation of 30% of the public from immediately beyond the evacuation area. Hotspot relocation is 5 rem at 4 hours after plume arrival. Normal relocation is 1 rem at 8 hours after plume arrival.
2	Large (48 hour): Exceeds PAG beyond EPZ.	General public evacuation, including early evacuation of 30% of the public.	Shadow evacuation of 30% of the public from immediately beyond the evacuation area. Delayed evacuation to a distance of 30 miles. Shelter in place (SIP) for the 30 to 40-mile area. Shadow evacuation of 20% of the public from the SIP area. Hotspot relocation is 5 rem at 4 hours after plume arrival. Normal relocation is 1 rem at 16 hours after plume arrival. (Rapid implementation of relocation is based on having 48 hours to prepare before release begins)
3	Large (24 hour): Exceeds PAG beyond EPZ.	General public evacuation, including early evacuation of 30% of the public.	Shadow evacuation of 30% of the public from immediately beyond the evacuation area. Delayed evacuation to a distance of 30 miles. Shelter in place of 30 to 40-mile area. Shadow evacuation of 20% of the public from this Shelter-in-place (SIP) area. Hotspot relocation is 5 rem at 26 hours after plume arrival. Normal relocation 1 rem at 38 hours after plume arrival.

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 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. The evacuation time estimate was used to inform the development mobilization times and travel speeds for the public. To model evacuation in MACCS2, each cohort was loaded onto the roadway network at a specified time, and a set of speed values were applied per cohort for the early, middle and late periods of the evacuation. 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. The rate of evacuating the public is 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) and was modeled as a separate cohort.

An assessment of travel distance and time was initially 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 necessarily oriented directly outward from the plant. Consistent with the location of the reference plant, 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 offsite response organizations from neighboring states would adopt the same protective action decisions.

The following general assumptions were applied in this analysis:

- The EPZ is modeled as the area within 10 miles of the site, as an approximation.
- Protective actions would be implemented within the EPZ were an accident to occur.
- Protective actions would be expanded beyond the EPZ as necessary.
- Dose projections would be developed and available to support protective action decisions.
- Residents would expect they cannot return and would take more belongings with them, than what was considered in the past, e.g. NUREG/CR-7009, thereby increasing mobilization times.
- Residents would generally be aware of an impending emergency through media broadcasts.
- For the delayed release sequences in which a releases do not start for more than 24 hours, schools beyond the EPZ would be closed rather than evacuated.
- Evacuees are transported to safe distances.
- There is no loss of power beyond 20 miles. Communications, traffic signals, and emergency alert system messaging are not impacted in this area.

The chosen time period for the emergency phase begins with the initiating event and continues for one week following the initial release. This assumption gives time for the plume to pass and deposit radioactive material onto the ground so that all the calculated acute exposures are captured. The one-week period for the emergency phase is different than the four-day period used for emergency-phase dose projections, which were used to inform the evacuation models. The four-day period was chosen to be consistent with the EPA PAGs (EPA, 1992).

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 chosen to be radially outward to simplify modeling of evacuation in these areas. Speed adjustment factors were applied at the grid element level to speed up vehicles in the rural uncongested areas and to slow vehicles in more urban settings in which the modeling indicates that speeds are lower than the average values used in the analyses.

The MACCS2 potassium iodide (KI) model used in this analysis assumes that KI would be distributed only within the EPZ. Half the residents within the EPZ are assumed to have access to their KI and to take it within the specified timeframe.

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 an evacuation-speed multiplier to reduce travel speed when precipitation is occurring (indicated from the meteorological data file). The evacuation speed multiplier factor was set to be 0.7, which effectively slows down the evacuating public to 70 percent of the fair-weather travel speed when precipitation exists.

The analyses of the effect of the seismic event on emergency response developed for NUREG-1935 were applied in this analysis, as the reference plant in this study was one of the plants studied in NUREG-1935. The evaluations of the potential failure of roadway infrastructure conducted for NUREG-1935 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 roads would have 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 is assumed to cause the loss of all onsite and offsite power within the EPZ, which can affect the response timing and actions of the public. Sirens would be sounded following the GE declaration, and because the reference plant will have a fully backed up siren system in 2013, it is assumed sirens sound for this analysis. The residents within the EPZ would have felt the earthquake, which effectively serves as the initial warning; however, the loss of power would affect the number of residents receiving instructions via emergency alert system messaging. It is expected that the residents use multiple methods of communication, such as cell phones, telephones, websites (where power is available), 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 and that most intersections are controlled with stop signs. The loss of power would cause traffic signals to default to a four-way stop mode, which is less efficient than normal signalization. It is expected that emergency response personnel would respond to these intersections and direct traffic as indicated in the site evacuation time estimate. Therefore, the loss of signalization should have a limited impact on the evacuation. It is assumed that at distances beyond 20 miles, there is no loss of power and traffic signals, and emergency alert system messaging is not impacted.

7.1.5 Long-Term Protective Action Modeling

The long-term phase is the period following the seven-day emergency phase and is modeled for 50 years. The 50 year duration of the long-term phase has been chosen to provide a reasonable time period for calculating consequences from exposure for the average person. Exposure is mainly from external radiation from trace contaminants that remain after the land is decontaminated, or in lightly contaminated areas where no decontamination was required. However, internal exposures may also occur during this period, including inhalation of resuspended radionuclides and ingestion of food and water with trace contaminants. Depending on the relevant PAGs and the level of radiation, food and water below a certain limit could be considered adequately safe for ingestion, and lightly contaminated areas could be considered habitable.

A long-term cleanup policy for recovery after a severe accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be decided by a number of local, state, and federal jurisdictions and would most likely 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 may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations. Three protective actions were modeled to occur for contaminated land during the long-term phase: interdiction, decontamination, and condemnation. As used in the MACCS2 model, interdiction and condemnation refer to the relocation of people from contaminated areas according to the habitability criterion. Interdiction is defined in the MACCS2 model as temporary relocation during which the contamination levels are reduced by decontamination, natural weathering, and radioactive decay. Condemnation is defined in the MACCS2 model as a permanent relocation when contamination levels cannot be adequately reduced by decontamination, natural weathering, and radioactive decay.

Decontamination is modeled in a manner consistent with both NUREG-1150 and NUREG-1935. Two levels of decontamination (a decontamination factor of 3 and 15) are each assumed to take one year, but the cost of the higher decontamination factor (15) is assumed to be greater, reflecting the greater effort needed to achieve the higher level of decontamination. This study uses the values in NUREG/CR-7009 for the cost of decontamination. During the decontamination period, the land is temporarily interdicted (e.g. the population is temporarily relocated), and may be interdicted for an additional period to allow for radioactive decay and natural weathering to reduce contamination levels if needed to restore habitability. If land cannot be restored to habitability in 30 years, the MACCS2 model defines the land as condemned and residents are modeled to not return during the long-term phase. The MACCS2 models assume that decontamination will only take place if it is projected to make 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 is projected to be higher than the land value, the individuals on that land are assumed to be permanently relocated. Because both the land values and the level of decontamination affect decisions on whether contaminated areas can be restored to habitability, they affect predicted long-term doses, health effects, and economic costs.

Site-specific values are used to determine long-term habitability, whereas farmability is defined to be consistent with NUREG-1150. For 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, consistent with the location of the reference plant, the analysis includes the State of Pennsylvania position using a habitability criterion of 500 mrem per year beginning in the first year, which is the value that is used for this study. For consistency and practicality reasons, the same standard for estimating habitability is applied to all affected areas in this study. The values used to define farmability were taken from NUREG-1150. During the year of the accident, the allowable committed dose equivalent from consumption of dairy products to an organ or tissue is 2.5 rem (7.5 rem for the thyroid), as well as an additional dose of 2.5 rem (7.5 rem for the thyroid) for all other foods. In subsequent years, the maximum allowable dose to the organ or tissue from all foods, including dairy products, is 500 mrem (1.5 rem for the thyroid). Agricultural lands projected to be contaminated to such an extent that agricultural products would exceed these levels are defined to be unfarmable, and the crops growing on these lands at the time of the accident are assumed to be disposed. No farming is allowed until the farmability criterion is satisfied.

7.2 Offsite Consequence Results

Several consequence metrics have been selected to characterize the impacts resulting from a severe spent fuel pool accident. Individual risk of early and latent cancer fatality, as well as societal risk of latent cancer fatalities, are measures of the radiological health impact of the accident and consistent with NRC's safety goals (NRC, 1986). In this study, collective dose is used as a surrogate for the societal impact of latent cancer fatalities. In addition, certain metrics that would influence the values considered by the NRC in regulatory analysis and documented in NUREG/BR-0058, such as measures of offsite property damages, the number of displaced individuals (either temporarily or permanently), and the extent over which such actions may be needed, are also presented. These metrics provide a benchmark for understanding the nature and extent of a severe spent fuel pool accident. These measures are subject to considerable uncertainty, as the details of how long-term protective actions would be carried out would have a significant effect on the actual values reported herein.

All results presented in this section are conditional upon a pool leak following a specified severe (0.71g peak ground acceleration) seismic event on the SFP at the reference plant. In the event of a pool leak following a severe seismic event, a number of potential outcomes could occur, depending upon when in the operating cycle the event occurred, the severity of the leak, and whether effective mitigation (in the form of either makeup water or pool sprays) was able to be successfully deployed prior to the beginning of the release. Staff has evaluated the likelihood of these different conditions. The relative likelihood of a seismic event during a particular operating cycle phase is simply proportional to the duration of the phase. The relative likelihood of significantly different leak rates is discussed in Section 4. Because these probabilities can be quantified with a reasonable degree of certainty, the offsite consequence results are weighted by the relative likelihood of these factors to yield an average over the different operating cycle phases and leak rates.

In contrast, the likelihood of successful deployment of 10 CFR 50.45(hh)(2) mitigation has not been quantified. NRC staff judgment is that the likelihood of successful mitigation can in many cases be high, but that it is affected by a number of factors that are difficult to quantify (see Section 5.3). Related to this, a human reliability assessment (HRA) is provided in Section 8. Although the HRA does not provide a quantitative value required to determine the overall likelihood of mitigation, it does provide significant insights into the likelihood of mitigation during this seismic event for certain damage states. To quantify the overall likelihood of successful mitigation, a PRA type analysis would be required. For this reason, the results of the study are presented as a range of mitigation effects related to successfully deployed mitigation and mitigation that is unsuccessful for 3 days.

This analysis examines the relative effects of a low-density and a high-density fuel loading configuration. Therefore, results are reported for two configurations, those being a high-density loading case with a 1x4 loading pattern and for a low-density loading case with a mixture of 1x4 and checkerboard loading patterns, as portrayed in Figure 44 through Figure 48.

In this chapter, the results for each selected metric are discussed for each loading configuration (high-density and low-density). In addition, the factors that affect the results and how those results vary with dose truncation assumptions and with distance are discussed. To the extent possible, the relationship between the results presented here and the results obtained in previous studies is discussed.

Table 33 Overall Consequence Results

SFP Fuel Loading	High Density (1x4)		Low Density	
Seismic Hazard Frequency[1] (/yr) (PGA of 0.5 to 1.0g)	1.7E-05		1.7E-05	
50.54(hh)(2) Mitigation Credited	Yes	No	Yes	No
Conditional[2] Probability of Release	0.036%	0.69%	0.036%	0.69%
Hydrogen Combustion Event	"Not Predicted"	"Possible"	"Not Predicted"	"Not Predicted"
Conditional[3] Consequences (Release Frequency-Averaged[4])				
Cumulative Cs-137 Release at 72 hours (MCi)	0.26	8.8[(8)]	0.19[(7)]	0.11
	Measures Related to Health and Safety of Individuals			
Individual Early Fatality Risk	0	0	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles	3.4E-04	4.4E-04	3.4E-04	2.0E-04
	Measures Related to Cost Benefit Analysis			
Collective Dose (Person-Sv)	47k	350k	47k	27k
Land Interdiction[6] (mi^2)	230	9,400	230	170
Long-term Displaced Individuals[6]	120k	4,100k	120k	81k
Consequences per year (Release Frequency-Weighted[4])				
Release Frequency (/yr)	6.1E-09	1.2E-07	6.1E-09	1.2E-07
	Measures Related to Health and Safety of Individuals			
Individual Early Fatality Risk (/yr)	0	0	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles (/yr)	2.1E-12	5.2E-11	2.1E-12	2.4E-11
	Measures Related to Cost Benefit Analysis			
Collective Dose (Person-Sv/yr)	2.9E-04	4.1E-02	2.9E-04	3.2E-03
Land Interdiction[6] (mi^2/yr)	1.4E-06	1.1E-03	1.4E-06	2.0E-05
Long-term Displaced Individuals[6] (Persons/yr)	7.1E-04	4.9E-01	7.1E-04	9.5E-03

[1] Seismic hazard model from USGS (Peterson et al., 2008)
[2] Given specified seismic-event occurs
[3] Given atmospheric release occurs
[4] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions (as applicable); additionally, "release frequency-weighted" results are multiplied by the release frequency.
[5] LNT and population-weighted
[6] 1st year post-accident; calculation uses a dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51
[7] Mitigation can moderately increase release size (see Section 6.3); the effect is small compared to the reduction in release frequency.
[8] Largest releases here are associated with small leaks (although sensitivity results show large releases are possible from moderate leaks). Assuming no complications from other SFPs/reactors or shortage of available equipment/staff, Section 8 shows that there is a good chance to mitigate the small leak event.

7.2.1 Individual Early Fatality Risk

For all scenarios, no offsite early fatalities attributable to acute radiation exposure are predicted to occur. Due to radioactive decay, spent fuel pools tend to have significantly less shorter-lived radionuclides (e.g. I-131) than reactors. Despite this, in at least one case that was analyzed, doses close to the site did reach levels that can induce early fatalities. Therefore, the potential (although remote) for early fatalities exists. However, emergency response as treated in this study effectively prevents any early fatality risk from acute radiation exposure, at least in part because the modeled accident progression results in releases that are long compared to the implementation of emergency response in the areas of most concern.

The projections of no early fatalities in this study is lower than that reported in some previous studies of risks from spent fuel pool accidents, such as NUREG/CR-6451 and NUREG-1738, and consistent with the earlier studies documented in NUREG-1353. Tables 4.1 and 4.2 of NUREG/CR-6451 project anywhere from approximately one to one hundred early fatalities within a 500 mile radius in the event of an accident involving the full spent fuel pool, with the higher values associated with high release fractions. NUREG-1738 (Table 3.7-1 and Table 3.7-2) reported similar values, ranging from no fatalities for low Ruthenium source terms with early evacuation to up to 192 early fatalities for an accident shortly (30 days) after shutdown with high Ruthenium source terms and late evacuation. NUREG-1353 does not provide quantitative estimates of early fatality risk but states that "…there are no "early" fatalities and the risk of early injury is negligible". On balance, the scenarios analyzed here are consistent with the lower end of the reported range from previous studies, in that no early fatalities are projected to occur.

7.2.2 Individual Latent Cancer Fatality Risk

Despite the large releases in certain circumstances, the risk of latent cancer fatality to the average individual within 10 miles of the plant is low. When averaged over the likelihood of different event timings and leak sizes, the conditional risks within 10 miles are in the 1E-04 to 1E-03 range for cases both with and without successful 50.54(hh)(2) mitigation and for high-density and low-density cases, when using an LNT dose response model. This range does not appreciably increase even if the releases for different leak sizes or operating cycle phases are shown separately.

Individual latent cancer fatality risk is low because:

- The predicted release frequency of this event is very small
- Protective actions, especially those for long-term chronic doses, are estimated to avert significant amounts of public exposure.

Because of the nature of the event, this risk is predominantly from long-term chronic exposures. With effective long-term protective measures (e.g. temporary and permanent land interdiction), essentially no individuals receive any long-term risks greater than those associated with the dose limits for protective actions. Therefore, independent of the release magnitude of the event, these dose limits form an upper limit to individual long-term risk. In addition, emergency response is assumed to be very effective in evacuating and relocating the public. For instance, individuals within the 0-10 mile distance (representative of the plume exposure pathway EPZ) essentially only receive LCF risk if they return to low risk, habitable areas. The conditional individual LCF risks within ten miles are comparable to or lower than the projections from earlier studies of spent fuel pool accident risk. For example, NUREG-1738 reports conditional individual latent cancer fatality risks ranging from 8E-4 to 8E-2 for a range of initiating events where large seismic events contributed the most to the overall estimate of risk. These conditional risks were driven largely by the previous estimates of ruthenium volatility and by the effectiveness of evacuation.

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 and related to a single spent fuel pool, staff concludes that since these risks are several orders of magnitude smaller than 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).

Because the health effects that would be induced by low dose radiation are uncertain, staff performed a sensitivity analysis to understand how the risks would change if computed health risks were limited to those arising from higher doses. The upper truncation level used in this sensitivity analysis corresponds to a treatment consistent with the HPS position statement (5 rem annually and 10 rem lifetime). The second truncation level corresponds to the average annual dose to the public from medical and background radiation exposures in the United States (620 mrem annually).

Using truncation levels that do not quantify the effects of doses below the dose levels chosen here significantly reduces the estimated individual LCF risk. This is because individual LCF risk using an LNT dose response model mainly comes from doses less than those specified in protective action guidelines. Table 34 (which shows risk to residents living within ten miles, not including risk from ingestion or risk to decontamination workers) shows the use of the dose truncations that are analyzed here lowers the estimated individual LCF risk within 10 miles by several orders of magnitude. Because the dose truncations are greater than the dose limits for land interdiction, it is difficult for doses from the long-term phase to contribute to the quantified LCF risk. Therefore, emergency-phase exposures play a more significant role in the doses that exceed the truncation levels. However, the amount of early phase exposures that exceed the dose truncations is very small within 10 miles because emergency response is effective in protecting the evacuees.

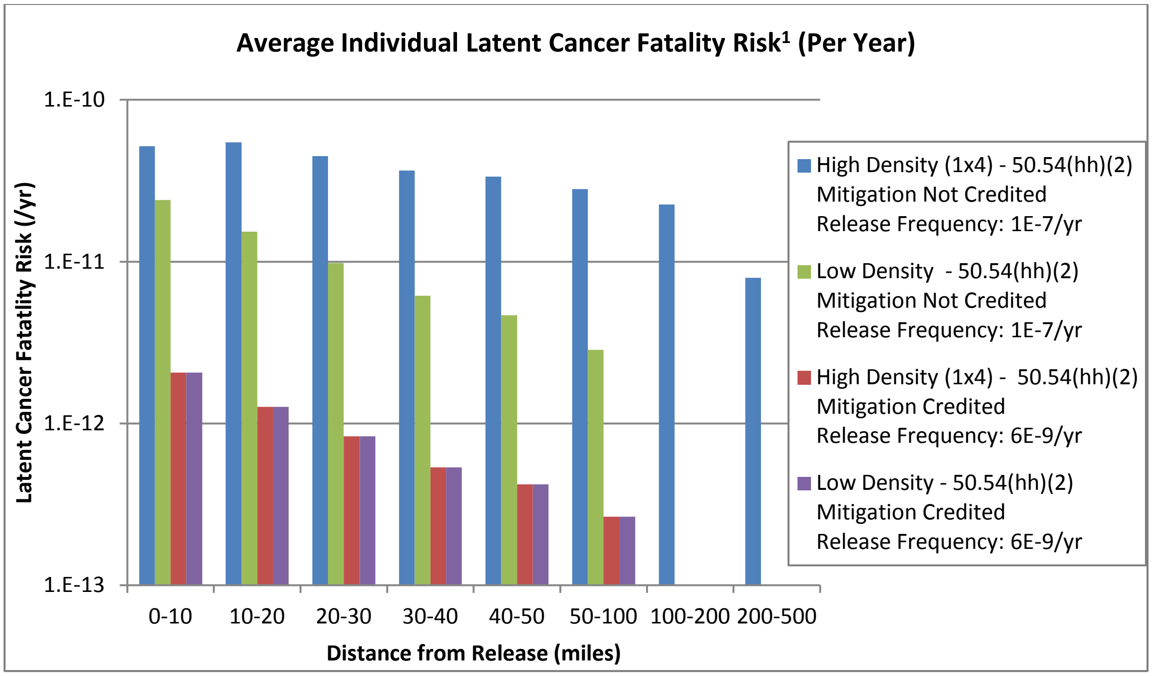
A number of factors can affect quantified individual LCF risks, particularly the very small values from dose truncation results. These include potential variations of the real application of protective actions, different protective action levels, or consideration of ingestion doses. Nevertheless, the overall conclusions that with an LNT calculation, individual LCF risk is mainly from long-term chronic exposures, and that dose truncations significantly lower the estimated individual LCF risk, remain valid.

Table 34 Dose-Response Model Results (LNT) and Dose Truncation Comparison

Dose-Response	High Density (1x4)		Low Density	
50.54(hh)(2) Mitigation Credited	Yes	No	Yes	No
Conditional[1] Individual Latent Cancer Fatality Risk Within 10 Miles (Release Frequency-Averaged[2])				
Linear, No Threshold	3.4E-04	4.4E-04	3.4E-04[(3)]	2.0E-04
620 mrem/yr truncation	6.1E-08	1.2E-07	6.1E-08[(3)]	3.4E-08
5rem/yr or 10rem lifetime truncation	1.8E-08	1.4E-07	1.8E-08[(3)]	5.6E-09
Individual Latent Cancer Fatality Risk Within 10 Miles (/yr) (Release Frequency-Weighted[2])				
Linear, No Threshold	2.1E-12	5.2E-11	2.1E-12	2.4E-11
620 mrem/yr truncation	3.8E-16	1.4E-14	3.8E-16	4.0E-15
5 rem/yr or 10 rem lifetime truncation	1.1E-16	1.6E-14	1.1E-16	6.6E-16

[1] Conditional on a release occurring
[2] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions; additionally, “release frequency-weighted” results are multiplied by the release frequency.
[3] Mitigation can moderately increase release size (see Section 6.3); the effect is small compared to the reduction in release frequency.

While individual latent cancer fatality risk is very low, it decreases slowly with distance, particularly for large releases such as may occur from an accident in a high-density pool with unsuccessful mitigation for 3 days. This is because offsite release models predict significant spread of contamination to far distances, mainly because of the slow deposition of aerosols from the plume. Increasing the magnitude of the release extends the range over which a plume can travel before the radioactive inventory of the plume is significantly depleted by deposition. Furthermore, because protective actions such as land interdiction are modeled to occur wherever the model predicts that the dose limits are exceeded, most distances are held to comparably low levels of individual LCF risk regardless of the magnitude of the deposition, as was seen in the results for individual LCF risks in Table 33. This can be seen in Figure **96**, which like the table, is also weighted by the release frequency.



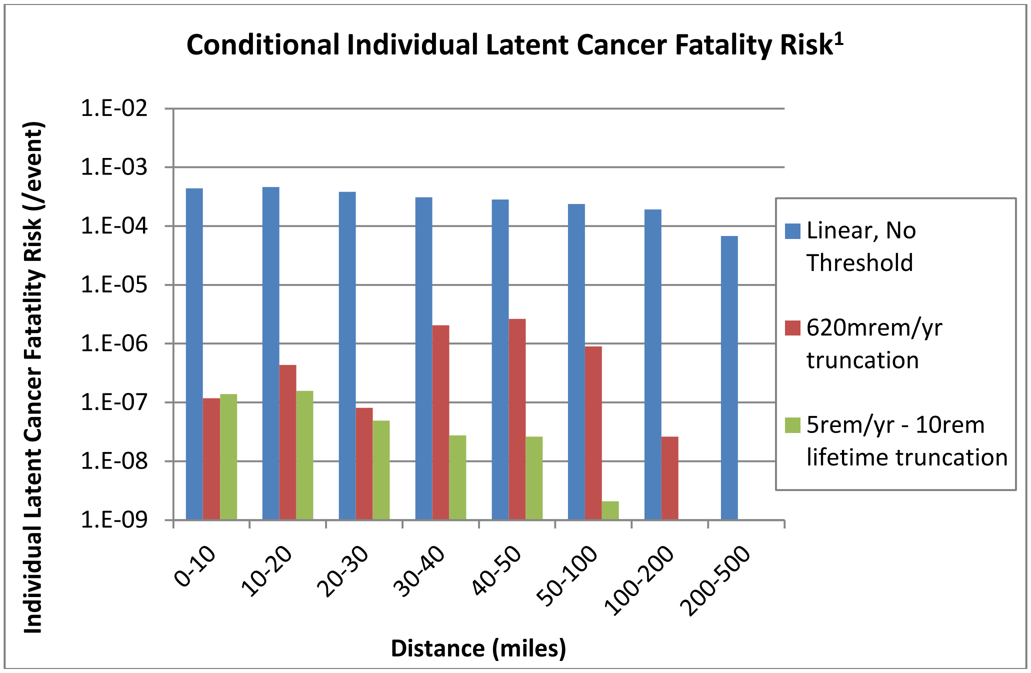
[1]Linear-no threshold, weather-averaged, release frequency-weighted, and population-weighted

Figure 96 Individual Latent Cancer Fatality Risk (per year)

The accuracy of atmospheric transport and deposition (ATD) models (e.g., the Gaussian plume segment model used in MACCS2) tend to decrease with distance, and therefore the results should be viewed with caution at longer distances. However, MACCS2 has been benchmarked against other ATD models, and the staff considers the broad conclusion remains valid—that risks would be small but drop slowly with distance in the event of large releases.

For affected areas with large populations, severe accidents can result in significant numbers of latent cancer fatalities. However, this should be weighed against the likelihood of the accident. Furthermore, because the individual doses are relatively small, this would be a small fraction of all cancer fatalities from all causes. This risk mainly comes from doses that are constrained to be less than dose limits for protective actions from an LNT dose response model; dose truncations predict significantly fewer latent cancer fatalities.

Figure 97 compares the quantified individual LCF risk for different dose truncations and for a variety of reported distance ranges for a high-density (1x4) configuration with unsuccessful mitigation for 3 days. The figure shows that dose truncation significantly lowers the quantified LCF risk. This is similar to Table 34; however this figure shows risks for a range of distances.



[1] High Density (1x4)—Unsuccessful Mitigation, weather-averaged, release frequency-averaged, and population-weighted

Figure 97 Conditional Individual LCF Risk for Different Dose Truncations and Distances

The effect of protective actions can be observed from Figure 97. For the release modeled in this scenario, the LCF risk within 10 miles is slightly less than at the 10–20 mile range. This small variation in risk with distance is because different modeled protective actions (such as evacuation, sheltering, early relocation, decontamination, temporary interdiction, and permanent condemnation) will depend on the level of contamination expected at a particular location. For example, the higher contamination levels closer to the source may result in relatively longer periods of relocation. Because no exposure to these populations would occur during this period these individuals could have lower overall doses than individuals further away under some situations. The 620 mrem annual dose truncation in particular demonstrates the effect of reduced individual LCF risk at these distances compared to longer distances. In Figure 97, the 620 mrem annual dose truncation best illustrates the effect of emergency response because this dose truncation does not quantify the significant contributions from chronic, long-term exposures.

7.2.3 Land Contamination

As the values in Table 33 suggest, conditional on a release (with a frequency of 1E-7 per year, or lower) occurring, the total land contamination area can be considerable. The low-frequency, large releases are significantly affected by hydrogen combustion events, which are currently predicted in some high-density loading situations without successful mitigation for 3 days, but not in other scenarios. For relatively small releases from a SFP, the extent of contaminated land could range to hundreds of square miles. For a large release, such as a release from a high-density pool without successful deployment of 50.54(hh)(2) mitigation that leads to a hydrogen combustion event, the amount of contaminated land can be two orders of magnitude higher (Table 35 partially reflects this range, although it reports average values). The levels of potential land contamination in the event of a release should be weighed against the likelihood of the accident. When the amount of contaminated land is weighted by the annual likelihood of occurrence (as seen in Table 33), the expected impact is relatively low. In addition, only a small portion of these interdicted areas are expected to be permanently interdicted, as the level of contamination is expected to significantly decrease with time as decontamination, radioactive decay, and weathering occur.

The amount of land affected depends on the dose criterion selected. For the purposes of this study, land contamination is defined as the area impacted by protective actions, specifically either temporary or permanent land interdiction. Because of the location of the reference plant, the particular protective action level the study uses is the Pennsylvania standard for habitability (dose limit of 500 mrem each year). The study uses this measure to estimate land contamination starting in the first year after a potential severe accident. In reality, the annual dose limit for what is considered “habitable” can change when crossing a state boundary. However, for consistency and practicality reasons, the same standard for estimating land contamination area is applied to all affected areas in this study, and the measure chosen for this study is only meant to be an indicator of land contamination.

Consistent with the observations of a relatively slow decline in individual latent cancer fatality risk with distance, the results of the analysis indicate that protective actions such as temporary relocation may be needed at long distances. The table below displays an average amount of interdicted land within different distances for high- (1x4) and low-density fuel loading.

Table 35 Average Land Interdiction* (square miles per event)

SFP Loading Pattern	High Density (1x4)		Low Density	
10 CFR 50.54(hh)(2) mitigation credited	Yes	No	Yes	No
Release Frequency (/yr)	6.1E-9	1.2E-7	6.1E-9	1.2E-7
0-50 miles	210	1,200	210**	160
0-100 miles	230	3,100	230**	170
0-500 miles	230	9,400	230**	170

* Weather-averaged and release frequency-averaged; Dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51
** Mitigation can moderately increase release size (see Section 6.3); the effect is small compared to the reduction in release frequency.

A release in the high-density fuel loading situation without successful 50.54(hh)(2) mitigation is capable of large releases, and therefore an average release from this situation is capable of causing significantly more land contamination at longer distances than in the other situations. In contrast, releases from situations with low density fuel loading (and/or successfully deployed 50.54(hh)(2) mitigation equipment) cause a relatively smaller amount of land contamination beyond 50 miles, and none beyond 100 miles when using land interdiction as a measurement of land contamination. This is because on average, a release in these situations contaminates significantly less area. However, because of the release magnitude of any of the analyzed SFP releases, the total amount of land contamination that remains within ten miles is relatively small.

On land contamination, past results are expected to be broadly consistent with this study. However some previous studies did not report land contamination and some reported different metrics for estimating areas, so a direct comparison is not possible. NUREG/CR-6451 reports values for condemned farmland that includes hundreds of square miles within a 50-mile radius and thousands of square miles within a 500 mile radius, albeit for a full core off-load. NUREG-

1353 reports values for land contamination based on NUREG/CR-4982 that range into the hundreds of square miles, albeit largely within a 50-mile radius of the plant. These differences, as well as different choices for the land contamination criteria that can significantly affect the estimated areas, make a quantitative comparison less meaningful. However, it is clear that both this study and past studies have predicted that SFP accidents can lead to significant land contamination.

7.2.4 Displaced Individuals

Consistent with the results for land contamination, relatively large numbers of people may be impacted following a large release from a spent fuel pool. Displaced individuals, also known as relocated individuals, are people who are predicted to be temporarily or permanently relocated due to interdiction of contaminated land, based on the dose limit for land interdiction starting in the first year following an accident. These individuals are not necessarily the same as evacuees, who evacuate during the emergency phase (although an individual could be both of these).

Conditional on a release (with a frequency of 1E-7 per year or lower) occurring, the total number of temporarily relocated individuals could be considerable. For relatively small releases of an SFP, the number of displaced individuals could range into the hundreds of thousands. For a large release, which is predicted in some high-density loading situations early in the operating cycle without successful 50.54(hh)(2) mitigation, the number of displaced individuals can be two orders of magnitude higher. (Table 36 partially reflects this range, although it reports average values).

Also consistent with the observations related to the amount of land contamination with distance, the results of the analysis indicate that protective actions such as temporary relocation may be needed at long distances. The table below displays the average number of displaced individuals for different distances for high (1x4) and low density fuel loading.

Table 36 Average Number of Long-term Displaced Individuals* (per event)

SFP Loading Pattern	High Density (1x4)		Low Density	
10 CFR 50.54(hh)(2) mitigation credited	Yes	No	Yes	No
Release Frequency (/yr)	6.1E-09	1.2E-07	6.1E-09	1.2E-07
0-50 miles	100k	780k	100k**	72k
0-100 miles	120k	2,000k	120k**	81k
0-500 miles	120k	4,100k	120k**	81k

* Weather-averaged and release frequency-averaged; Dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51
** Mitigation can moderately increase release size (see Section 6.3); the effect is small compared to the reduction in release frequency.

These estimates should be weighed against the likelihood of the accident. When the number of displaced individuals is weighted by the annual likelihood of occurrence (with a frequency of 1E-7 per year or lower; as seen in Table 33), the expected impact is relatively low. An average release in the high-density fuel loading situation without successful 50.54(hh)(2) mitigation causes significantly more relocation at longer distances than in the other situations because it is capable of larger releases. In contrast, releases from situations with low density fuel loading (and/or successfully deployed 50.54(hh)(2) mitigation equipment) cause a relatively small amount of relocation beyond 50 miles, and none beyond 100 miles because on average, a release from these scenarios contaminates significantly less area. For all situations, the number of displaced persons from the 0 to 10 mile area is relatively small because the number of people living on this area is relatively small.

7.3 Offsite Consequence Comparison

A goal of the study is to compare the results of the scenario-specific, high- and low- density fuel loading seismic events. To facilitate the comparison, results of different scenarios are compared to each other by dividing the results from one scenario by another scenario, for a variety of consequence metrics. The ratios of the consequence metrics are indicators of the scenario specific safety benefit between the two scenarios.

These comparisons should consider the scenario release frequency as well as conditional on a release occurring, appropriate. In the first comparison below, the high-density (1x4) fuel loading and low-density fuel loading had the same release frequency. Therefore, for this comparison, there is no additional reduction when the likelihood of occurrence is also considered.

Table 37 Consequence[1] Comparison – High (1x4) Density / Low Density Loading without Successful 50.54(hh)(2) Mitigation

SFP Fuel Loading	High Density (1x4)	Low Density	Reduction Factor (dimensionless)
Release Frequency	1.2E-07	1.2E-07	1.0
Individual Latent Cancer Fatality Risk[2] within 10 Miles	4.4E-04	2.0E-04	2.1
Collective Dose (Person-Sv)	350k	27k	13
Land Interdiction[3] (mi^2)	9,400	170	56
Long-term Displaced Individuals[3] (Persons)	4,100k	81k	51

[1] Conditional on a release occurring (frequency of 1E-7 per year, or lower); results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions
[2] Linear-no threshold, population-weighted
[3] 1st year post-accident; calculation uses a dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51

The most significant reduction factor in a low-density loading pattern is in the reduction in land interdiction and associated displaced individuals. This is because these consequences are more closely related to the size of release than the other results. In addition, a small amount of contamination can occur before land reaches the dose level for interdiction. This dose threshold effect means smaller releases more-than-proportionally reduce the amount of land interdiction.

The reduction in collective dose (and latent cancer fatalities) in a low density loading pattern is also due to the smaller release magnitude. This reduction is significant, although not as significant as the reduction in land interdiction. This is because larger releases are predicted to have considerably more temporary and permanent interdiction to protect the public. This is especially true at shorter distances, as indicated by the reduction factor for LCF risk for 0-10 miles. One significant reason a smaller release magnitude is expected in the low-density loading situations is because hydrogen combustions are currently not predicted in these situations.

The next table reports the reduction of the consequences with successful deployment of 50.54(hh)(2) mitigation equipment. Because successfully deployed mitigation can prevent fuel release, it affects the reduction factors for release frequency-weighted consequences (per year) differently than consequences conditional on a release occurring. For brevity, the consequence values are not displayed here, although can be seen in the previous section.

Table 38 Consequence Comparison – Unsuccessful/Successful Deployment of 50.54(hh)(2) Equipment

Fuel Loading Density	High Density (1x4)	Low Density
	Reduction Factor (dimensionless)	
Change in Release Frequency (/yr)	19	19
	Conditional[1] Consequences (Release Frequency-Averaged[2])	
Type of Consequence	Reduction Factor (dimensionless)	
Individual Latent Cancer Fatality Risk[3] within 10 Miles	1.3	0.61
Collective Dose (Person-Sv)	7.4	0.59
Land Interdiction[4] (mi^2)	40	0.72
Long-term Displaced Individuals[4] (Persons)	36	0.70
	Consequences per year (Release Frequency-Weighted[2])	
Type of Consequence	Reduction Factor (dimensionless)	
Individual Latent Cancer Fatality Risk[3] within 10 Miles (/yr)	25	12
Collective Dose (Person-Sv/yr)	140	11
Land Interdiction[4] (mi^2/yr)	780	14
Long-term Displaced Individuals[4] (Persons/yr)	690	13

[1] Conditional on a release occurring (frequency of 1E-7 per year, or lower)
[2] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions; additionally, "release frequency-weighted" results are multiplied by the release frequency.
[3] Linear-no threshold, population-weighted
[4] 1st year post-accident; calculation uses a dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51

For both high- and low-density fuel loading, the release frequency was reduced by about a factor of 20 with successful deployment of 50.54(hh)(2) mitigation.

Conditional on a release occurring (middle portion of Table 38), successful deployment of 50.54(hh)(2) mitigation reduces all of the average consequences of the high-density fuel loading pattern, although to varying degrees. These varying degrees of consequence reductions are similar to that predicted in Table 37 for using a low-density loading pattern, although to a somewhat lesser extent. A significant portion of this reduction may be attributable to the fact that hydrogen combustions are not predicted with successful deployment of 50.54(hh)(2) equipment.

Contrary to what might be expected, 50.54(hh)(2) mitigation is predicted to slightly increase the average conditional consequences of a release from a low-density fuel loading pattern. While successful deployment of 50.54(hh)(2) equipment is usually effective at preventing releases, it is not as effective at mitigating release from the low-density fuel loading pattern when deployed in a capacity specifically to provide makeup water through injection, as sometimes assumed. In these conditions, release from a SFP can sometimes be somewhat larger with deployed mitigation. In addition, the situations for which 50.54(hh)(2) equipment prevented release for the low-density loading events were the situations with the smallest release magnitudes, which has the non-intuitive effect of increasing the average consequence of a release.

The bottom section of Table 38 shows the combined benefit of prevention and mitigation from successfully deployed 50.54(hh)(2) equipment, which combines the reduction factors of a lower release frequency with the changes in the average consequences of a release. In the high-density loading situation, the overall benefit of 50.54(hh)(2) equipment is very significant (more than a factor of 100 reduction in most of the risk metrics) if successfully deployed. For low-density loading, the deployment of the 50.54(hh)(2) equipment has a somewhat negative effect on the average conditional consequence; however, this is far outweighed by the benefit it provides in preventing release.

8. HUMAN RELIABILITY ANALYSIS

Consistent with the limited scope of the SFPS, a limited scope human reliability analysis (HRA) was performed, to develop initial insights into the likelihood of successful operator actions to prevent spent fuel damage for the specific seismic event and consequence scenarios studied. A full scope HRA would primarily be useful as part of a PRA analysis. A PRA would necessarily consider a much broader scope than the SFPS. Such a scope would include the likelihoods of all initiating events, the plant damage states for the two reactors and spent fuel pools, the availability of all installed or portable mitigation equipment, and the availability of on-site (and possibly off-site) personnel. Thus the limited scope HRA results presented here must be viewed from the context of its specific assumptions, including assumptions that remove likely complexities that impact operator performance.

In this context, to perform an HRA for this study, successful mitigation must be defined. For this HRA, mitigation success is defined as preventing radioactive release from the fuel rods of the Unit 3 SFP fuel (or gap release). The reference plant site has two reactor units (Unit 2 and Unit 3) in operation. The status of the Unit 2 and 3 reactors, the Unit 2 SFP, and the other plant SSCs would affect Unit 3 SFP mitigation, but successful mitigation, as defined in this analysis, is only determined by the Unit 3 SFP fuel status.

The effective SFP mitigation strategies, to prevent fuel overheating and release of radioactive material from the damaged fuel rods, are to either inject or spray water into the SFP from the refueling floor. The refueling floor on top of the reactor cavity is part of the primary containment that insulates the refueling floor from the reactor. In situations involving reactor damage with intact primary containment, access to the refueling floor is still possible. Over the refueling floor is the secondary containment which is a light-weight steel structure. During an SFP accident, the secondary containment can reduce the radioactivity released from the SFP to the environment. During refueling, the primary containment head is removed to expose the reactor cavity. The reactor vessel head is also removed for defueling and refueling. Therefore, during refueling outage, the refueling floor is no longer insulated from the reactor. Heat and radiation generated from the reactor would directly affect the work environment on the refueling floor. In addition to the strategies of spraying water from the refueling floor to the SFP, strategies to spray water from outside of the secondary containment (e.g., by ladder fire trucks) to the secondary containment or the SFP (through containment breaches) are available. However, as these strategies are aimed at mitigating releases to outside of the secondary containment and not at preventing fuel overheating, they are not credited in this HRA study.

The SFPS ran a number of computer simulations to understand the effects of a set of factors affecting SFP fuel radioactive release after an earthquake damaged the normal SFP cooling system. The set of factors include SFP leak size, spent fuel loading pattern, OCP, mitigation deployment, mitigation flow rate, and types of mitigation (i.e., injection and spray). These simulations generated information that served as the foundation for the HRA study. Section 8.1 summarizes the SFPS results relevant to the HRA study and discusses their implications to the HRA study. Section 8.2 discusses the staffing, mitigation equipment, strategies, and procedures of the reference plant relevant to the SFP mitigation. Section 8.3 discusses the HRA study framework, scope, and approach. The conduct of an HRA is normally done in conjunction with a PRA to identify each event sequence (i.e., scenario) following an initiating event. For each event sequence, the PRA model would explicitly specify the status (i.e., success or failure) of each component, system, and human action that affects the event sequence's progression and end consequence. For this reason, the development of a PRA model would require significant effort. For this limited scope HRA study, a detailed PRA (i.e., using event trees branched to represent various possible scenarios) was not performed. Instead, scenarios are classified based on the status of a few key SSCs (i.e., electric power availability, and the status of the Unit 3 reactor and primary containment). The Unit 3 reactor and primary containment status are included in the HRA study because of their significant effects on the Unit 3 refueling floor work environment (i.e., where the SFP mitigation strategies are performed). Table 39 summarizes the scope and assumptions applied to the HRA study. Section 8.4 summarizes the insights of this study.

Table 39 The scope and assumptions of the HRA study

#	Scope and assumptions	Notes
1	Success criterion: prevent radioactive release from the fuel rods of Unit 3 SFP fuel	- Do not include strategies designed to reduce radioactivity released to the environment. The effective mitigation is to inject or spray water into the SFP. - The status of fuel in the Unit 2 and 3 reactors and Unit 2 SFP are not considered in the success criterion.
2	Classify plant damage states as a result of the earthquake and estimate the mitigation failure probability for each plant state.	The probabilities of the plant damage states as a result of the earthquake were not estimated.
3	The installed equipment for SFP mitigation is not available. Operators have to use the 10 CFR 50.54(hh)(2) equipment for mitigation.	If the installed equipment (e.g., fire system and residual heat removal system) is available, the SFP mitigation would have a much higher success likelihood than this study's estimates.
4	The SFP mitigation uses the minimum flow rate specified in NEI 06-12 guidance for complying with 10 CFR 50.54(hh)(2).	The actual flow rate is expected to be higher than the minimum NEI recommended flow rate (i.e., 500 gpm of injection or 200 gpm of spray)
5	10 CFR 50.54(hh)(2) equipment and water sources for Unit 3 SFP mitigation is available.	Earthquake-caused damage to the 10 CFR 50.54(hh)(2) equipment is not included in the study. Further, dividing equipment to mitigate multiple reactor and SFP problems is not considered.
6	Sufficient plant staff is available to perform the Unit 3 SFP mitigation.	Staffing information is discussed but different staffing scenarios are not factored into the analysis. For multiple reactors and SFPs damaged by the hypothetical earthquake, the personnel sufficiency would be a key factor affecting mitigation success.
7	Non-plant, off-site support (e.g., off site fire trucks) are not considered.	For an SFP event, the primary function of off-site support is to keep radioactivity release within the plant site. Off-site support for preventing SFP fuel rod damage is not credited.

8.1 Summary of Spent Fuel Pool Study Analysis Results Relevant to Human Reliability Analysis

8.1.1 High Level Scenarios Classification

The SFPS concludes that the following four scenarios do not lead to gap release with a 72-hour-truncated simulation time (see Table 40 for a tabulate classification):

(1) Boil off Scenario with No SFP Leaks. As mentioned earlier, the SFP water level in this scenario would take more than 7 days to decrease to the top of the fuel rack. Because of the long time available for response, multiple opportunities are available to prevent damage to the SFP; therefore, the human error probability (HEP) (in this study the HEP is equivalent to mitigation failure probability) is negligible.

(2) Mitigated Scenario for Small Leaks. No fuel damage occurred when the makeup water was injected into the SFP at the time specified by the SFPS. The SFPS suggests that, as long as the spent fuel is covered with water, SFP failure would not occur. Therefore, the available time for operators to respond is longer than the SFPS injection time in some scenarios. For these scenarios, the HEPs were calculated.

(3) Unmitigated Scenario in Late Phases (i.e., OCPs 4 and 5). These scenarios have low decay heat. Even when relying only on natural air circulation, heat convection and radiation, and other natural means of heat transfer, overheating of the spent fuel can be prevented. For these scenarios, the HEPs were not calculated.

(4) Mitigated Moderate Leak Scenarios in OCP2, OCP3, OCP4, and OCP5. When SFP water is drained to the top of fuel rack, the radiation level is considered too high to deploy SFP mitigation strategies on the refuel floor (discussed later). In OCP2, the SFP water takes almost 6 hours to drain to the top of the fuel rack but only about 2.5 hours in OCP3. The HEPs were calculated for OCPs 2 and 3. The HEPs for OCPs 4 and 5 were not calculated because SFP decay heat is insufficient to cause fuel damage, as noted above.

The SFPS shows that the SFP status is stable in the four scenario classes listed above following termination of the computer simulations at 72 hours after the initiating event. This result implies that, for the unmitigated scenarios, if spent fuel damage does not occur within the first 72 hours, spent fuel damage would not occur afterward.

Table 40 The SFPS Simulation Results.

	No Leak (90%)	Small Leak (5%)	Moderate Leak (5%)
OCP 1 (0.9%)			~ 0.05%
OCP 2 (2.4%)		~0.8%	
OCP 3 (5.0%)			
OCP 4 (25.7%)			
OCP 5 (66%)		~ 99.2%	

- OCP: Operating Cycle Phase
- Percentages above are percent of the time for corresponding condition.

Table 40 provides an overview for performing an HRA as follows:

- The green cells represent that either the HEP is negligible or mitigation does not affect the end consequence. For the SFP no leakage scenario, the SFPS calculated that SFP water would take more than 7 days to boil to the top of the fuel rack. Because of the long time available for mitigation, the HEP is negligible. For the scenario in which the earthquake occurs during OCPs 4 and 5, the SFP fuel decay heat is insufficient to cause a gap release event even without the provision of SFP makeup flow; therefore, mitigation does not affect the end consequence. These two scenario classes (i.e., no leakage and the occurrence of the earthquake during OCPs 4 and 5) are colored as green cells and total about 99.2 percent of the conditional probability. An HRA is not performed for the scenarios in the green cells.

- The OCP 1 moderate leakage scenario (i.e., the red cell with a ~0,05% conditional probability in Table 40) would result in a gap release regardless of whether mitigation has taken place because the current NEI guidance for complying with 10 CFR 50.54(hh)(2) is insufficient (providing at least 500 gpm of injection flow or 200 gpm of spray flow[40]). The flow rates are provided by two flow paths using fire hoses. Significantly increasing the mitigation flow rate requires setting up additional fire hoses to provide additional flow paths. Because the procedures do not provide instructions on when additional flow paths should be established, this study concludes that no additional flow path other than the two procedure-instructed flow paths will be used for SFP mitigation. Therefore, gap release would occur in the OCP1 moderate leak scenarios. This is not because the mitigation flow cannot be deployed in time, but is because the flow rate is insufficient for the assumed OCP 1 decay heat load as determined by SFPS section 6.3.2.[41]

- The yellow colored cells represent conditions where gap release can be prevented if the minimum NEI recommended SFP makeup flow (i.e., 500 gpm of injection or 200 gpm of spray) is deployed in time. This HRA focuses on these scenarios for which mitigation would prevent gap release.

8.1.2 Key Factors Affecting Available Time for Mitigation

The SFPS divides the reference plant operation cycle into five OCPs. OCPs 1 and 2 occur during refueling in which the SFP and reactor cavity are hydraulically connected. Because the reactor cavity and SFP are located within the same reactor building and they are hydraulically connected, a reactor problem would affect the refueling floor work environment in which the effective mitigative actions to prevent SFP fuel damage are performed. OCPs 3, 4, and 5 occur during at-power operations in which the SFP and reactor cavity are hydraulically disconnected. This HRA assumes different rates of spent fuel decay heat for each OCP, which in turn affects

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40 NEI 06-12, “B.5.b Phase 2 and 3 Submittal Guideline,” issued December 2006 (ADAMS Accession Nos. ML070090060 and ML070080351) recommends minimum of 500 gpm of injection and 200 gpm of spray for implementation of the requirements in 10 CFR 50.54(hh).

41 In comparison with OCP 1, moderate leakage, and mitigated scenarios, the OCP 2 scenario has the same makeup type (i.e., injection), makeup flow rate, and makeup deployment time. However, gap release did not occur in the OCP 2 scenario because the hottest 88 assemblies for OCP 1 at approximately 4 days have a decay heat of 1,927 kilowatts (kW) or 65 percent of the whole SFP (2,951 kW), whereas the hottest assemblies for OCP 2 at 13 days have a decay heat of 1,143 kW or 32 percent of the whole SFP (3,567 kW).

the required mitigation flow and, to some degree, the available time for mitigation necessary to prevent SFP damage.

The SFPS groups the SFP damage caused by the earthquake into three classes: (1) no leakage, (2) small leakage, and (3) moderate leakage with a corresponding conditional probability of 90 percent, 5 percent, and 5 percent, respectively. The small leakage scenario is represented by 40 small tears in the stainless steel liner at the backup bar locations. The small cracks create an initial leakage rate of about 250 gpm. The leakage flow rate depends on the SFP water level. As the SFP water level decreases, the leakage rate reduces. The moderate leakage is represented by a long crack with a combination of the stainless steel SFP liner tear and a through-wall concrete crack at the bottom of the SFP wall. Section 4.1.5 of this report discusses the SFPS damage states in detail. The moderate leak creates an initial leakage rate of about 1,900 gpm.

The HRA assumes that the SFP leak rate affects the available time necessary for mitigation because, when the SFP fuel is not covered by water, the radiation level at the locations in which mitigative equipment is stored and mitigative actions are performed is assumed to be too high for performance of the mitigative actions in this study. Thus, the SFP leak rate directly affects the SFP fuel uncovery time. Table 41 shows the time to SFP fuel uncovery in the various scenarios.

Table 41 Approximate Time of Fuel Uncovery

Time	No Leak	Small Leak	Moderate Leak
OCPs 1 and 2	> 7 days	40 hours	6 hours
OCPs 3, 4, and 5	> 7 days	19 hours	2.5 hours

Figure 98 shows the approximate dose rate contours in the refueling area at the time of defueling when the SFP water level is at the top of the fuel rack. The radiation at the mitigation equipment storage location ranges from 3–30 rem per hour and the radiation level at the locations of the spray nozzles for SFP makeup is in the range of 10 to 300 rem per hour. Working at this radiation level could cause emergency responders who perform mitigation actions to receive doses greater than those in EPA's PAGs (EPA, 1992). This radiation map is the basis for specifying that the SFP makeup must be deployed before the SFP water level reaches the top of the fuel rack in order to credit mitigation success.

In addition to radiation, high temperature on the refueling floor is another factor that affects mitigation success. In this study, 140 °F (60 °C) is used as the temperature threshold. The refueling floor reaches 140 °F before the SFP water level is drained to the top of fuel rack only in the OCP 1 and 2 small leak scenarios. In these scenarios, the reactor head is open. Boiling in the reactor cavity significantly increases the temperature on the refueling floor. Figure 99 shows the time history of the refueling floor temperature of the OCP 1 small leak scenarios. The temperature reaches 140 °F in about 13.5 hours. Figure 100 shows the time history of the refueling floor temperature of the OCP 2 small leak scenarios. The temperature reaches 140 °F in about 26 hours. Because of the long available response time and steep temperature increase at the time of 140 °F reached, changing the temperature threshold to a higher temperature does not affect the HRA results.

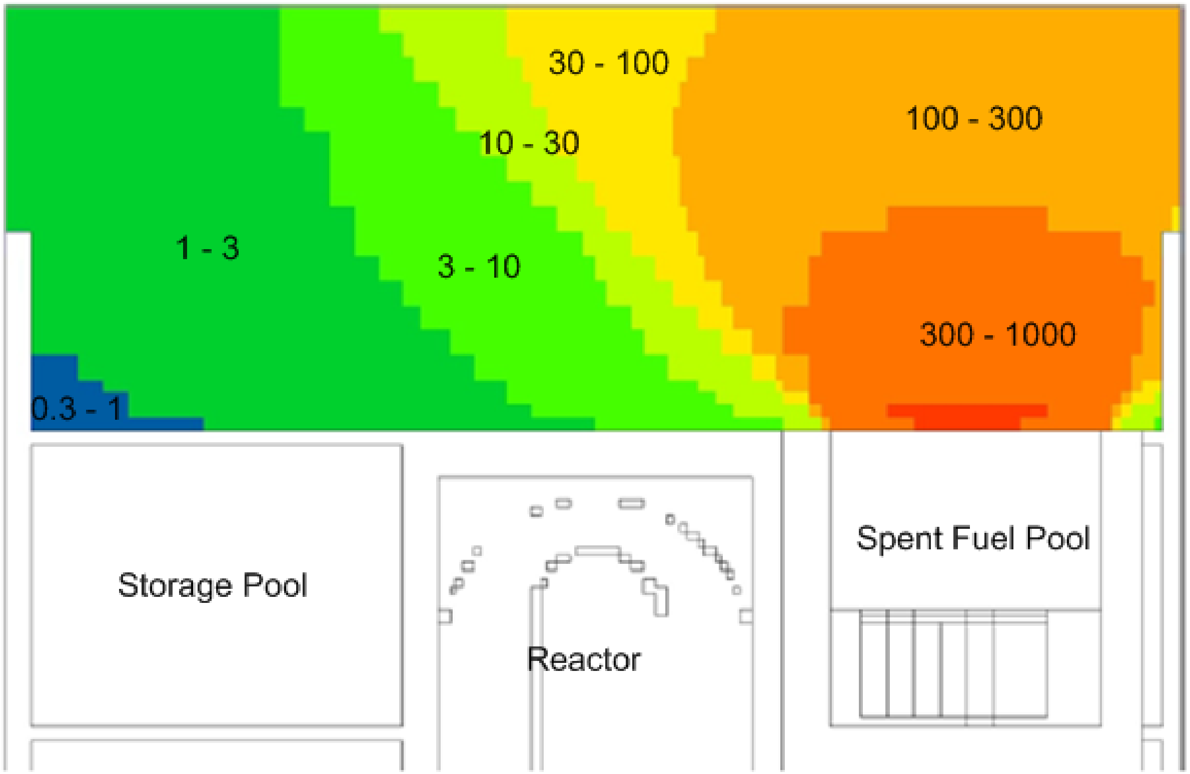


Figure 98 Approximate dose rate of elevation contours, water at the top of fuel hardware, around the time of defueling (rem per hour).

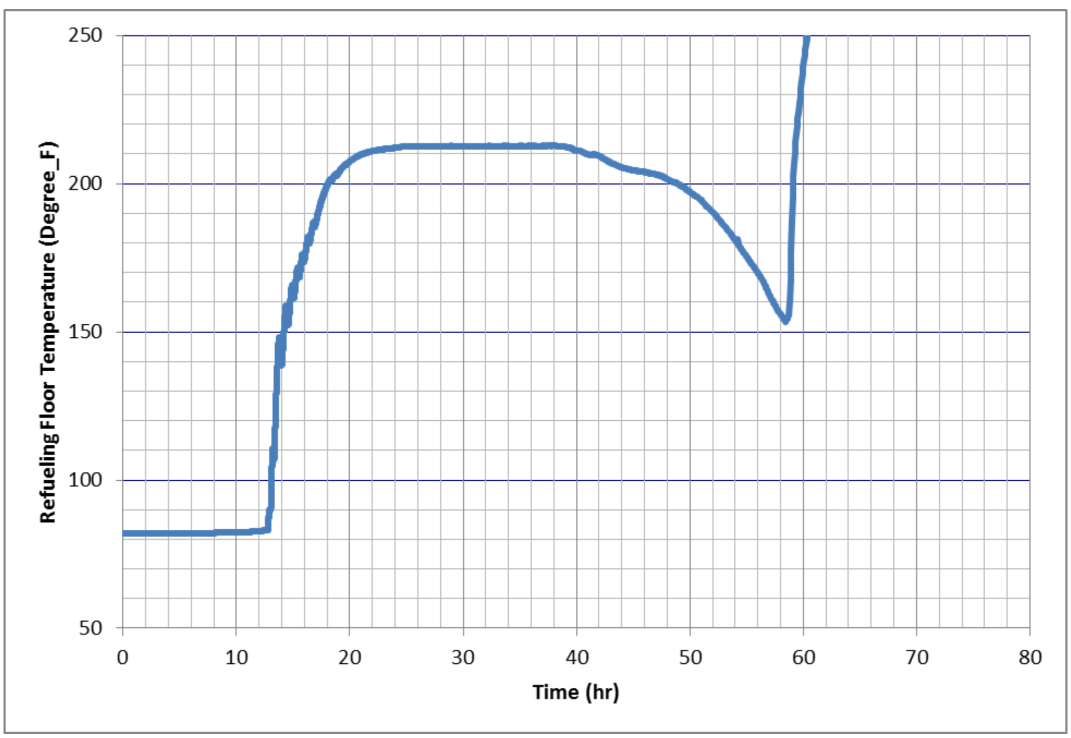


Figure 99 The refueling floor temperature of OCP1 small leak scenarios.

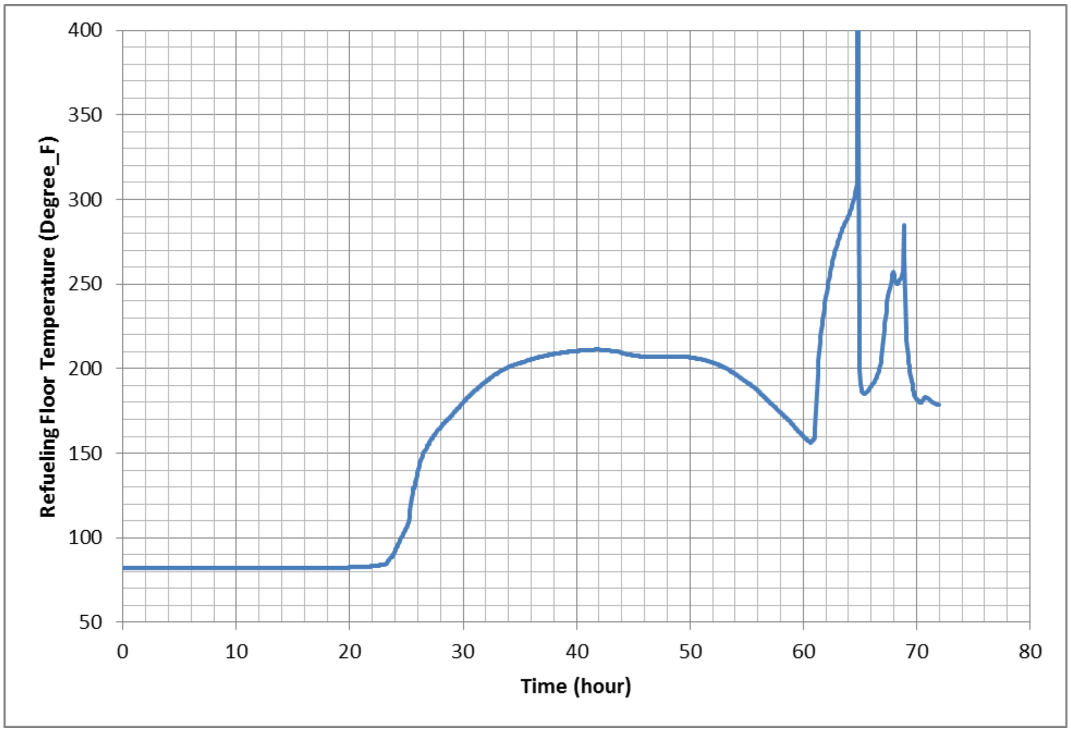


Figure 100 The refueling floor temperature of OCP 2 small leak scenarios.

In summary, successful deployment of the mitigation strategy has to be done before the earliest of either the SFP water reaching the top of the fuel rack or the reactor building atmosphere reaching 140 °F. Table 42 shows these available times for the scenarios of interest.

Table 42 The available time*

	Small Leak (hr)	Moderate Leak (hr)
OCP 1	13.5**	6***
OCP 2	26**	6***
OCP 3	19***	2.5***

*Assume Unit 3 reactor is not damaged
**Due to refueling floor temperature reaching 140°F
***Due to SFP water level draining to the top of fuel rack

8.2 Staffing, Mitigation Equipment, Strategies, and Procedures

8.2.1 Staffing, Procedures, Training, and Response Time

Staffing

This HRA assumes that sufficient plant staff is available for Unit 3 SFP mitigation. In the situation that the hypothetical earthquake causes damage to multiple SSCs, additional events (e.g., fire), and personnel injury, the assumption may not be applicable to some scenarios.

The reference plant uses a combined main control room for its two reactor units. Consistent with NEI 12-01, the on-shift personnel are assumed to be limited to the minimum complement allowed by the site emergency plan. This represents a staffing level during backshift, weekend or holiday. The staffing level of the reference plant, Units 2 and 3, during the backshift, weekend, and holiday includes the following:

- Main Control Room
  - One Shift Manager (Licensed Senior Reactor Operator (SRO)). The shift manager oversees the control room activities and assesses the emergency action level.
  - One Shift Technical Advisor (Licensed SRO). The shift technical advisor performs independent plant status assessment.
  - Two Control Room Supervisors (Licensed SROs). The control room supervisors implement procedures as a team with the reactor operators (ROs).
  - Two Licensed ROs. The ROs perform control board actions according to the control room supervisors' instructions and answer emergency phone calls.
  - Two Assistant (or Spare) Licensed ROs. The assistant operators perform the same functions as the ROs.
- On Site
  - One Field Supervisor (Licensed SRO). The field supervisor oversees onsite activities.
  - Nine Auxiliary Operators. The auxiliary operators will report to the main control room after the earthquake to obtain the master keys for the assigned tasks. Five of the nine auxiliary operators are on the fire brigade.
  - Additional Staff. Additional staffing comprises health physicists, chemical staff, maintenance personnel, and security staff onsite who can support mitigation (e.g., health physicists will provide refueling floor radiation information). However, these people are not expected to directly perform SFP mitigative actions.

The above summary describes a typical staffing level during backshift, weekend or holiday of the reference plant instead of the minimum staffing requirement, or during a normal weekday or refueling. If the earthquake occurs during normal working hours or if either Unit 2 or 3 is in a refueling outage, the staffing level would be significantly higher.

To augment staffing, except calling for the off-site plant staff (e.g., mobilize emergency response facilities), the reference plant can also call for the nearby Delta-Cardiff Volunteer Fire Company to assist in tasks such as SFP mitigation, fire mitigation, and treatment of injured personnel. The fire company could send engines, tankers, a ladder fire truck, an air unit, an ambulance and personnel to the reference plant site. Based upon the above assumptions, this analysis assumes that there is sufficient staff for Unit 3 SFP mitigation. No detailed analysis is performed on the staffing situation for all scenarios.

Procedures and Operator Initial Responses

In the hypothetical earthquake that causes a station blackout (SBO), the general response is that the control room supervisors work with the ROs to implement the emergency response procedures. In this case, the entry conditions of the following three procedures are met:

(1) SE-11, "Loss of Offsite Power"
(2) SE-5, "Earthquake"
(3) TSG-4.1, "Operational Contingency Guideline"

The control room supervisors work with ROs to implement the above three procedures in parallel. The immediate objectives are to ensure that the reactor is properly tripped and ensure sufficient electricity, equipment, and water to maintain reactor cooling. Because a high-priority task in an SBO scenario is the provision of emergency electric power, the control room supervisors would send two auxiliary operators to inspect the emergency diesel generators and would direct one assistant (or auxiliary) operator to implement SE-11 to connect the dedicated power supply from the Conowingo Hydroelectric Generating Station (Conowingo) to the reference plant. If the earthquake has not affected Conowingo, connecting its supply power to the reference plant would take about 1 hour during normal conditions. The other auxiliary operators will be tasked with performing a plant walkdown and SFP inspection in accordance with SE-5.

Training

Training related to the implementation of TSG-4.1 and 10 CFR 50.54(hh)(2) includes the following:

- annual training in emergency response organization mobilization and implementation of the TSG-4.1 and TSG-4.2, "Extreme Damage Mitigation Guidelines for Loss of Large Area of the Plant," procedures and the related requirements in 10 CFR 50.54(hh)(2)

- biannual training on security threat responses

- initial training on procedures and equipment related to 10 CFR 50.54(hh)(2)

Response Time

NEI 06-12, Revision 2, "B.5.b Phase 2 & 3 Submittal Guidance," states that plants should be able to deploy a flexible means of providing SFP makeup (i.e., either 500 gpm of injection or 200 gpm of spray per unit) within 2 hours from the time in which plant personnel diagnose that external SFP makeup is required. This HRA study uses the 2-hour deployment time as the action time for deploying mitigation. The total mitigation time is the sum of delay time, diagnosis time, and action time (discussed in Section 8.3.2.2).

The analysis in Volume 1 of NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project," estimates that, given the hypothetical earthquake event that causes SBO but with dc power, the technical support center (TSC) is assumed operational within 2.25 hours after the earthquake. The NEI 12-01 guideline assumes the following site accessibility: (1) no site access within the first 6 hours; (2) limited site access between 6 to 24 hours; and (3) improved site access after 24 hours. The assumptions apply to a large-scale external event that occurs that results in: (1) all on-site units affected; (2) extended loss of AC power, and (3) impeded access to the reactor buildings. The emergency response facilities most relevant to responding to the hypothetical earthquake are the operational support center (OSC), which is an onsite assembly area separate from the control room, and the TSC to which licensee operations support personnel report in an emergency. NUREG/CR-7110 does not provide an estimated time in which the OSC will be operational. Therefore, for the purposes of this study the TSC assumption of 2.25-hours is also used for the OSC when neither Unit 2 nor Unit 3 is in a refueling outage. The OSC provides additional man power to mitigate plant damage, but this additional staff is not considered in this HRA study.

8.2.2 Mitigation Equipment

This HRA study assumes that portable mitigation equipment is available but the installed equipment is not available for Unit 3 SFP mitigation. The portable equipment includes the two portable diesel pumps discussed in this section. The installed equipment includes the fire system and residual heat removal system. Under the hypothetical earthquake, equipment may be damaged. If the earthquake causes damage to multiple reactors and SFPs that consequently requires mitigation equipment, the assumption of having sufficient mitigation equipment for Unit 3 SFP mitigation may not be a valid assumption. For the purposes of this study, portable mitigation equipment was assumed to be available.

The reference plant relies on the following onsite equipment and systems for SFP makeup:

- Fire System: One motor-driven fire pump and one diesel-driven fire pump are necessary to pressurize the fire header. The diesel-driven fire pump is designed to operate for 6.4 hours at full load. Making up the diesel fuel requires the use of a temporary 120-voltage ac power source to restore a fuel oil transfer pump to deliver fuel for the diesel-driven fire pump. In situations in which both fire pumps are lost and cannot be repaired within 1 hour, the reference plant will contact the York County 911 center for a fire engine to pressurize the onsite fire header. If the reference plant cannot obtain offsite support and if the situation allows, the reference plant can use one of the two portable diesel pumps to pressurize the fire header.

  The two fire pumps are housed in a seismic Class I tornado-resistant structure. Therefore, the diesel-driven fire pump is assumed functional after the earthquake. However, the underground fire pipes may be damaged by the earthquake. Depending on the damage, the fire system may still be available by isolating the damaged section or sections or by using a fire hose in place of the fire main. The fire system is the preferred water source for the most effective mitigation strategies necessary to prevent spent fuel gap release. If the fire system is not available, the Conowingo pond or torus storage tank is the alternative water source.

- Diesel-Driven Portable (DDP) Pump. The diesel-driven portable pump has the capability of delivering 600 gpm of water. A trailer stationed near the pump stores all piping, connectors, and spray nozzle. A dedicated pickup truck will be used to tow the pump and trailer to the specified location for operation. With a full tank, which is the normal condition, the pump can continue to run for more than 12 hours. The DDP pump has a 4” discharge connection. To deliver the flow rate of 500 gpm of injection or 250 gpm of spray the plant staff uses a wye adapter to connect the 4” discharge to two 2.5” hoses. The reference plant demonstrated that the combination flow rate met the 500 gpm of injection and 200 gpm of spray requirements.

- Diesel-Driven High-Capacity Portable (DDHCP) Pump. The diesel-driven high-capacity portable pump has the capability of delivering 1,300 gpm of water. The DDHCP pump has two 4” discharge connections. To deliver the NEI recommended flow rate, the plant staff uses a wye adapter to connect a 4” discharge to two 2.5” hoses. Four 2.5” discharging hoses would be needed to reach the pump maximum discharge capacity. The TSG-4.1 instructs the plant staff to connect two 2.5” hoses to a 4” discharge connection for SFP makeup. The reference plant demonstrated that the combination flow rate of using two hoses exceeds the 500 gpm of injection and 200 gpm of spray requirements. Like the diesel-driven portable pump, dedicated pickup trucks will be used to tow the portable pump to the designated location for operation.

The HRA team identified the three systems listed above during a site visit to the reference plant in July 2012. The HRA team was aware that PBAPS planned to purchase more equipment to address Order EA-12-49 mitigating strategies; however, this HRA study does not credit the additional equipment.

The reference plant stores much of its mitigation equipment at grade level. Section 2.4.3.5 of PBAPS’ FSAR discusses the effect of a simultaneous failure of the upstream Holtwood dam on the site. The FSAR indicates that the upstream Holtwood dam failure would not increase the level of the Conowingo pond such that it would exceed the grade level at the site. Therefore, a simultaneous Holtwood dam failure is not assumed to affect the availability of mitigation equipment.

8.2.3 Mitigation Strategies

NEI 06-12 discusses implementation strategies for SFP makeup and spray. The mitigation strategy is required to be implemented within two hours after the decision of deploying the mitigation strategy. This NEI guidance defines the actions that should be taken in situations in which normal procedures or command and control structures are not available. The notes in the parentheses below include items not considered applicable to the accident scenarios for the SFPS. The assumptions in the guidance include the following:

- An immediate threat warning does not occur.
- Access to the control room is lost (not expected in SFPS scenarios).
- Equipment or supplies normally located in the control room or in the building that houses the control room are lost (not expected in SFPS scenarios).
- Access to the building that contains the control room is lost (not expected in SFPS scenarios).
- All personnel normally in the control room are lost (not expected in SFPS scenarios).
- All ac and dc power required for operation of plant systems is lost (i.e., both class 1E and non-class 1E sources).
- Only minimum site staffing levels are available (i.e., weekend/backshift). Note: the minimum staff mentioned in the NEI guidance is not the minimum staff requirements.

Instead, it refers to the normal staffing level during weekend or backshift. This assumption does not apply when either Unit 2 or Unit 3 is in refueling outage.

- Other onsite control rooms and personnel in separated building are unaffected. (Personnel injury is likely to occur given the hypothetical earthquake.)
- Operations personnel who are not normally located in the control building are available for implementation of extensive damage mitigation guidelines.
- Nonlicensed personnel, typically an auxiliary operator, can take actions.
- The level of training on implementing procedures and guidance should be consistent with actions under severe accident management guidelines and should be consistent with utility commitments made under B.5.b Phase 1.
- Before the event, the plant systems are in a normal configuration with the reactor at 100-percent power. (This SFP safety analysis includes refueling outages (i.e., OCPs 1 and 2).)

The above items that are noted in parentheses as not being expected in the SFPS scenarios apply to TSG-4.2. TSG-4.2 may not apply to the SFPS scenarios. Instead, TSG-4.1 is the most applicable procedure for the SFPS scenarios. The sections below discuss the SFP mitigation strategies in accordance with TSG-4.1.

Internal Makeup

This strategy connects two fire hoses to the two existing fire system standpipes on the refueling floor to provide a minimum of 500-gpm total injection flow to the SFP. The fire system must be pressurized to implement this strategy. To implement this strategy, the operators need to remove the existing 1.5-in reducer from the two fire standpipes, connect two 2.5-in fire hoses to the two standpipes, and route the two fire hoses to the SFP. Operators can deliver makeup flow by fastening the hose to the SFP side for direct injection into the SFP. Operators can also deliver makeup flow by connecting the two fire hoses to the two spray nozzles to spray water into the SFP. This strategy will deliver a total spray flow of more than 200 gpm. All equipment mentioned is available on the refueling floor. This strategy assumes that the refueling floor is accessible for local makeup.

External Makeup and Spray

This strategy uses any of the two portable diesel pumps (Section 8.2.2) to inject or spray water into the SFP. This strategy requires the plant staff to (1) tow the portable diesel pump to the desired location at grade level, (2) lay two approximately 200-ft fire hoses that are connected by two sections from the refueling floor through a stairwell to the grade level (about 100 ft in elevation difference) to connect the hoses to the charging output of the portable pump, (3) connect the hose end on the refueling floor to an spray nozzle, and (4) connect the portable pump’s suction to a fire hydrant. The hoses for connecting the pump discharge to the spray nozzles are stored on the refueling floor. Each spray nozzle can be adjusted for spray or to obtain full flow (i.e., injection).

The external makeup and spray mitigation strategy uses the fire water system as the default water source to the portable diesel pumps. Under situations in which the fire water system is not available, the reference plant's procedure SO-37L.1.a, "Diesel Driven Portable and Diesel Driven High Capacity Portable Pump Startup and Shutdown," identifies additional water sources, including the inner pond, discharge pond, and Conowingo pond. In the event of a seismically induced failure of the Conowingo pond, the loss-of-pond procedure provides cooling water management strategies. In addition, the reference plant Assignment Report No. 01001590 identifies the candidate water sources, including high-pressure service water, fire water, residual heat removal water, condensate transfer water, and cross connections to the opposing unit's spent fuel water supply. Detailed step-by-step instructions for using water from the alternative water sources are not available. However, this HRA study credits the use of the alternative water sources in the situations when the fire water is not available because of the similarity of using water from these sources to drafting fire water.

<u>External Local Spray or Scrub</u>

This strategy uses any of the following three procedures, individually or in combination, to provide spent fuel cooling or secondary containment spray to scrub potential radionuclides released from the SFP primary or secondary containment structures:

(1) Use the portable diesel pump to provide water to the two spray nozzles on the refueling floor to spray water into the SFP. This strategy requires operators to lay out fire hoses from the refueling floor to grade level, as described in the section above entitled, "External Makeup and Spray."

(2) Use the portable diesel pump to provide water to one or two of the two spray nozzles located on the turbine building roof to spray water to the secondary containment or the refueling floor through building breaches. This strategy requires operators to lay out hoses from the turbine building roof to grade level to connect the portable pump and the spray nozzle.

(3) Use a ladder truck to spray water into the SFP area through building breaches from the steel structure surrounding the SFP floor. This strategy requires the use of an offsite fire company's 100-ft ladder fire truck. Exelon Generation Company, LLC (owner of the reference plant), has a letter of agreement with nearby Delta-Cardiff Volunteer Fire Company. Upon dispatch and without additional complexity, the fire truck could arrive at the reference plant within 30 minutes (Assignment Report No. 01001590). The Delta-Cardiff Volunteer Fire Company possesses two fire trucks from Seagrave Fire Apparatus, LLC, either one of which can perform the portable diesel pump function. Other nearby fire companies could support the reference plant mitigative efforts.

Items (2) and (3) above are useful primarily to mitigate the release of radioactivity off-site, but are not effective in preventing radioactive release from the SFP fuel rods. Because the mitigation success of this HRA study is to prevent radioactive release from the SFP fuel rods, items (2) and (3) are not credited in this HRA study. TSG-4.1 requires the plant to use the external local spray or scrub strategy after it has attempted the internal makeup and external makeup and spray strategies (as discussed earlier). For consistency with the NEI guidance, the reference plant uses the flowchart shown in Figure 101 as general guidance for deployment of SFP mitigation strategies.

Makeup with Residual Heat Removal Pump from the Torus

This strategy requires that electrical power is available for a residual heat removal pump to pump torus water into the SFP at a flow rate of 10,000 gpm. This flow rate is much larger than the maximum moderate leakage flow rate (i.e., approximately 1,900 gpm). In an SBO scenario, this strategy is not available because power is not available for the residual heat removal pumps. this HRA study assumes this strategy is not available.

Leakage Control

The reference plant has a list of stocked materials that could help to reduce the leakage flow rate, including steel plates, plywood, bag stopper, sealants, ropes, and rubber matting. Certain materials would require a crane for moving (e.g., 5/8-in by 4-ft by 4-ft steel plates. Based on its emphasis on the initiation of makeup strategies and the 72-hr scope of the analysis, the study did not consider repair options.

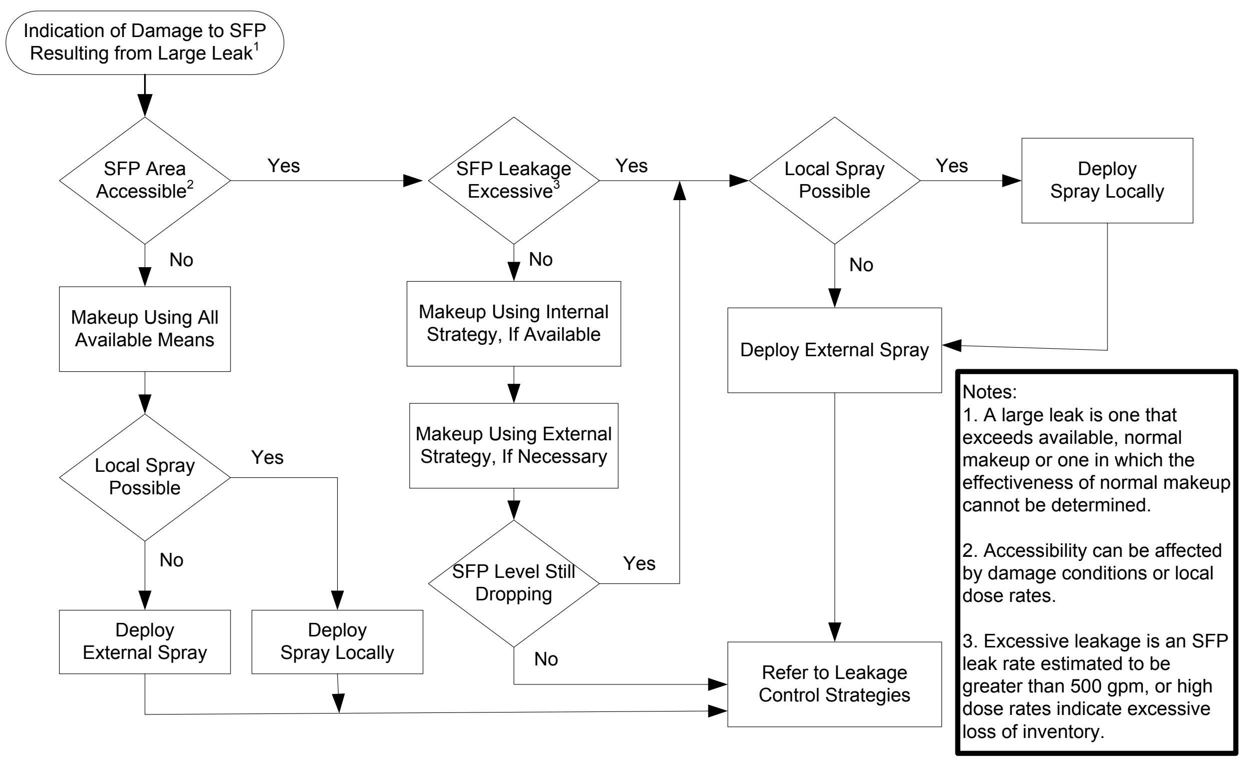


Figure 101 Generalized guidance for SFP makeup and spray decisions

8.3 Study Framework, Scope, and Approach

8.3.1 Study Framework and Scope

Preventing gap release of the Unit 3 SFP fuel is the success criterion defined in this HRA study. Because human performance is sensitive to the extent of earthquake damage to the plant, the study identifies a set of plant damage states and estimates an HEP for each damage state. Identification of the damage state is based on the status of a few key SSCs that include electric power availability, reactor status, and fire system availability. Figure 102 illustrates the framework and scope of this HRA study. The large rectangle with dashed lines shown in Figure 102 represents the scope of this HRA study. Each box within the dashed lines represents a probability. The partial enclosure of the damage state signifies that the scope of this HRA study only identifies the plant damage states; it does not estimate the probabilities of the damage states. Therefore, the gap release (and no gap release) probabilities could not be estimated, and the HEPs were computed to provide initial general insights.

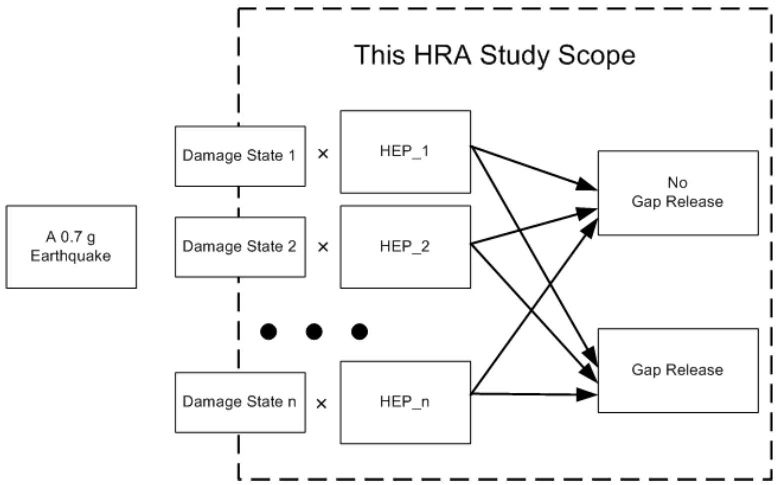


Figure 102 The study framework

8.3.2 Approach to Human Error Probability Estimates

8.3.2.1 A Two-Phased Approach to HEP Estimates

This HRA study used a two-phase approach to estimate mitigation failure probabilities or human error probabilities (HEPs). Phase 1 estimates HEPs for mitigating the reference plant SFP leak. These estimates consider the status of the electric power, reactor and primary containment, and fire system of Unit 3, the Unit 3 OCP, and Unit 3 SFP leakage rate. These are the dominant factors that affect mitigation of a single SFP leak. In Phase 2, other damage to the site that affects mitigation is discussed. Phase 2 involves situations that combine reactor and SFP problems or multiple unit problems caused by the same earthquake. The four discrete steps listed below represent the HEP estimation process used in this analysis. The four sections that follow discuss these steps in detail.

(1) Identify the time required for deployment of SFP makeup.
(2) Identify the damage states and corresponding available time.
(3) Estimate the HEP of each damage state.
(4) Identify additional feasibility considerations.

8.3.2.2 Step 1—Identify the Time Required

In this study, the total time necessary for deployment of the SFP makeup is the sum of the following three time segments:

(1) Delay Time: In an earthquake-induced SBO scenario, the control room operator's primary focus is on reactor safety. Although the SFP trouble alarm is triggered soon after the earthquake, a time delay occurs for starting a diagnosis process to investigate an SFP problem. The cue for starting to investigate the SFP is the earthquake procedure SE-5. Step 9 of the procedure instructs the operators to check the SFP, SFP cooling system, and fuel floor blowout panels. Based on an interview with PBAPS staff, the delay time ranges between 30 minutes and 1 hour. This study uses 45 minutes for the SBO scenarios, 30 minutes for LOOP scenarios, and 60 minutes for SBO without dc power scenarios because, based on crew interviews, the control room supervisor would, at a minimum, simultaneously implement SE-11 and SE-5. When less electricity is available for maintaining reactor safety margin, the operators would put more effort into restoring electricity (i.e., SE-11). As a result, less time is spent on SE-5, which consequently would delay implementation of Step 9 in SE-5 to send auxiliary operators to check the SFP status.

(2) Diagnosis Time: Diagnosis time is the time between when auxiliary operators are deployed to inspect the SFP and when they report SFP leakage back to the control room operators. Based on the leakage rate (both small leakage and moderate leakage) and leakage locations (i.e., the SFP bottom at the elevation of a few inches above the 195-ft floor), detecting SFP leakage is not a challenging task. Based on the same interview with PBAPS staff and a plant walkdown of the path that the auxiliary operators would normally take to inspect the SFP, the diagnosis time was determined to be 15 minutes.

(3) Action Time: The 2-hr implementation expectation in NEI 06-12 is used for deployment of the portable diesel pump to provide SFP makeup. The HRA uses the 2 hours as the action time at which the fire system is available because TSG-4.1 instructs the staff on how to use the fire system as the water source. When the fire system is not available, using water from the alternative water sources would require additional time. An additional 1 hour of action time is necessary when the fire system is not available.

Table 43 and Table 44 summarize the time estimates based on the above discussion. Table 43 shows the mitigation time estimates in the scenarios for which fire water is sufficient for mitigation. Table 44 adds 1 hour of action time to Table 43 to account for the effect of unavailable or insufficient fire water. In Table 43 and Table 44, note that the total time difference between the LOOP scenarios and SBO without dc scenarios is only 30 minutes. However, the conditional reactor core damage probability in these two scenarios would be significantly different. However, the conditional reactor core damage probability in these two scenarios would have a significant difference. That difference directly affects the refueling floor accessibility and, in turn, the mitigation success probability. This HRA study assesses HEPs for the SBO and SBO without dc scenarios with and without reactor core damage separately.

Table 43 Estimates of the Time Required for the Operator to Deploy SFP Makeup If Fire Water Is Available

	Delay Time	Diagnosis Time	Action Time	Total Time Required
LOOP	30 minutes	15 minutes	2 hours	2 hours 45 minutes
SBO	45 minutes	15 minutes	2 hours	3 hours
SBO without dc	60 minutes	15 minutes	2 hours	3 hours 15 minutes

Table 44 Estimates of the Time Required for the Operator to Deploy SFP Makeup If Fire Water Is Not Available or If It Cannot Deliver Sufficient Flow

	Delay Time	Diagnosis Time	Action Time	Total Time Required
LOOP	30 minutes	15 minutes	3 hours	3 hours 45 minutes
SBO	45 minutes	15 minutes	3 hours	4 hours
SBO without dc	60 minutes	15 minutes	3 hours	4 hours 15 minutes

8.3.2.3 Step 2—Identify the Damage States and Available Time

The key factors that affect the likelihood of successful mitigation of the Unit 3 SFP include SFP leakage size; OCP; and the status of the electric power, reactor and primary containment, and fire system of Unit 3. These factors characterize the damage states (as shown in Table 47). SFP leakage size and whether the SFP and the reactor cavity are hydraulically connected (i.e., during refueling and nonrefueling) largely determine available time. As discussed earlier, the available time is determined by the shorter time of either the SFP water reaching the top of the fuel rack or the refueling floor reaching 140°F. Table 45 shows the time required and time available of the damage states of interest assuming the Unit 3 reactor is not damaged.

Table 45 Estimates of time required and time available for mitigation

<table>
<tr><th></th><th></th><th colspan="2">Small Leak</th><th colspan="2">Moderate Leak</th></tr>
<tr><th></th><th></th><th>Time Required(hr)</th><th>Time Available(hr)</th><th>Time Required(hr)</th><th>Time Available(hr)</th></tr>
<tr><td rowspan="3">OCP 1</td><td>LOOP</td><td>2.75(3.75)</td><td rowspan="3">13.5</td><td>2.75(3.75)</td><td rowspan="3">6</td></tr>
<tr><td>SBO</td><td>3.0(4.0)</td><td>3.0(4.0)</td></tr>
<tr><td>SBO w/o DC</td><td>3.25(4.25)</td><td>3.25(4.25)</td></tr>
<tr><td rowspan="3">OCP 2</td><td>LOOP</td><td>2.75(3.75)</td><td rowspan="3">26</td><td>2.75(3.75)</td><td rowspan="3">6</td></tr>
<tr><td>SBO</td><td>3.0(4.0)</td><td>3.0(4.0)</td></tr>
<tr><td>SBO w/o DC</td><td>3.25(4.25)</td><td>3.25(4.25)</td></tr>
<tr><td rowspan="3">OCP 3</td><td>LOOP</td><td>2.75(3.75)</td><td rowspan="3">19</td><td>2.75(3.75)</td><td rowspan="3">2.5</td></tr>
<tr><td>SBO</td><td>3.0(4.0)</td><td>3.0(4.0)</td></tr>
<tr><td>SBO w/o DC</td><td>3.25(4.25)</td><td>3.25(4.25)</td></tr>
</table>

*The numbers outside the parentheses are the time required when the fire system is available. The numbers inside the parentheses are the time required when the fire system is not available.
**These values assume that the Unit 3 reactor is not damaged and the staff uses the portable diesel driven pumps for SFP mitigation

8.3.2.4 Step 3—Estimate Basic HEPs of a Single Unit Event

This step estimates the basic HEPs for each damage state based on the following assumptions and practices:

- The required mitigative equipment stored outside of the reference plant, Unit 3, and water sources are available. Step 4 considers equipment and water unavailability and other factors.
- The plant staff is available for performing the mitigation activities.
- The earthquake damaged much of the nonsafety piping and equipment.
- The purpose of including some situations in the HRA (e.g., core damage within the specified available time) is to explicitly identify the key factors that affect human performance. Estimating the likelihood of the occurrence of these situations is outside the scope of this HRA study. Estimating the likelihood of each situation would require the conduct of a PRA analysis.

The main considerations necessary for assessing HEPs are based on the time margin and workload that affect staffing availability. Electric power availability strongly affects workload. The power availability is classified into: (1) LOOP only, (2) SBO, and (3) SBO without dc. The three classes of power availability impose significant differences in operator workload that, in turn, affect personnel availability to perform all required tasks. The flow diagram in Figure 103 shows the HEP estimation procedure, which is based on NUREG-6883 “The SPAR-H Human Reliability Analysis Method” issued in 2005, supplemented with the NRC staff’s expert judgment.

SPAR-H’s low power and shut down diagnosis worksheets classifies time margin effects into five classes as shown in Table 46.

Table 46 Time margin effects on human error probability in the SPAR-H HRA method for cognitive activities in low power /shutdown operations

Class	HEP or HEP Multiplier	Note
Insufficient time	HEP = 1.0	Less than 2/3 of normally time required
Barely adequate time	HEP multiplier = 10	~2/3 of normally time required
Nominal time	HEP multiplier = 1	About the normally time required
Extra time	HEP multiplier = 0.1	Equal to or greater than 5 times of normal time required
Expansive time	HEP multiplier = 0.01	Equal to or greater than 50 times of normal time required

The SPAR-H's action worksheets use slightly different time scales to adjust the HEP. The adjusting factor 1 in Figure 103 represent time margin effects on HEP based on SPAR-H's classification. The adjusting factor 2 in Figure 103 represents the performance shaping factors of "complexity" and "ergonomics/human machine interface" of SPAR-H. Table 47 shows the HEP calculation results. Note that OCP1 moderate leak scenarios are likely to have gap release because the NEI recommended minimum mitigation flow rates are insufficient to prevent gap release. This is not reflected in Table 47 because this is not considered as human error in a typical HRA application.

Table 47 shows that fire system availability in general does not have significant effects on human error probability. Table 48 summarizes the qualitative results of the HRA with respect to the likelihood of gap release in various plant states with the assumption of no reactor core damage. Two plant states have an HEP of 1.0: moderate leak in OCP1 and moderate leak in OCP3. In OCP1 moderate leak scenarios, the high likelihood is because the NEI recommended minimum mitigation flow rates are insufficient to prevent gap release. The high likelihood is not shown in Table 47 because the failure is considered as a design issue rather than a human error from a conventional HRA perspective. In OCP3 moderate leak scenarios, the high likelihood is because of the short time available for response (i.e., about 2.5 hours).

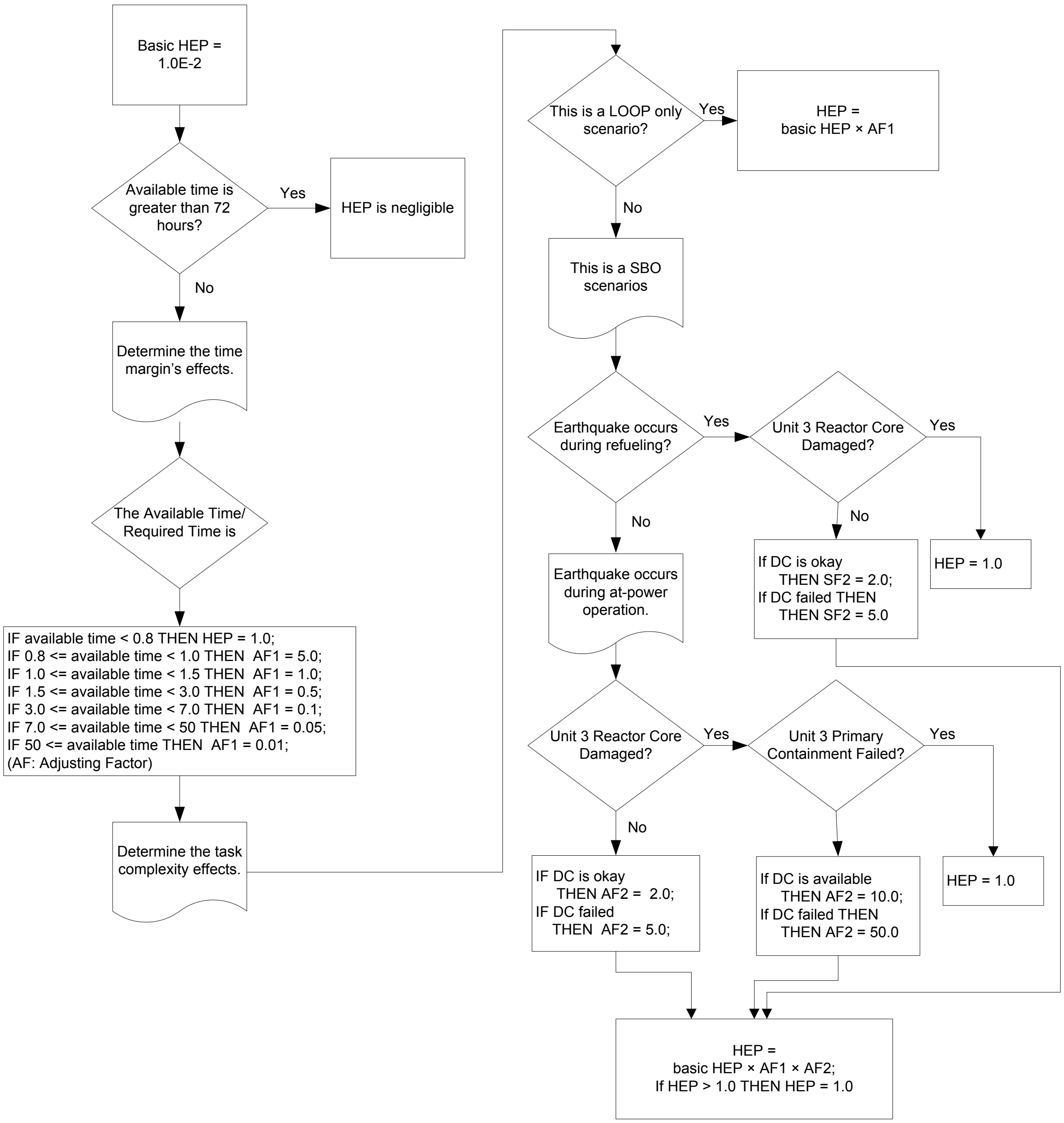


Figure 103 Flow chart for estimating HEPs for a single reactor unit event

Table 47 Human error probability estimates of a single unit event

			Small Leak	Moderate Leak
OCP 1	LOOP*		0.001 (0.001)	0.003 (0.003)
	SBO	No CD	0.006 (0.006)	0.002 (0.002)
		CD	1 (1)	1 (1)
	SBO w/o DC	No CD	0.015 (0.015)	0.005 (0.005)
		CD	1 (1)	1 (1)
OCP 2	LOOP*	No CD	0.003 (0.003)	0.0005 (0.001)
	SBO	No CD	0.006 (0.006)	0.001 (0.002)
		CD	1 (1)	1 (1)
	SBO w/o DC	No CD	0.015 (0.015)	0.0025 (0.005)
		CD	1 (1)	1 (1)
OCP 3	LOOP*	No CD	0.01 (0.1)	0.001 (0.001)
	SBO	No CD	0.2 (1)	0.002 (0.002)
		CD; CTM intact	1 (1)	0.05 (0.05)
		CD, CTM breach	1 (1)	1 (1)
	SBO w/o DC	No CD	0.5 (1)	0.005 (0.005)
		CD; CTM intact	1 (1)	0.05 (0.05)
		CD, CTM breach	1 (1)	1 (1)

*Assume no reactor core damage (CD)
**The numbers outside the parentheses are the HEPs when the fire system is available. The numbers inside the parentheses are the HEPs when the fire system is not available

Table 48 The likelihood of gap release*

	Small Leak (5%)**	Moderate Leak (5%)**
OCP 1 (0.9%)**	Low[1]	High[2]
OCP 2 (2.4%)**	Low[1]	Low[3]
OCP 3 (5.0%)**	Low[1]	High[4]

*Assumes only one SFP damaged without concurrent reactor core damage
**The probabilities are conditional probabilities given that the studied earthquake occurs
[1]The available time for response is long. The SFP fuel is submerged if SFP makeup is deployed in time.
[2]The NEI recommended minimum mitigation flow rate is insufficient to prevent gap release.
[3]The NEI recommended minimum mitigation flow rate is sufficient to prevent gap release.
[4]The available time for response is short so that the SFP makeup will likely not be deployed in time to prevent gap release.

8.3.2.5 Step 4—Additional Feasibility Considerations

This final step (i.e., Step 4) identifies situations that occur outside of the reference plant, Unit 3, that would have adverse effects on Unit 3 SFP mitigation. These effects are not considered in Step 3. These additional considerations include the following:

- Equipment demand cannot be met: When the earthquake causes extensive damage to the reactors and SFPs of Unit 2 and Unit 3 and the normal reactor and SFP cool down mechanisms are not available, the two portable pumps may not be available for Unit 3 SFP makeup given the multiple demands. The DDHCP pump has two 4” discharge connections, and the DDP pump has one 4” discharge connection. In combination, the two portable diesel pumps can deliver three times the NEI recommended minimum mitigation flow rate. The operators have to decide how to use the limited equipment for multiple problems for the reference plant’s two reactors and two SFPs. The decision will strongly depend on the situation.

- Damage to the mitigation equipment (e.g., the DDHCP pump and DDP pump) and support equipment (e.g., pump accessories and the designated truck to tow the pumps) would reduce the available equipment or delay mitigation.

- Simultaneous large or multiple fire events that demand more plant staff personnel than those available.

- Structural damage causes plant personnel injury that could result in less than adequate personnel available for SFP mitigation.

- Unit 3 Refueling floor is inaccessible for reasons such as Unit 3 reactor damage causing high radiation in the access path or other damage.

The cells in Table 48 with low gap release likelihood can be split into two groups:

- Greater than 13 hours for all small leak scenarios

- About 6 hours for OCP 2 moderate leak scenarios

In either situation, sufficiency of plant response personnel is likely not an issue because of the long available time of the small leakage scenarios and in refueling outage of the OCP 2 scenarios. Though not accomplished through a full scope PRA, this HRA attempted to account for the complexity of handling multiple reactors and SFP damage events. As such, an adjusting factor of 50 (based on the SPAR-H's performance shaping factors of "high complexity" and "low experience/training") was applied to Table 47. The results are summarized in Figure 104 and Table 49.

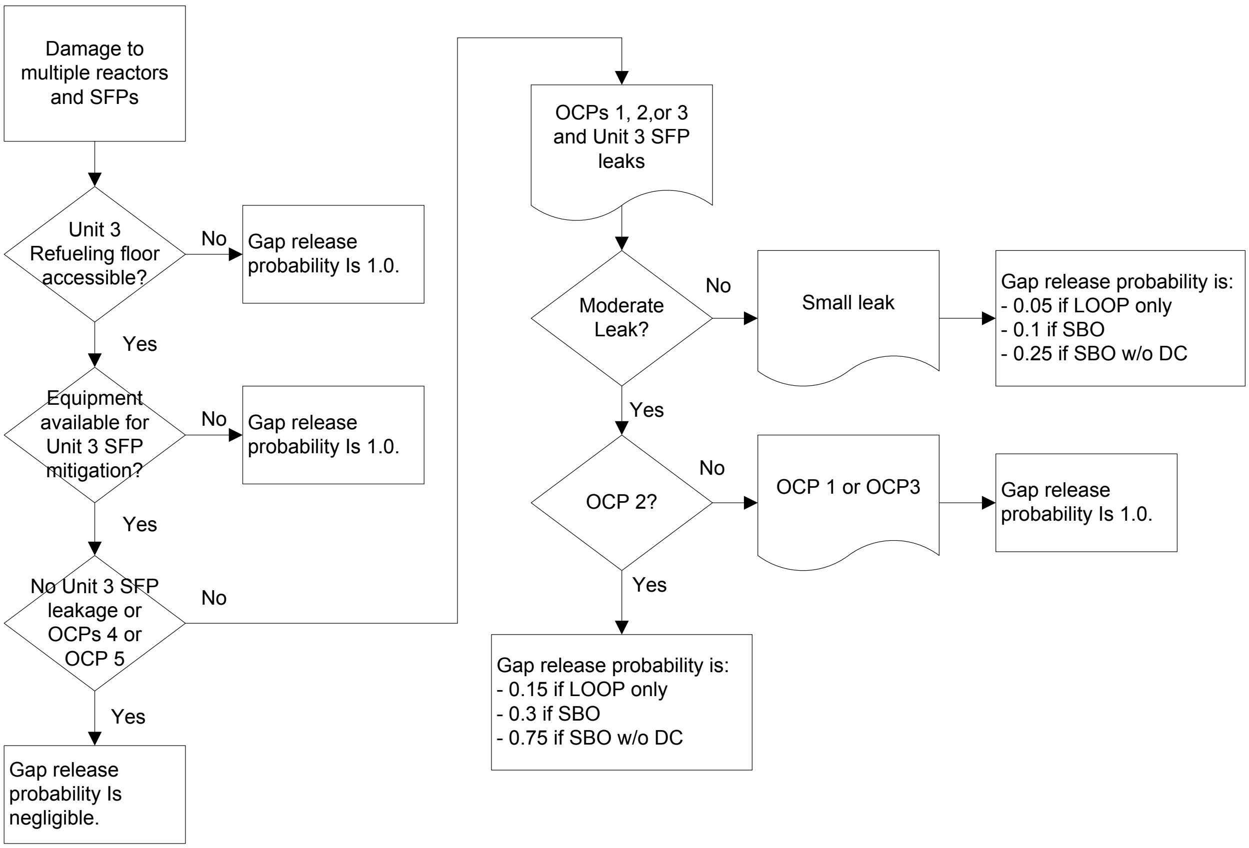


Figure 104 The gap release probability assessments given damage to multiple reactors and SFPs.

Table 49 shows three levels of likelihood of having radioactive release from the Unit 3 SFP fuel rods. Three colored coded regions are discussed below:

- Green Cells
  Two sub groups in the green coded cells: (1) the "no leak" scenarios have long available time (greater than 7 days) for response. The mitigation failure probability is determined to be negligible; and (2) The OCP4 and OCP5 have low decay heat. Even without mitigation, radioactive release is not expected.
- Yellow Cells:
  For the small leak scenarios, the available time ranges from more than 13 hours to more than 1 day. Give the long time available, time is not a critical factor affecting mitigation success. The mitigation failure probability is estimated to range from one failure out of

twenty to one failure out of four. For the OCP2 moderate leak scenarios, the time available is 6 hours. This increases the mitigation failure probability compared to the small leak scenarios. The mitigation failure probabilities for OCP 2 moderate leak scenario range from one failure out of twenty to three failures out of four.

- Red Cells:
  Two red cells are in Table 49. The OCP1 moderate leak scenario is red because the 500 gpm of injection or 200 gpm spray is insufficient to prevent fuel overheating. The OCP3 moderate leak scenario has only a short time available (2.5 hours), and mitigation is not expected to be deployed in time.

Table 49 Scenario Specific Human Error Probability Estimates*.

<table>
<tr><th></th><th>No Leak
(90%)</th><th>Small Leak
(5%)</th><th>Moderate Leak
(5%)</th></tr>
<tr><td>OCP 1 (0.9%)</td><td rowspan="4">Negligible</td><td rowspan="3">- 0.05 if LOOP only
- 0.1 if SBO
- 0.25 if SBO w/o DC</td><td>1.0**</td></tr>
<tr><td>OCP 2 (2.4%)</td><td>- 0.15 if LOOP only
- 0.3 if SBO
- 0.75 if SBO w/o DC</td></tr>
<tr><td>OCP 3 (5.0%)</td><td>1.0***</td></tr>
<tr><td>OCP 4 and
OCP 5 (91.7%)</td><td colspan="2">Inconsequential</td></tr>
</table>

- OCP: Operating Cycle Phase
- Percentages above are the percent of the time for the corresponding condition.
* Assume mitigating equipment is available for Unit 3 SFP, and Unit 3 reactor status does not deny access to the Unit 3 refueling floor.
**The NEI recommended minimum mitigation flow rate is not sufficient to prevent gap release. The procedure (i.e., TSG-4.1) does not instruct operators to establish an additional SFP makeup flow path to significantly increase the SFP makeup flow rate to be greater than the minimum flow rate recommended by NEI. The HEP is set to 1.0 to indicate that gap release would occur.
***Primarily due to short time available for response (i.e., ~ 2.5 hours). OCPs 1 and 2 (i.e., during refueling) have the reactor cavity and SFP hydraulically connected, which provides more time than OCP3.

8.4 Discussion and Summary

This SFP HRA study identifies a set of plant damage states and calculates the corresponding HEPs; however, it does not calculate the conditional probabilities of the damage states. The following information summarizes the human performance insights:

- The HEPs of the SFP no leakage scenarios are negligible because of the long time available for response. The scenarios in OCPs 4 and 5 would not lead to gap release of the SFP fuel because of the low spent fuel decay heat. These two groups of scenarios share 99.2 percent of the probability (i.e., 0.992). In other words, given the 0.5–1.0g earthquake, the SFPS estimates minimum 99.2-percent conditional probability that a gap release would not occur.

- 500 gpm of injection is not sufficient to prevent gap release in the OCP1 moderate leak scenarios, as determined in the SFPS. The SFPS did not perform sensitivity calculations to determine the NEI recommended flow rates (either injection or spray) to prevent gap release in this case. Therefore, this HRA study assumes that the plant staff would need to connect more than the procedurarlized two hoses to the portable diesel pump and use more than two spray nozzles to provide sufficient cooling. Because TSG-4.1 only provides instructions on establishing two hoses and two spray nozzles, the lack of procedures and insufficient equipment (i.e., hoses and spray nozzles) are assumed to cause the mitigation to fail in the OCP1 moderate leak scenarios. Even though the reference plant flow rates are greater than the NEI recommended minimum flow rate, sensitivity calculations on the actual flow rate from the spray nozzle would be needed for a more detailed assessment.

- The available time for SFP mitigation is determined by the shorter time of either the SFP water draining to the top of the fuel or the refueling floor reaching 140°F. In the OCP 1 and 2 small leakage scenarios, the refueling floor reaches 140°F earlier than the time necessary for the SFP water to drain to the top of the fuel rack thus causing refueling floor temperature to become the limiting factor for determining HEPs

- The two spray nozzles (as illustrated in TSG-4.1) for SFP makeup are set up in high radiation areas. Delivering the same amount of flow from a low-dose area (e.g., near the wall next to the storage pool) would significantly increase the available time because using the time necessary for the SFP water level to reach the top of the fuel rack as a criterion is based on the radiation level at the locations of the spray nozzles, as specified in TSG-4.1. Moving the spray nozzles setup locations to a lower dose area would significantly increase the mitigation success probabilities for moderate leak scenarios for which the time necessary for SFP water to drain to the top of the fuel rack is the limiting factor.

- The fire system availability (from earthquake-induced fire piping rupture) affects OCP 3 moderate leakage scenarios but not small leak scenarios because the small leak scenarios have at least 13 hours for mitigation deployment. Instructions on how to quickly determine whether the fire system can deliver sufficient flow for mitigation may improve the probability of successful mitigation.

The success criterion of this HRA study is to prevent a radioactive release from the Unit 3 SFP fuel rods. The mitigation strategies that emphasize keeping the radioactivity released from the fuel rods on site are not within this HRA scope. Deploying these strategies could mitigate radioactive releases to the environment.

The HEP results shown in Table 49 are based on the assumptions that mitigation equipment is available, there is no combination of Unit 3 reactor core damage and primary containment failure that causes inaccessibility of the refueling floor, and there is sufficient staff to deploy for the Unit 3 SFP mitigation. If the earthquake damages multiple reactors and SFPs some of the above assumptions may not apply. An analysis of these issues would require the performance of a combination of probabilistic risk assessment and associated HRA.

9. CONSIDERATION OF UNCERTAINTY

This section catalogues a set of sensitivity analyses to better understand the potential effect of certain assumptions on the results of this study. The sensitivity analyses include those for analyzing additional plant states (e.g., 1x8 pattern in a high-density loading configuration) and for analyzing parameter/model uncertainties (e.g., hydrogen combustion ignition). The assumptions analyzed were chosen from the list of key assumptions compiled in Section 2, based on their perceived importance and project constraints.

9.1 Sensitivity to Hydrogen Combustion (MELCOR)

A sensitivity calculation was performed to examine the response of the SFP to the hydrogen combustion ignition criterion. This calculation involved reducing the hydrogen concentration from 10 percent to 7 percent given the inherent uncertainties in this parameter discussed before. The case that showed the strongest sensitivity to this parameter is the unmitigated high-density, moderate leak size scenario from OCP 2. The base case reactor building concentration of gases in Figure 105 shows that, by the time the hydrogen concentration exceeds the ignition criterion of 10 percent, the oxygen concentration is below the 5-percent limit and no hydrogen combustion is predicted. However, at about 18 hours, both the hydrogen and oxygen concentrations are above 7 percent, which can support a hydrogen combustion. Figure 106 shows the mole fraction of gases for this sensitivity case. At about 18 hours, the hydrogen combustion consumes the hydrogen in the building as evidenced by the rapid decrease in the hydrogen concentration and is accompanied by a sudden increase in the oxygen concentration as the failure of the reactor building causes the outside air to enter. Following the air ingress, the clad oxidation power significantly increases (compare the base case in Figure 107 with the sensitivity case in Figure 108). The higher oxidation power leads to higher clad temperatures[42] (Figure 109 and Figure 110) and additional release of fission products from the fuel and release to the environment (Figure 111 and Figure 112). The cesium release fraction of 50 percent for this sensitivity is much higher than the base case of 1.6 percent (see Table 27), and it is comparable to the release fraction of 49 percent for the uniform pattern (see Table 50).

[42] The failure of the fuel rods leads to formation of debris that continues to release fission products.

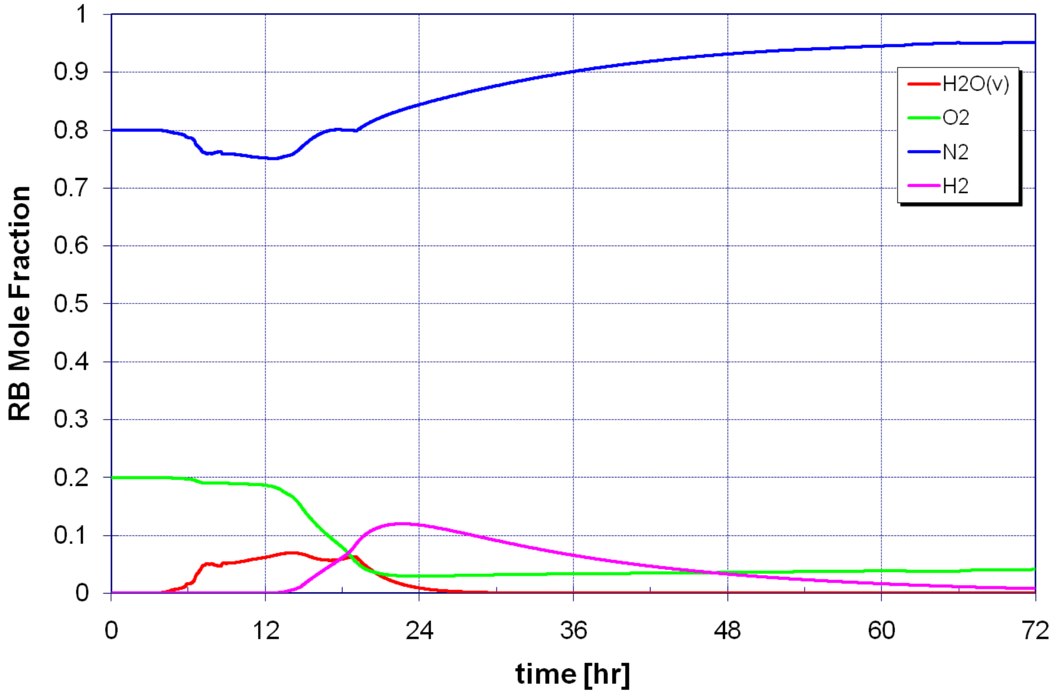


Figure 105 Reactor building mole fraction for unmitigated high-density moderate leak (OCP2)

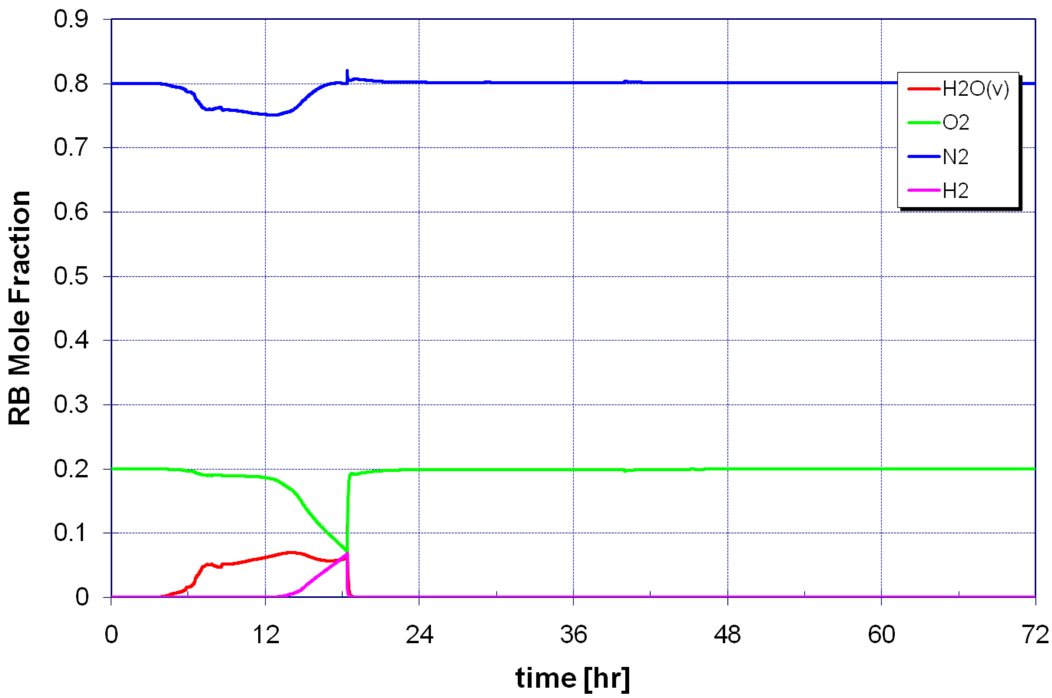


Figure 106 Reactor building mole fraction for unmitigated high-density moderate leak (OCP2-S)

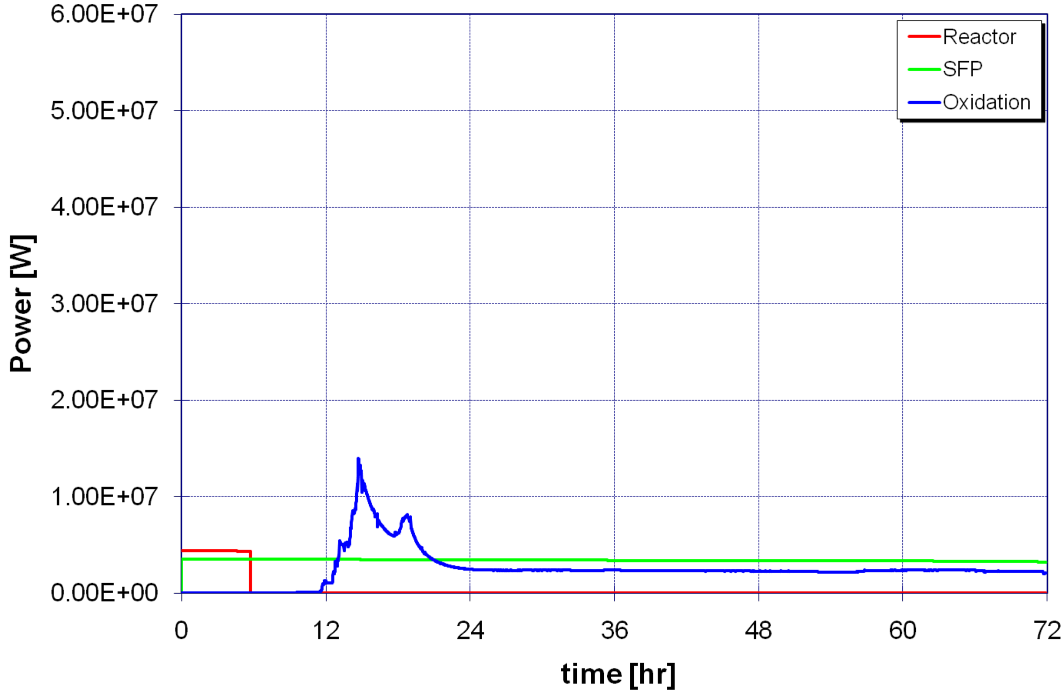


Figure 107 SFP power for unmitigated high-density moderate leak (OCP2)

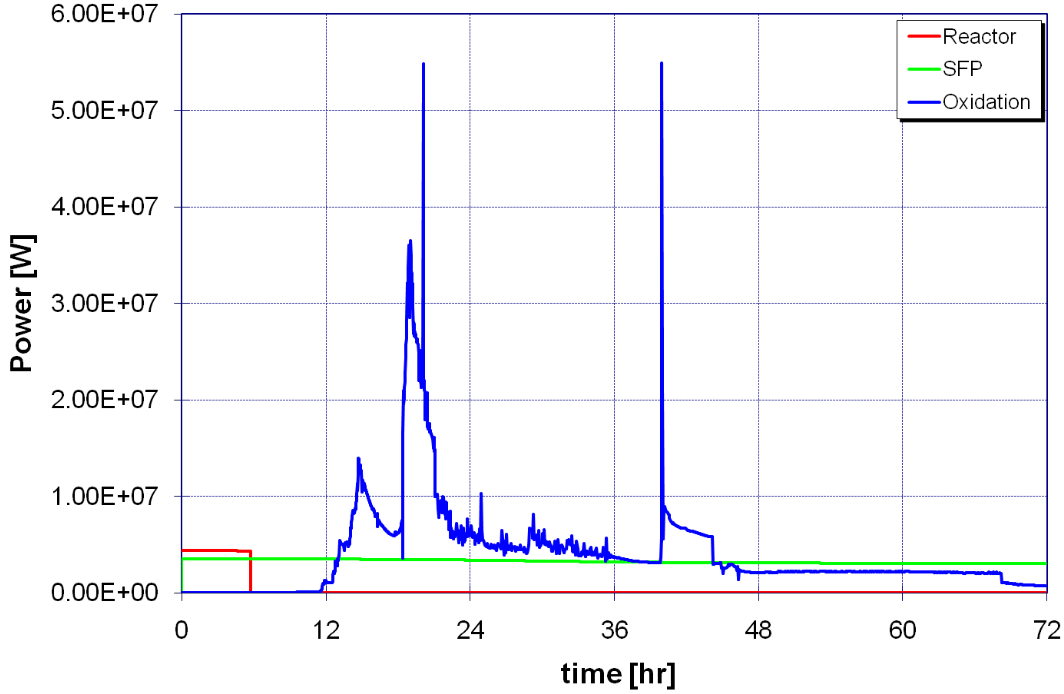


Figure 108 SFP power for unmitigated high-density moderate leak (OCP2-S)

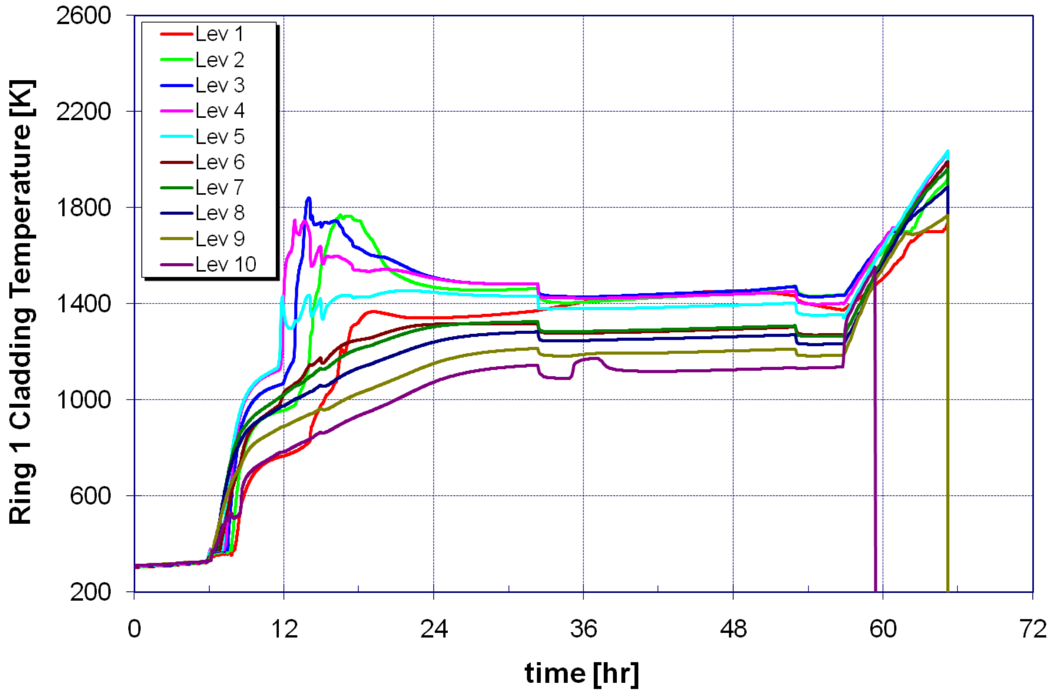


Figure 109 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP2)

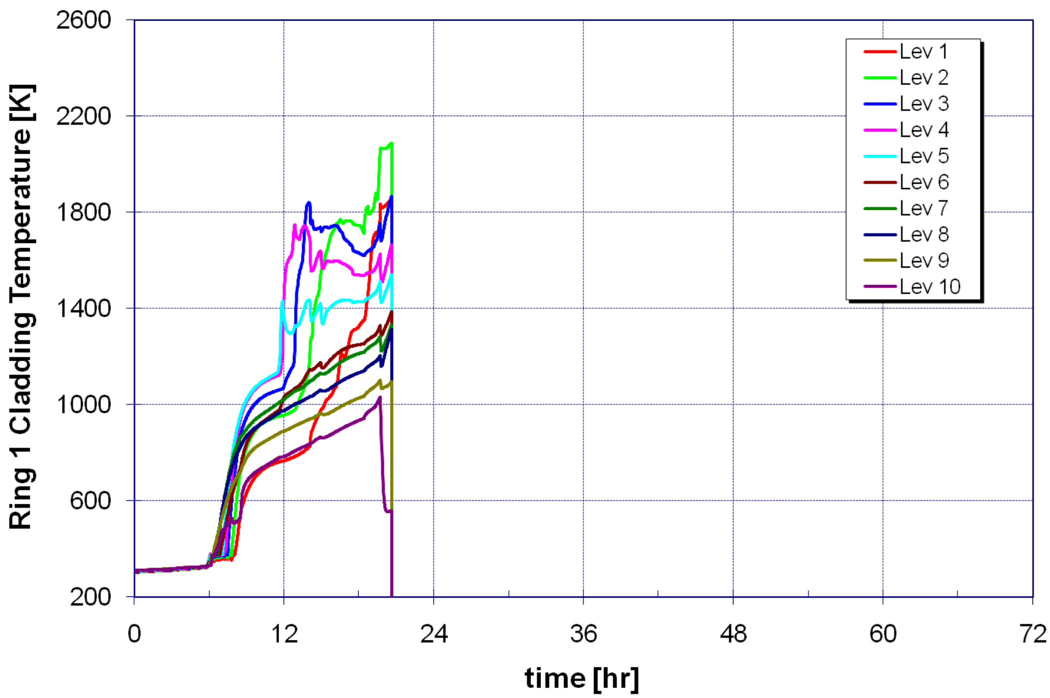


Figure 110 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP2-S)

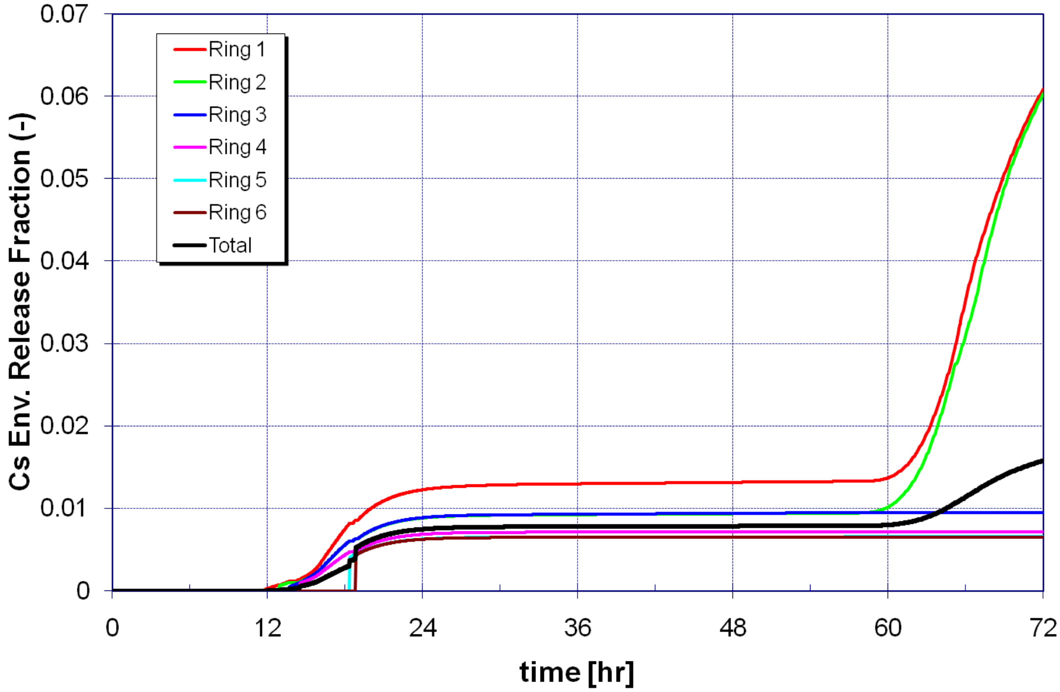


Figure 111 Cesium environmental release fraction for unmitigated high density moderate leak (OCP2)

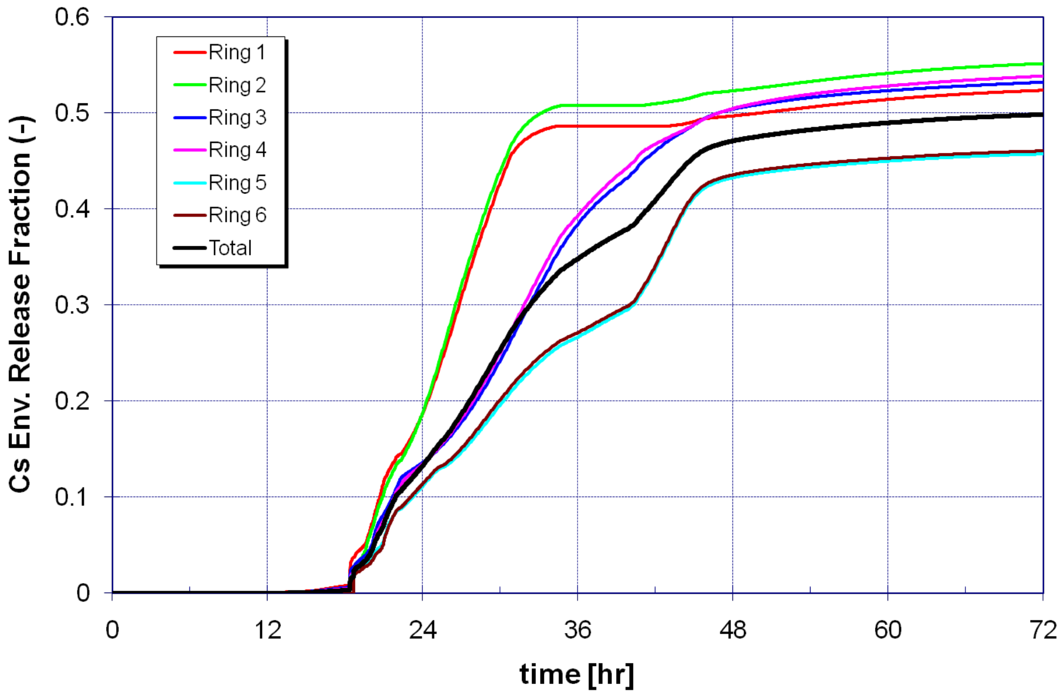


Figure 112 Cesium environmental release fraction for unmitigated high density moderate leak (OCP2-S)

9.2 Sensitivity to 1x8 Fuel Assembly Pattern (MELCOR)

This sensitivity involves a more favorable fuel pattern in which the hot assemblies are surrounded by eight cold assemblies. Figure 113 shows the assembly layout in a 1x8 pattern in which the 284 assemblies from the last offload are grouped into Rings 1, 3, and 5 (see Figure 46 for the 1x4 pattern). Rings 2, 4, and 6 contain all of the old fuel and have a total of 2,771 assemblies with their total decay heat distributed in each ring scaled by the number of assemblies.

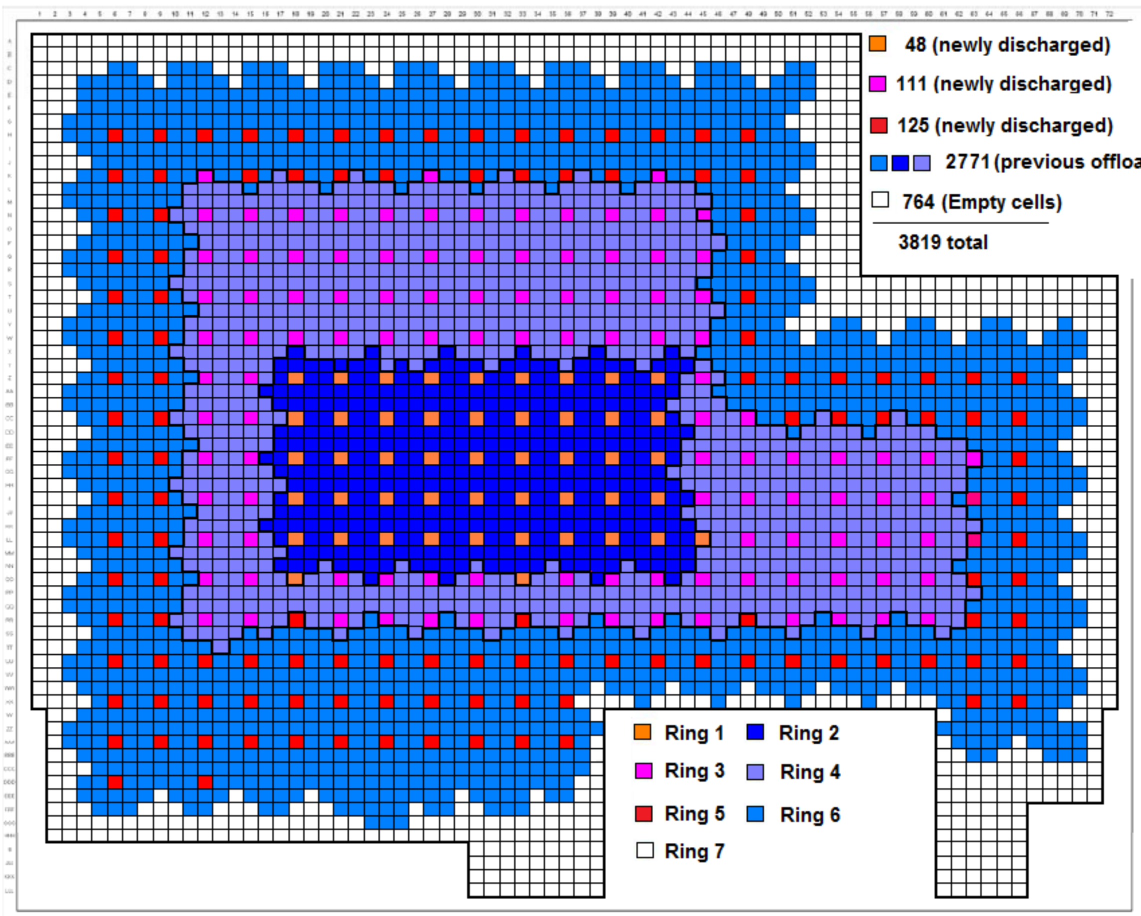


Figure 113 Layout of assemblies for OCP3 high density (1x8) model

A number of sensitivity calculations were performed for the high density, small leak scenarios in OCP2 and OCP3 (which had the highest release as a result of hydrogen combustion). Figure 114 and Figure 115 show the thermal response of the fuel in the 1x8 configuration for OCP3. Figure 114 shows that the highest power fuel assemblies in Ring 1 do not undergo a zirconium fire and the temperatures remain low enough to avoid gap release for the duration of the transient. The midplane fuel temperatures in the pool shown in Figure 115 have a more uniform heat up of the fuel assemblies than the comparable 1x4 pattern. There is more mass of the cold assemblies in the 1x8 pattern, which leads to lower heatup of the fuel. The fuel thermal response in the 1x8 pattern can be contrasted to the 1x4 pattern as shown in Figure 116 and Figure 117. For the 1x8 calculation, no release occurs from the fuel through 72 hours. In the 1x4 layout, a zirconium fire propagation began at 40 hours, which led to a 42-percent release of cesium inventory to the environment. For the OCP2 configuration results shown in Figure 118 and Figure 119, the decay heat is high enough to cause a zirconium fire in the hottest assemblies, even though the peak fuel temperatures in the 1x8 pattern are somewhat lower. The beneficial effect of the 1x8 pattern is also evidenced by the lower release fractions, as shown in Figure 120 and Figure 121.

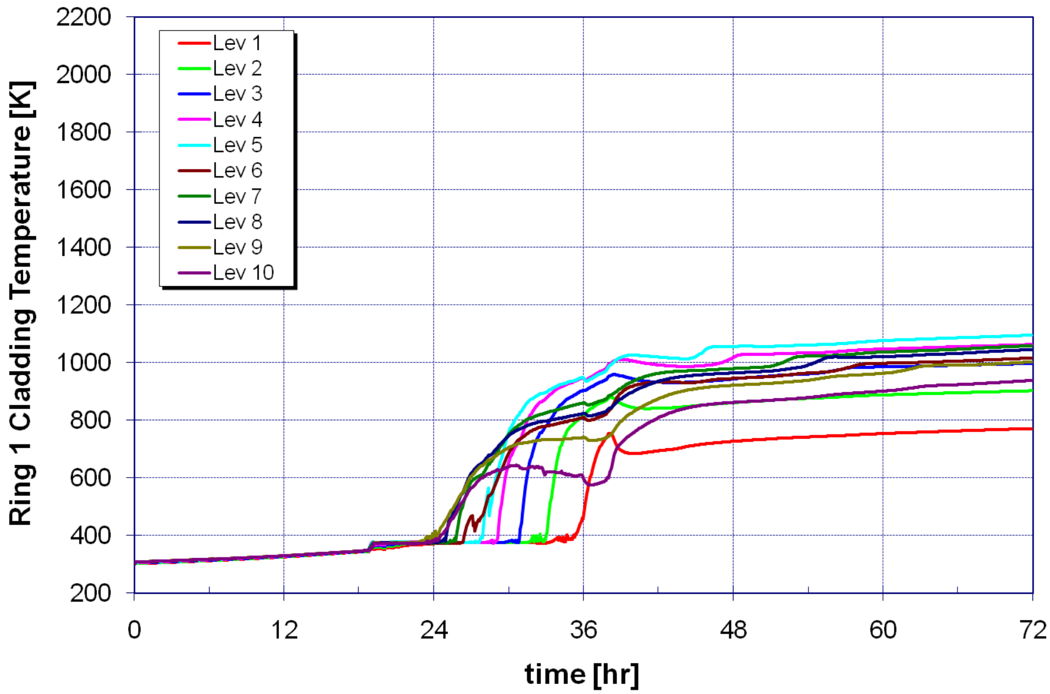


Figure 114 Ring 1 clad temperature for unmitigated high-density small leak (OCP3; 1x8)

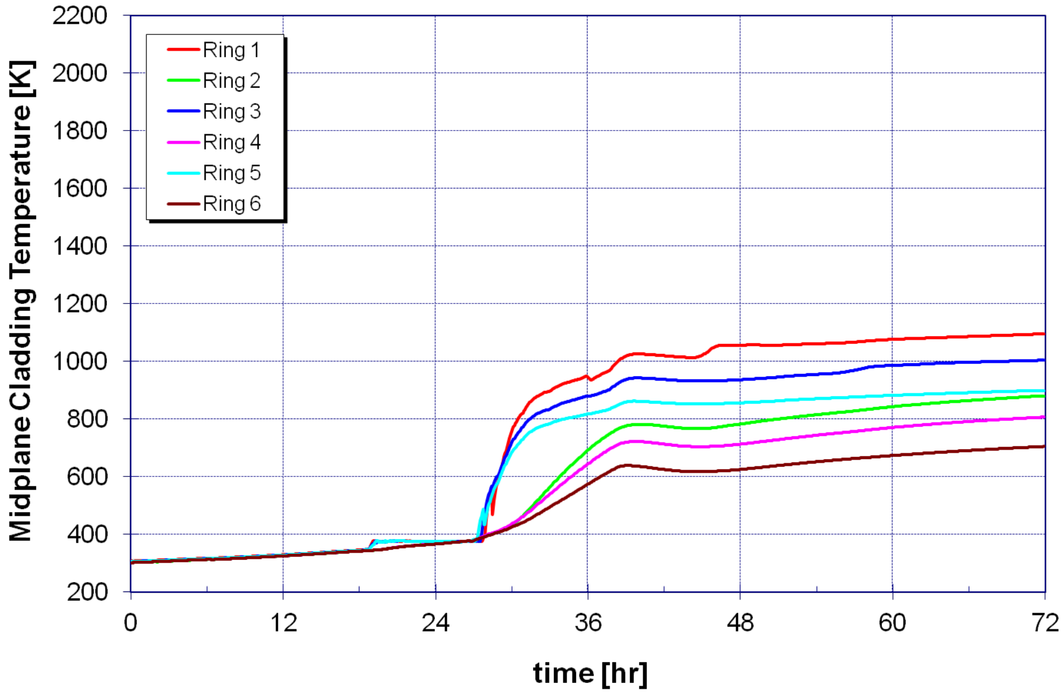


Figure 115 Midplane clad temperature for unmitigated high-density small leak (OCP3; 1x8)

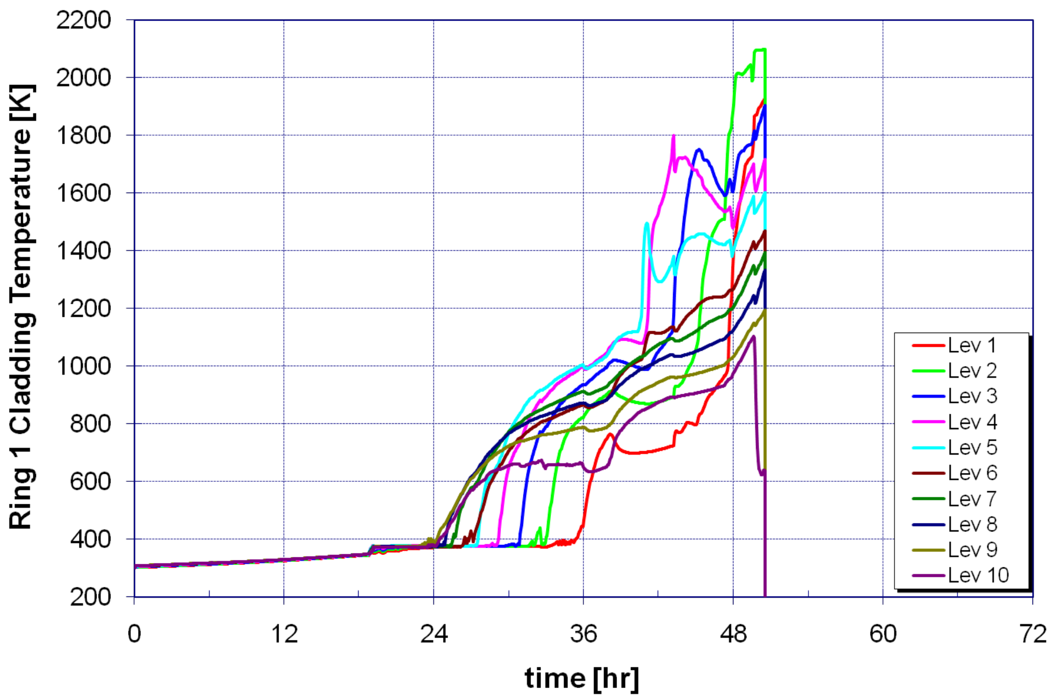


Figure 116 Ring 1 clad temperature for unmitigated high-density small leak (OCP3; 1x4)

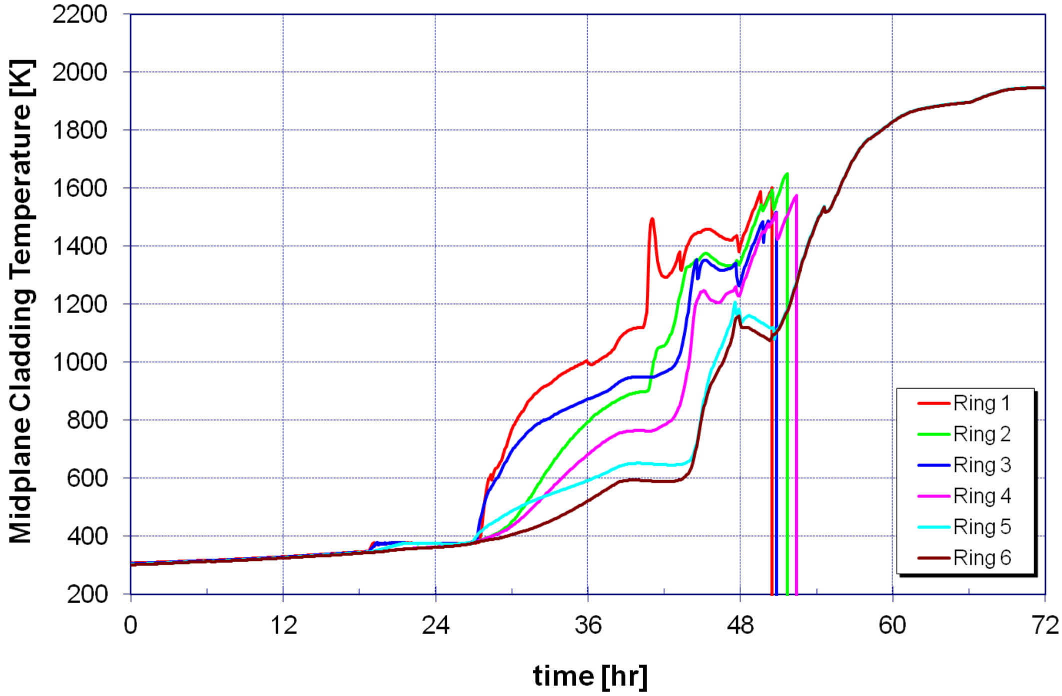


Figure 117 Midplane clad temperature for unmitigated high-density small leak (OCP3; 1x4)

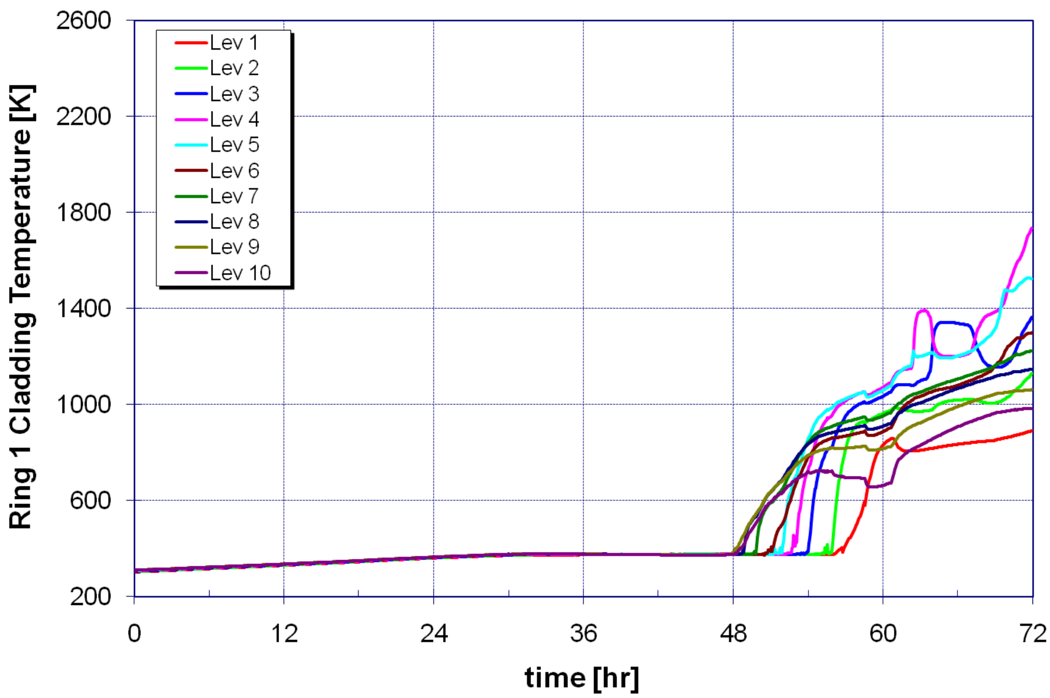


Figure 118 Ring 1 clad temperature for unmitigated high-density small leak (OCP2; 1x8)

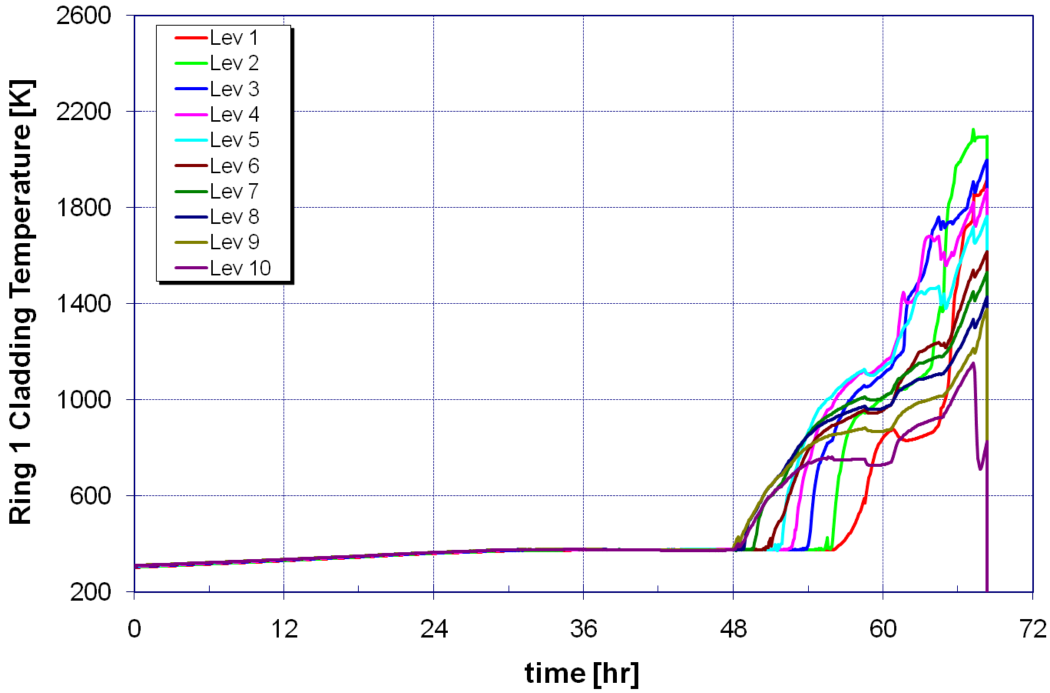


Figure 119 Ring 1 clad temperature for unmitigated high-density small leak (OCP2; 1x4)

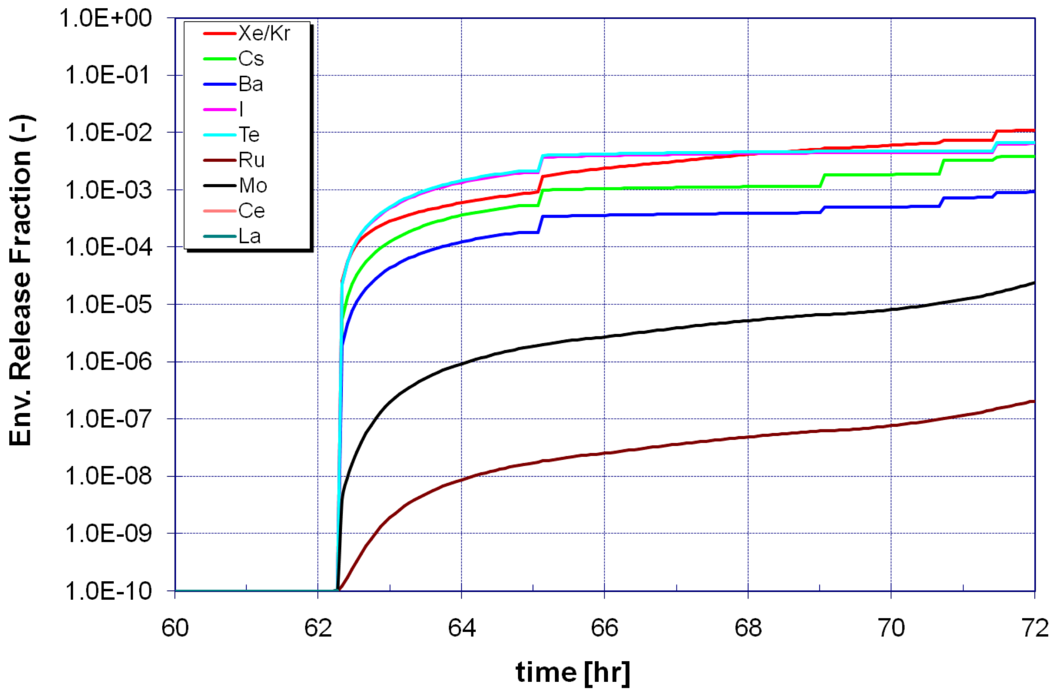


Figure 120 Environmental release fractions for unmitigated high-density small leak (OCP2; 1x8)

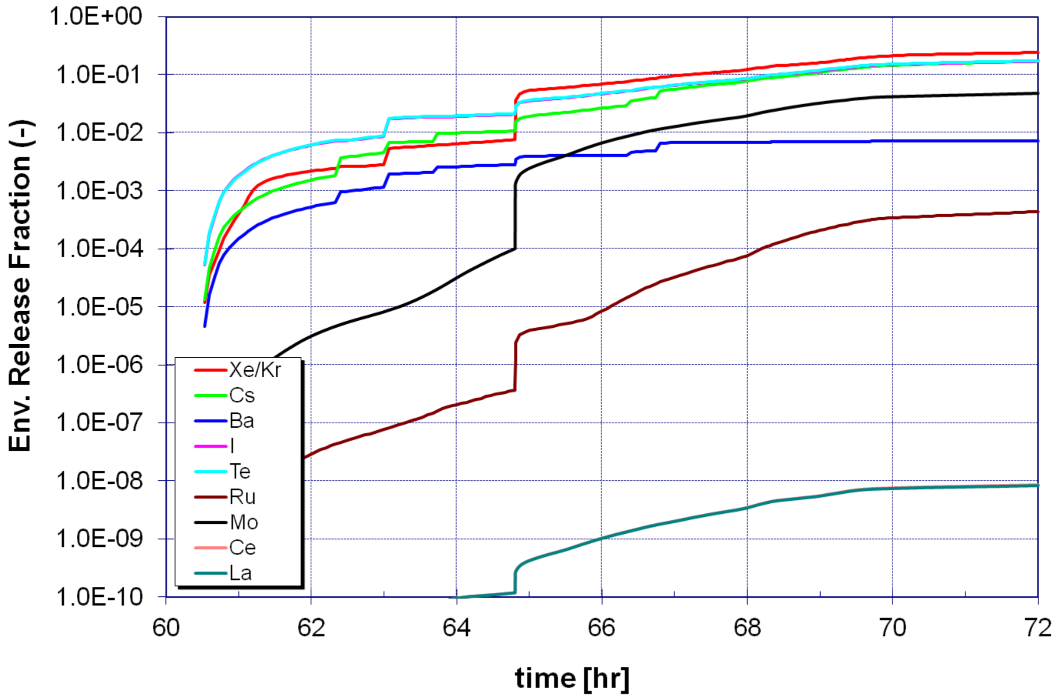


Figure 121 Environmental release fractions for unmitigated high-density small leak (OCP2; 1x4)

9.3 Sensitivity to a Contiguous (Uniform) Fuel Pattern during an Outage (MELCOR/MACCS2)

The reference plant studied has prearranged the SFP such that discharged assemblies can be placed directly into a 1x4 (actually 1x8 in the case of PBAPS) arrangement for the last two outages for both operating units. This approach is consistent with the requirements previously discussed in Section 5.1. However, those requirements do allow for the fuel to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. A requirement is 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 in which the fuel is unfavorably arranged during the outage to demonstrate the effect of this aspect on the results.

Figure 122 and Figure 123 show the layout of assemblies for the OCP1 and OCP2 uniform configuration. For the 1x4 pattern (see Figure 44), 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 pattern (Figure 122), the surface areas between Rings 1 and 2 and between Rings 3 and 4 were effectively reduced by about an order of magnitude, assuming that all of 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 was based on four panels per assembly. In the uniform pattern, the number of panels per assembly is estimated as 0.4 for Ring 1 ($4(88)^{1/2}/88$) and 0.3 ($4(196)^{1/2}/196$) for Ring 3. This is a stylized representation of a uniform configuration which limits the areas (and thus total heat transfer) between the hot rings and the rest of the assemblies in the pool.

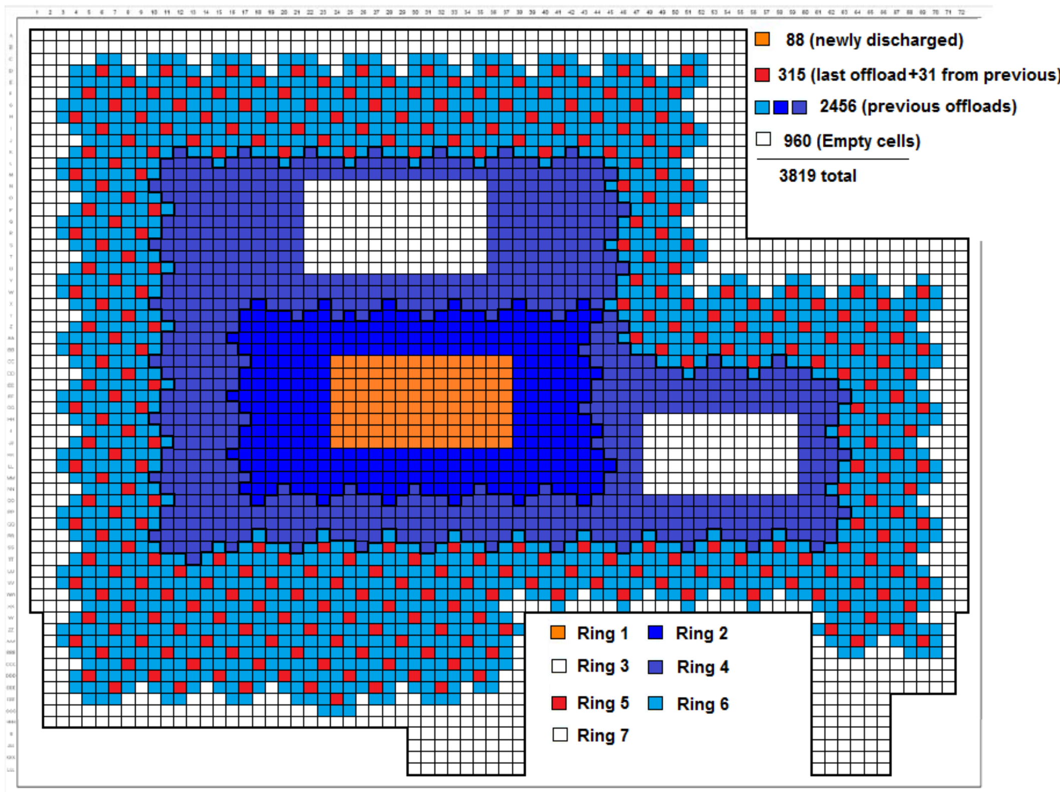


Figure 122 Layout of assemblies for OCP1 high-density (uniform) model

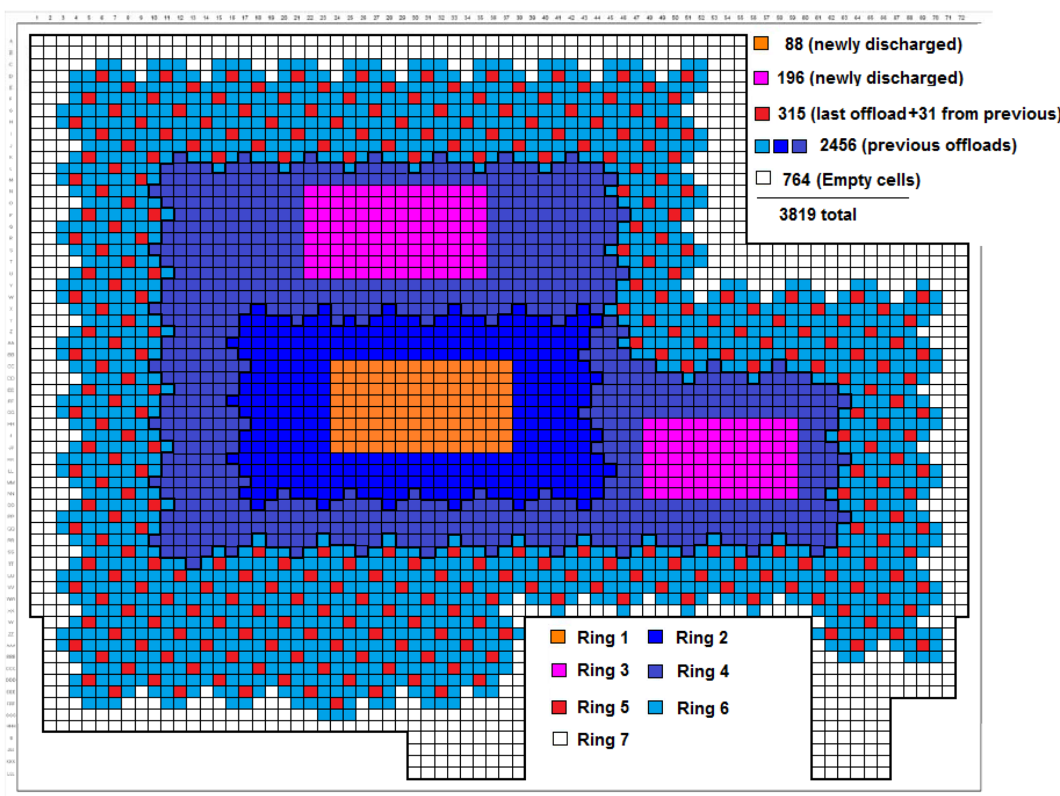


Figure 123 Layout of assemblies for OCP2 high-density (uniform) model

Unmitigated Moderate Leak (OCP1 Uniform) Scenario

Figure 124 and Figure 125 show the results of the calculation for the uniform OCP1. A comparison of the heatup with the 1x4 geometry (Figure 67) shows the higher temperatures in the uniform Ring 1 configuration because 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 heatup as the oxygen in the building is depleted, and formation of debris restricts airflow through the assemblies. Eventually, all of the fuel in Ring 1 collapsed and formed a debris bed. There is continuous release from Rings 1 and 2 and the overall cesium release to the environment is about twice of that in the 1x4 geometry (see Figure 72).

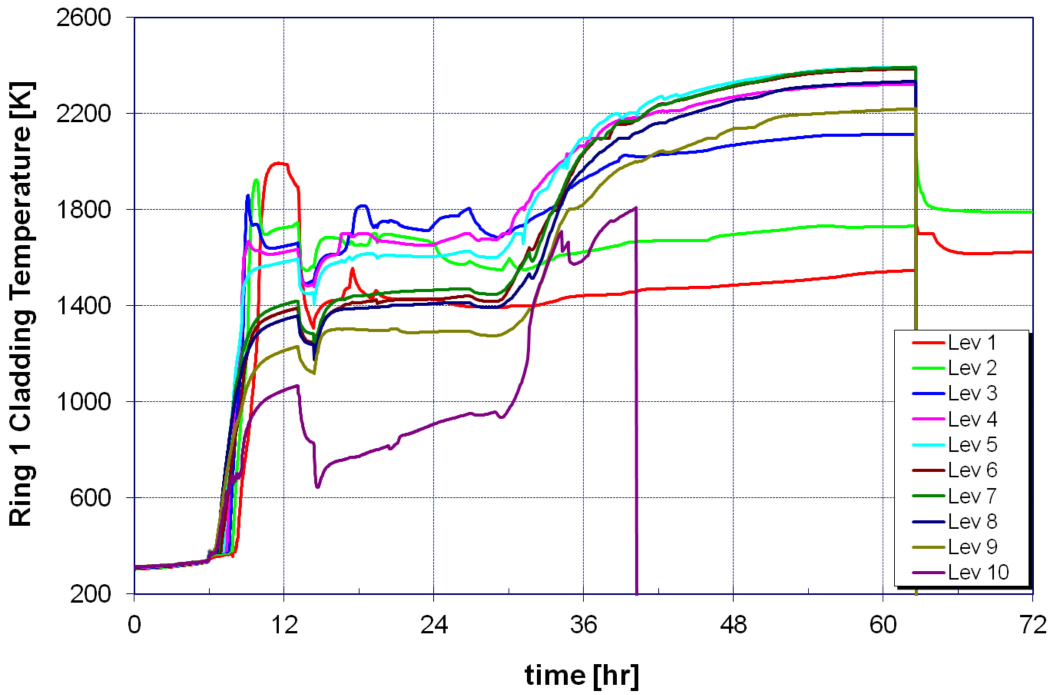


Figure 124 Ring 1 clad temperature for unmitigated uniform high-density moderate leak (OCP1)

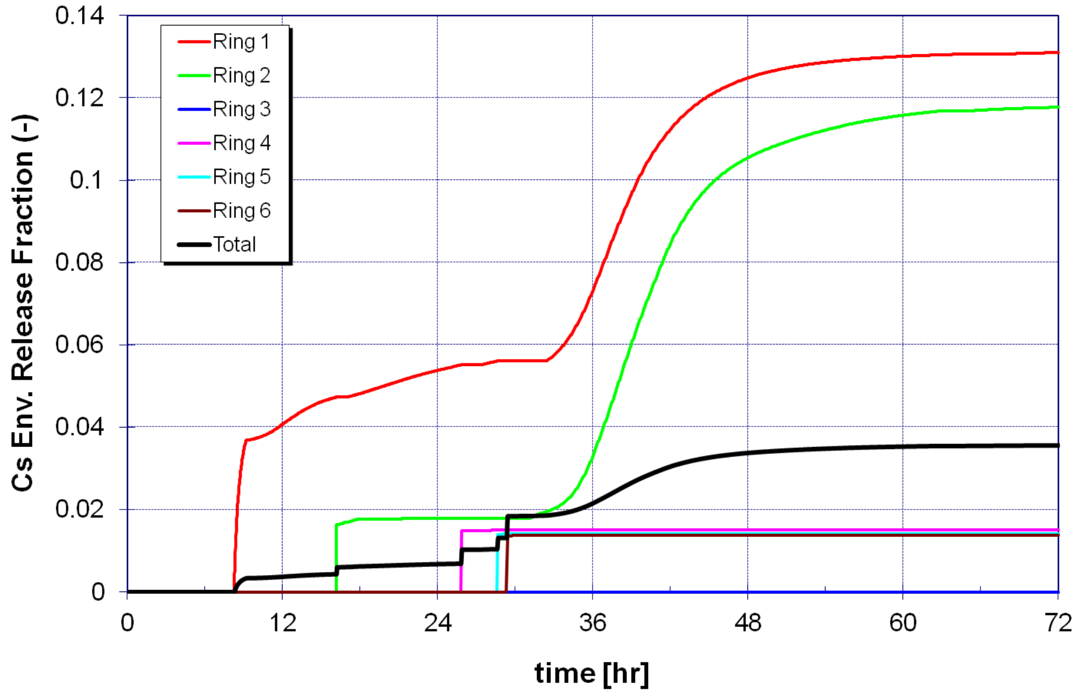


Figure 125 Cesium environmental release fraction for unmitigated uniform high-density moderate leak (OCP1)

Mitigated Moderate Leak (OCP2 Uniform) Scenario

For the mitigated case in the OCP2 uniform pattern that had the highest cesium release fraction (1.2 percent), a number of calculations were performed to determine the effectiveness of mitigation. The same scenario in the 1x4 pattern did not have any release. The overall behavior of fuel temperature is similar to the 1x4 pattern cases in OCP2 (not shown) and OCP 1 (Figure 77), but the fuel is experiencing a higher temperature that gradually decreases. For this base case (Figure 126), temperatures are high enough to cause a gap release and more gradual release of fission products from the fuel. Figure 127 illustrates the calculation for the 200-gpm spray instead of the 500-gpm makeup water, which actually shows a rapid heatup before the temperatures are stabilized.[43] A calculation was performed to test the effectiveness of a higher spray flow rate of 500 gpm and, as indicated in Figure 128, the fuel temperature is stabilized at much lower temperatures without release of fission products from the fuel. In all of the spray calculations performed in this study, the simple flow regime model was disabled because of a more stable and faster calculation, and the previous results from OCP3 had already demonstrated that both models predict comparable maximum clad temperatures.

43 The initial higher temperature spike for the 200 gpm spray, as compared to lower temperatures for the 500 gpm injection case, results from a combination of the leakage versus makeup rate for this particular scenario. For the 500 gpm injection case, the lower portions of the assemblies are covered with water and high decay heat promotes steam cooling of the exposed portions of the fuel. A larger hole size would not have the benefit of steam cooling, and the spray is expected to perform better for a wide range of conditions. Even in this particular case, under quasi-steady conditions the fuel temperatures are generally lower for the spray case.

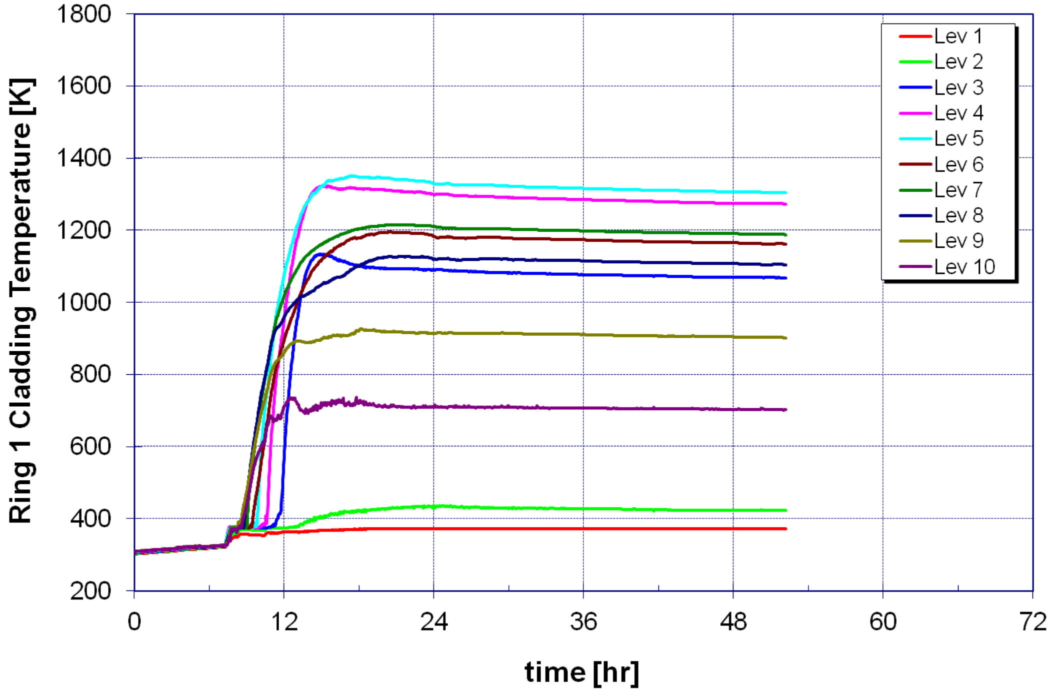


Figure 126 Ring 1 clad temperature for mitigated uniform high-density moderate leak (OCP2) with 500 gpm injection

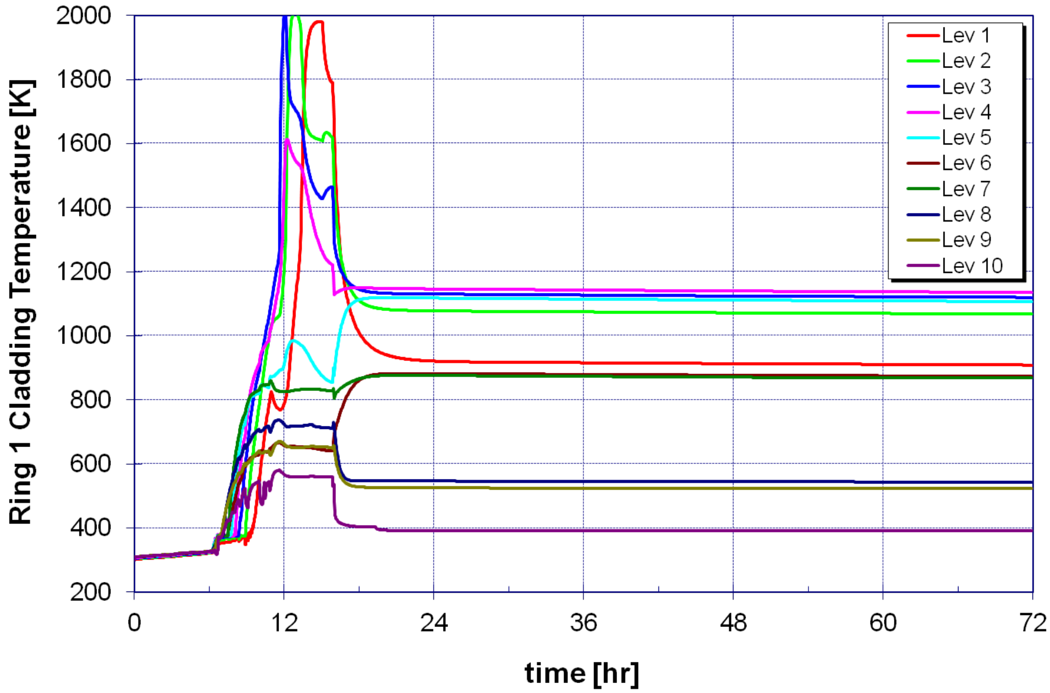


Figure 127 Ring 1 clad temperature for mitigated uniform high-density moderate leak (OCP2) with 200 gpm spray

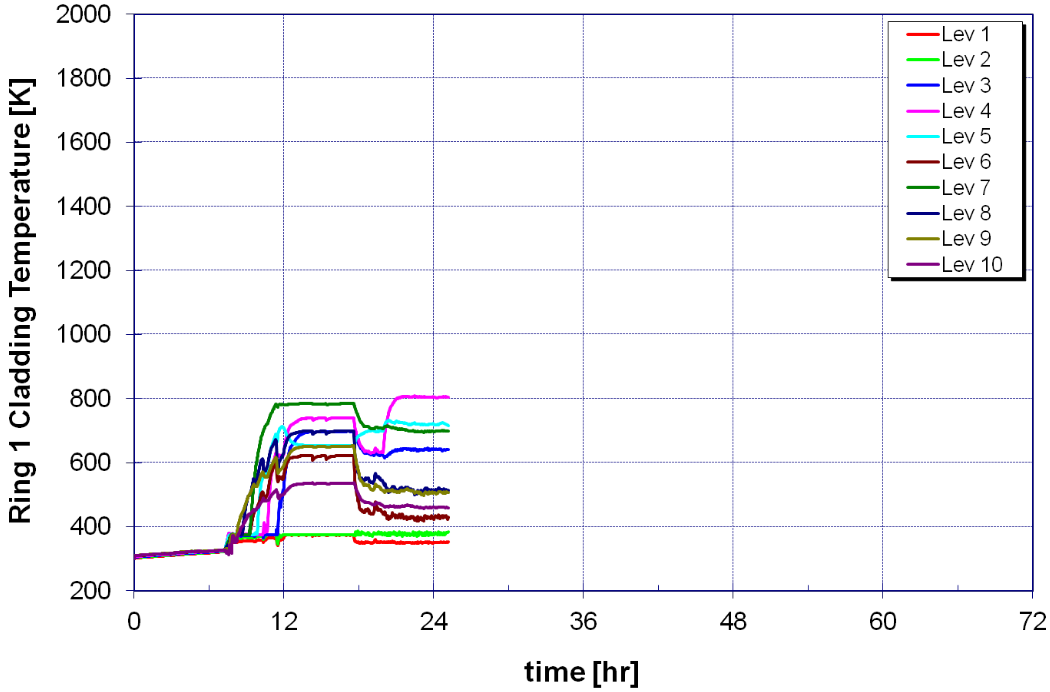


Figure 128 Ring 1 clad temperature for mitigated uniform high density moderate leak (OCP2) with 500 gpm spray

Table 50 Summary of Release Characteristics for High-Density, Uniform Pattern

	Scenario Characteristics					Release Characteristics			
High Density Case #	SFP Leakage?	50.54 (hh)(2) Equip-ment?	Fuel Uncovery (hr)	Gap Release (hr)	Hydro-gen Defla-gration (hr)	Cs release at 72 hours	Cs-137 (MCi) Released	I release at 72 hours	I-131 (MCi) Released
OCP1	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
OCP2	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	21.6	49.1%	22.71	68.4%	7.65

For the offsite consequence analysis, the sequences with recently discharged fuel in a uniform configuration were binned in a similar manner to the low-density and high-density (1x4) loading scenarios. Since the licensee must either preconfigure the SFP to allow direct placement of discharged fuel in or move their recently discharged fuel to a more favorable configuration after a certain amount of time, this sensitivity simply assumes that the high-density uniform case becomes identical to the high-density (1x4) case after OCP2 (i.e., that the actions to meet the requirements on fuel pattern discussed in Section 5.1 are taken at the end of OCP2). While the uniform case has different release categories, the situations that lead to release are largely the same as the low-density and high-density (1x4) base cases. The one exception is for OCP2 with a moderate leak and deployed 10 CFR 50.54(hh)(2) equipment, in which case a successful deployment of mitigation equipment is expected to prevent release for the high-density (1x4) and low-density scenario, but not for the sensitivity scenario of recently discharged fuel in a uniform configuration.

Table 51 Listing of Uniform Pattern Release Sequences

<table>
<tr><td colspan="8">High Density (uniform) Loading</td></tr>
<tr><td colspan="4">Unsuccessful mitigation</td><td colspan="4">Deployed 50.54(hh)(2)</td></tr>
<tr><td colspan="2">Sequence</td><td>Release Frequency (/yr)*</td><td>Release Category</td><td colspan="2">Sequence</td><td>Release Frequency (/yr)</td><td>Release Category</td></tr>
<tr><td rowspan="2">OCP1</td><td>small leak</td><td>6E-09**</td><td>RC12</td><td>OCP1</td><td>mod leak</td><td>6E-09</td><td>RC11</td></tr>
<tr><td>mod leak</td><td>6E-09</td><td>RC23</td><td>OCP2</td><td>mod leak</td><td>2E-08</td><td>RC23</td></tr>
<tr><td rowspan="2">OCP2</td><td>small leak</td><td>2E-08</td><td>RC23</td><td colspan="4" rowspan="4">No Release</td></tr>
<tr><td>mod leak</td><td>2E-08</td><td>RC33</td></tr>
<tr><td rowspan="2">OCP3</td><td>small leak</td><td>4E-08</td><td>RC33</td></tr>
<tr><td>mod leak</td><td>4E-08</td><td>RC11</td></tr>
<tr><td colspan="2">Total</td><td>1E-07</td><td></td><td colspan="2">Total</td><td>2E-08</td><td></td></tr>
</table>

* Release frequency = initiating event frequency * ac power fragility * OCP probability * liner fragility for the specified leak size (see Section 5.6.3 for conditional probabilities)
** Example calculation: 1.7×10^{-5}/yr · 0.84 · 0.0086 · 0.05 = 6×10^{-9}/yr

Table 52 reports the consequence results for the sensitivity scenario of recently discharged fuel in a uniform configuration. It is similar to Table 33 for the base scenarios.

Table 52 Uniform Pattern Consequence Results

SFP Fuel Loading	High Density (uniform)	
Seismic Hazard Frequency[1] (/yr) (PGA of 0.5 to 1.0g)	1.7E-05	
50.54(hh)(2) Mitigation Credited	Yes	No
Conditional[2] Probability of Release	0.14%	0.69%
Hydrogen Combustion Event	"Not Predicted"	"Possible"
Conditional[3] Consequences (Release Frequency-Averaged[4])		
Cumulative Cs-137 Release at 72 hours (MCi)	0.5	11
	Measures Related to Individual Health and Safety	
Individual Early Fatality Risk	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles	7.3E-04[(7)]	6.9E-04
	Measures Related to Cost Benefit Analysis	
Collective Dose (Person-Sv)	1.4E+05	4.9E+05
Land Interdiction[6] (mi^2)	1.1E+03	1.3E+04
Long-term Displaced Individuals[6]	6.2E+05	5.6E+06
Consequences per year (Release Frequency-Weighted[4])		
Release Frequency (/yr)	2.3E-08	1.2E-07
	Measures Related to Individual Health and Safety	
Individual Early Fatality Risk (/yr)	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles (/yr)	1.7E-11	8.1E-11
	Measures Related to Cost Benefit Analysis	
Collective Dose (Person-Sv/yr)	3.1E-03	5.7E-02
Land Interdiction[6] (mi^2/yr)	2.5E-05	1.5E-03
Long-term Displaced Individuals[6] (Persons/yr)	1.4E-02	6.7E-01

[1] Seismic hazard model from USGS (Peterson et al., 2008)
[2] Given specified seismic-event occurs
[3] Given atmospheric release occurs
[4] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions (as applicable); additionally, "release frequency-weighted" results are multiplied by the release frequency.
[5] LNT and population-weighted
[6] 1st year post-accident; calculation uses a dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51
[7] Slightly higher conditional risk with mitigation is due to a more prolonged release (allowing changes in wind direction to affect additional portions of the 10 mile area) and effective protective actions that limit individual risk (regardless of release magnitude); the difference is small compared to the reduction in release frequency.

The insights of the high density 1x4 scenario are also applicable here to the uniform pattern: There is very small likelihood of release. When there is a release, no offsite early fatalities attributable to acute radiation exposure are predicted. On average, significant land contamination is predicted when there is a release with unsuccessfully deployed mitigation. A significant numbers of latent cancer fatalities are also estimated; however, this is a small fraction of cancer fatalities from all causes, because protective actions are expected to keep doses below limits for habitation and ingestion. Overall, individual latent cancer fatality risk is very low, mainly because of the very small likelihood of release and protective actions.

Health effects that would be induced by low dose radiation are uncertain, and insights from a dose truncation for the uniform pattern scenario are similar to those for the high density 1x4 scenario. As can be seen in Table 53, dose truncation significantly lowers the estimated number of total latent cancer fatalities because the uncertain effects of small individual doses are excluded.

Table 53 Dose Truncation Comparison for Uniform Pattern

Dose-Response	High Density (1x4)	
50.54(hh)(2) Mitigation Credited	Yes	No
Conditional[1] Individual Latent Cancer Fatality Risk Within 10 Miles (Release Frequency-Averaged[2])		
Linear, No Threshold	7.3E-04[(3)]	6.9E-04
620 mrem/yr truncation	3.2E-07	1.1E-06
5rem/yr or 10rem lifetime truncation	2.3E-07	1.1E-06
Individual Latent Cancer Fatality Risk Within 10 Miles (/yr) (Release Frequency-Weighted[2])		
Linear, No Threshold	1.7E-11	8.1E-11
620 mrem/yr truncation	7.3E-15	1.3E-13
5 rem/yr or 10 rem lifetime truncation	5.3E-15	1.3E-13

[1] Conditional on a release occurring
[2] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions; additionally, “release frequency-weighted” results are multiplied by the release frequency.
[3] Slightly higher conditional risk with mitigation is due to a more prolonged release (allowing changes in wind direction to affect additional portions of the 10 mile area) and effective protective actions that limit individual risk (regardless of release magnitude); the difference is small compared to the reduction in release frequency.

Similar to the high density (1x4) scenario without deployed 50.54(hh)(2) equipment, the uniform scenario is sometimes predicted to have significant releases when there is a hydrogen combustion. Once again, this is because hydrogen combustion leads to much more zirconium oxidation from the influx of air, as well as a much smaller building decontamination factor. A comparison of this sensitivity analysis of a uniform fuel pattern and those of the base case (i.e. the high density 1x4 pattern and the low density configuration) are quantified in the Table 54 and Table 55.

Table 54 Consequence Comparison – High Density (1x4 and Uniform) Loading Without Successful 50.54(hh)(2) Mitigation

Benefit of High Density (1x4) vs. High Density (uniform) Fuel Loading (Scenario Specific, Weather-Averaged, Release Frequency-Averaged, Unsuccessful Deployment of 50.54(hh)(2))		
Type of Consequence	Consequences** (/yr)	Conditional* Consequences
	Reduction Factor (dimensionless)	
Release Frequency	1.0	-
Individual Latent Cancer Fatality Risk*** for 0-10 Miles	1.6	1.6
Collective Dose (Person-Sv)	1.4	1.4
Land Interdiction (mi^2)	1.4	1.4
Displaced Individuals (Persons)	1.4	1.4

* Conditional on a release occurring (frequency of 1E-7 per year, or lower)
** Release Frequency-Weighted
*** Linear-No Threshold, Population-Weighted

As can be seen in Table 54, without mitigation in the high-density configurations, consequences of the uniform pattern are discernibly higher than the 1x4 pattern. While other contributors could be partially responsible for this difference, this is largely because the accident progression analysis predicts a uniform pattern to sometimes have more detrimental hydrogen combustion events than the 1x4 pattern.

Table 55 compares consequences of high and low density with a uniform pattern for the high density loading, without mitigation. This is similar to Table 37 which uses a 1x4 pattern for the high density loading; however, Table 55 has larger differences because of the larger consequences predicted from a uniform pattern.

Successfully deployed mitigation in the high density configuration lowers the release frequency and most conditional consequences for both uniform and 1x4 patterns. For both patterns, hydrogen combustions are not predicted with MELCOR when 50.54(hh)(2) mitigation is successfully deployed, and therefore the relatively large releases are also not predicted. However, deployed mitigation is not quite as effective in the uniform pattern as it is for the 1x4 pattern. Additionally, deployed mitigation is predicted to be unsuccessful at preventing an additional release in the uniform pattern scenario as compared to the 1x4 pattern. The differences in the release frequencies and conditional consequences can be seen by comparing Table 52 and Table 33.

Table 55 Consequence Comparison – High (Uniform) Density / Low Density Loading Without Successful 50.54(hh)(2) Mitigation

Benefit of High (Uniform) Density / Low Density Loading (Scenario Specific, Weather-Averaged, Release Frequency-Averaged, Unsuccessful Deployment of 50.54(hh)(2))		
Type of Consequence	Consequences** (/yr)	Conditional* Consequences
	Reduction Factor (dimensionless)	
Release Frequency	1.0	-
Individual Latent Cancer Fatality Risk*** for 0-10 Miles	3.4	3.4
Collective Dose (Person-Sv)	18	18
Land Interdiction (mi^2)	78	78
Displaced Individuals (Persons)	70	70

* Conditional on a release occurring (frequency of 1E-7 per year, or lower)
** Release Frequency-Weighted
*** Linear-No Threshold, Population-Weighted

9.4 Sensitivity to Multiunit or Concurrent Accident Events (MELCOR)

These sensitivity calculations are intended to show the importance of the reactor building in the progression of accident in the SFP and the source term from a concurrent reactor accident. In the base calculation for a high-density, moderate leak scenario in OCP3 (see Figure 129), the fuel heats up and a zirconium fire is initiated. The reactor building refueling bay remains intact during the rapid draindown of the pool and there is very low hydrogen generation in the SFP. As the accident progresses, the atmosphere of the reactor building heats up as air is circulated through the assemblies. The oxidation of the SFP fuel depletes the oxygen in the reactor building and limits any long-term air oxidation and the associated exothermic power. Consequently, the long-term fuel heatup is limited primarily by decay heat, and the source term is relatively small (1.7-percent cesium release to the environment). The sensitivity calculations assume failure of the reactor building as a result of the hydrogen combustion caused by leakage from the containment (as evidenced from the SOARCA analysis and the Fukushima accident). The failure of the reactor building is based on the results of the PBAPS short-term SBO calculations for SOARCA (with and without reactor core isolation cooling (RCIC) blackstart). The reactor building failure times are at 8.5 hours (without RCIC blackstart) and 16.9 hours (with RCIC blackstart). It is further assumed that the failure of the reactor building and formation of debris in the pool results in a reduction of flow area at the exit of the assemblies (50 percent of nominal flow area) and increased flow losses.

Figure 130 and Figure 131 show the thermal response of the SFP with early (8.5 hours) and late (16.9 hours) failure of the reactor building. With early failure of the reactor building (before significant fuel heatup), the circulation of the cool air limits the fuel heatup and there is no release from the fuel. With the reactor building intact (Figure 129), the reactor building atmosphere keeps heating up, which limits the convective cooling of the assemblies. With late failure of the reactor building, the fuel becomes hot enough that a sudden increase in the flow of oxygen through the assemblies ignites and rapidly leads to significant air oxidation (zirconium fire). The fuel heats up leading to degradation and finally relocation (see Figure 131). This leads to 60-pecent cesium release to the environment. Finally in OCP4 (see Figure 132 and Figure 133), the decay heat and peak fuel temperatures are lower. The reactor building failure has no impact on the accident progression because the accident is not oxygen-limited. In fact, the reactor building failure (and thus lower temperature of circulating air) leads to lower fuel temperatures.

Table 56 compares the source term for low-density OCP1 for unmitigated small and medium leaks. In both cases, the loss of the reactor building, and thus the effectiveness of natural decontamination, leads to higher release by a factor of 2 to 4 depending on the radionuclide class.

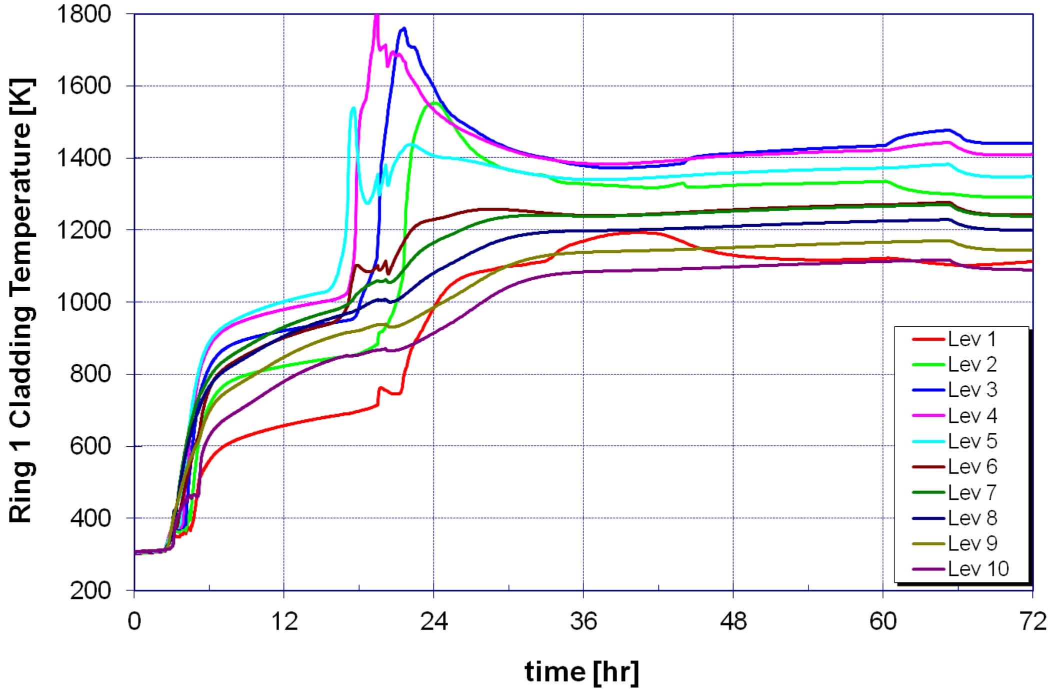


Figure 129 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP3)

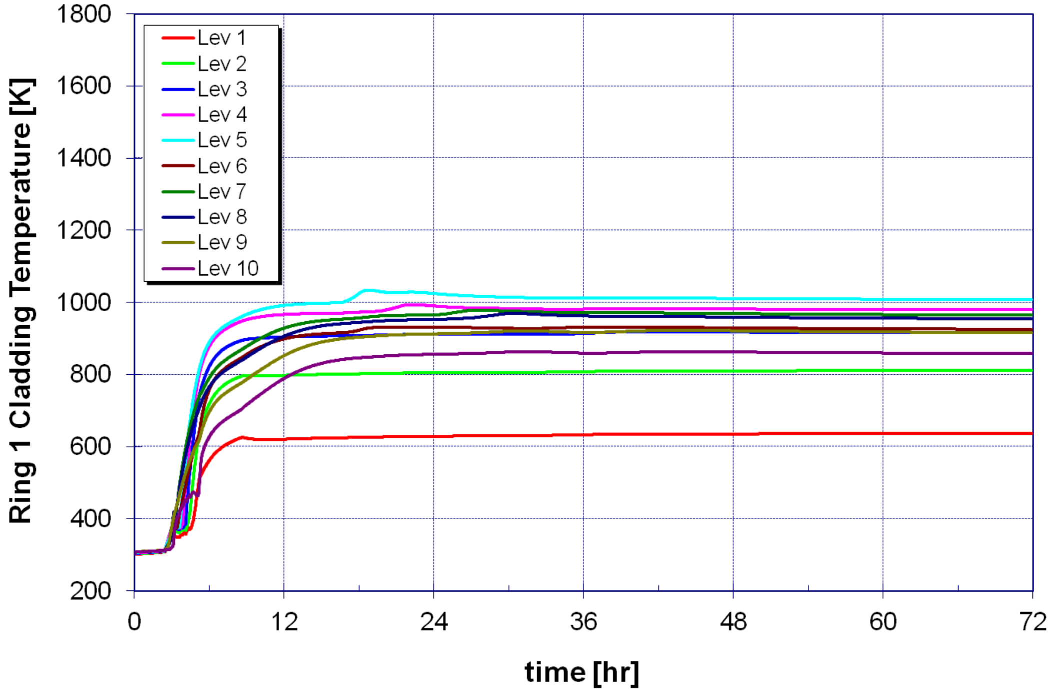


Figure 130 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP3; early reactor building failure)

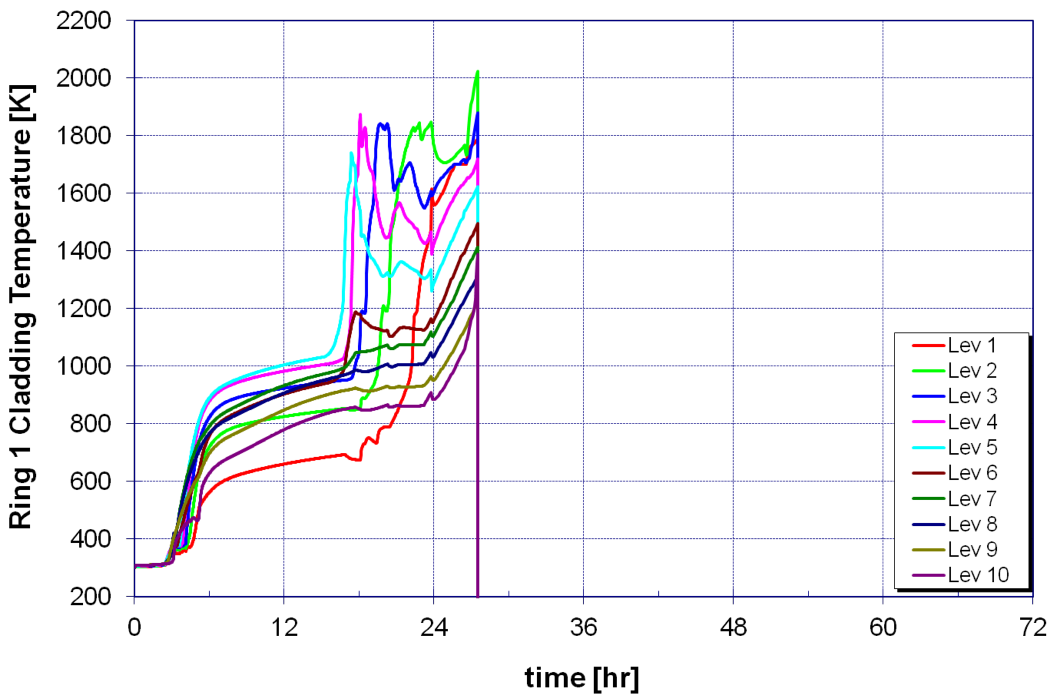


Figure 131 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP3; late reactor building failure)

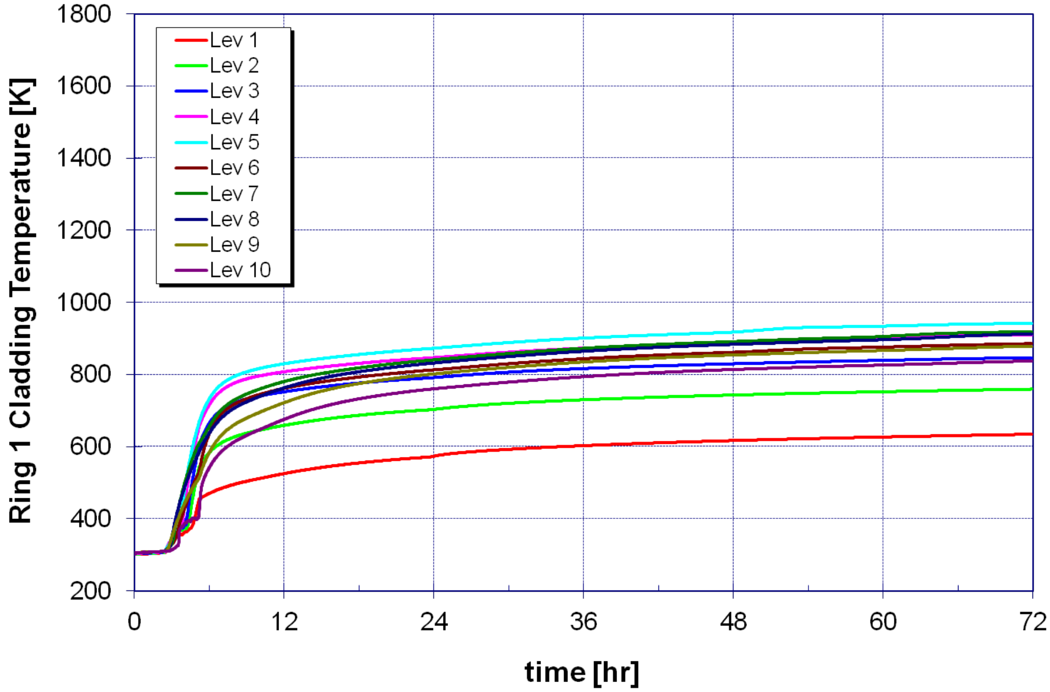


Figure 132 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP4)

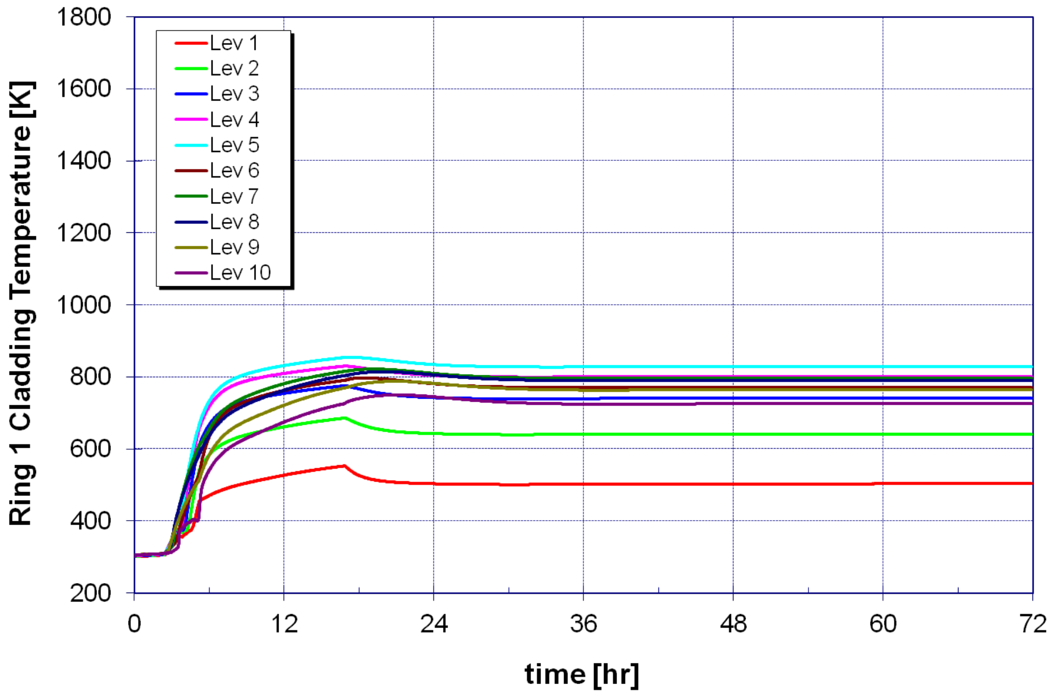


Figure 133 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP4; late reactor building failure)

Table 56 Comparison of Low-Density OCP1 Release Fractions for a Concurrent Reactor Accident

Environmental release fraction	Small Leak		Moderate Leak	
	Base case	Late reactor building failure	Base case	Late reactor building failure
Xe/Kr	1.39E-01	1.85E-01	8.54E-02	8.87E-02
Cs	3.13E-02	1.18E-01	4.58E-03	1.49E-02
Ba	4.39E-03	9.66E-03	1.08E-03	4.45E-03
I	4.55E-02	1.41E-01	1.66E-02	5.60E-02
Te	4.54E-02	1.40E-01	1.68E-02	5.76E-02
Ru	2.17E-05	9.84E-05	2.09E-05	4.93E-05
Mo	8.86E-03	3.51E-02	2.60E-03	6.13E-03
Ce	1.49E-09	6.08E-09	4.94E-10	1.01E-09
La	1.34E-09	5.67E-09	4.37E-10	8.96E-10

9.5 Sensitivity to Molten Core-Concrete Interaction (MELCOR/MACCS2)

Accident Progression Analysis (MELCOR)

This sensitivity is a variation of the previous sensitivity calculation with late reactor building failure caused by a concurrent reactor accident. Even without MCCI, the SFP concrete floor starts to heat up and by the end of 3 days, a portion of the concrete experiences temperatures in excess of its ablation temperature (assumed to be 1500 K). Figure 134 shows the contours of temperature in the SFP floor. In the present sensitivity calculation, MCCI is assumed to be initiated in a control volume that becomes active once the floor liner melts and the debris contacts the concrete. Figure 135 shows the environmental release fraction of cesium. Without MCCI, the releases from the fuel are dominated by diffusion from the fuel matrix grain boundaries as modeled in the CORSOR-Booth model in MELCOR. The MCCI releases are modeled by the VANESA model in MELCOR which takes into account sparging of the concrete decomposition of gases and the presence of metal in the melt.[44] The release fraction of cesium is identical in both calculations (see Figure 16) until MCCI starts in Rings 1 and 2 at about 35 hours. MCCI results in a sudden increase in cesium release (and other fission products) at 35 hours and then again at 40 hours (start of MCCI in Rings 3 and 4) as soon as zirconium is added to the melt interacting with the concrete. In general, the release fractions with MCCI are higher, and for cerium and lanthanum groups, the MCCI releases are orders of magnitude higher.

[44] There are some limitations in representing MCCI in a SFP using MELCOR. The MELCOR MCCI model was developed to represent a pour of core debris from a failed reactor into a confined reactor cavity. In contrast, the relocation of fuel onto the SFP liner could be highly dispersed, especially in a favorable configuration. There could be regions of low-decay heat assemblies surrounding failed high-powered assemblies or open regions under the racks where only the high-powered assemblies relocated to the SFP liner. The MELCOR MCCI model immediately mixes all debris into a uniform debris bed with uniform temperature and decay heat power. Nevertheless, the MCCI sensitivity calculations illustrate the potential impact of MCCI physics on the radionuclide chemical form (and volatility) and the associated release of radionuclides to the environment. Certain radionuclide species can become more volatile in the presences of sparging ablation gases, which leads to the differences in Table 57.

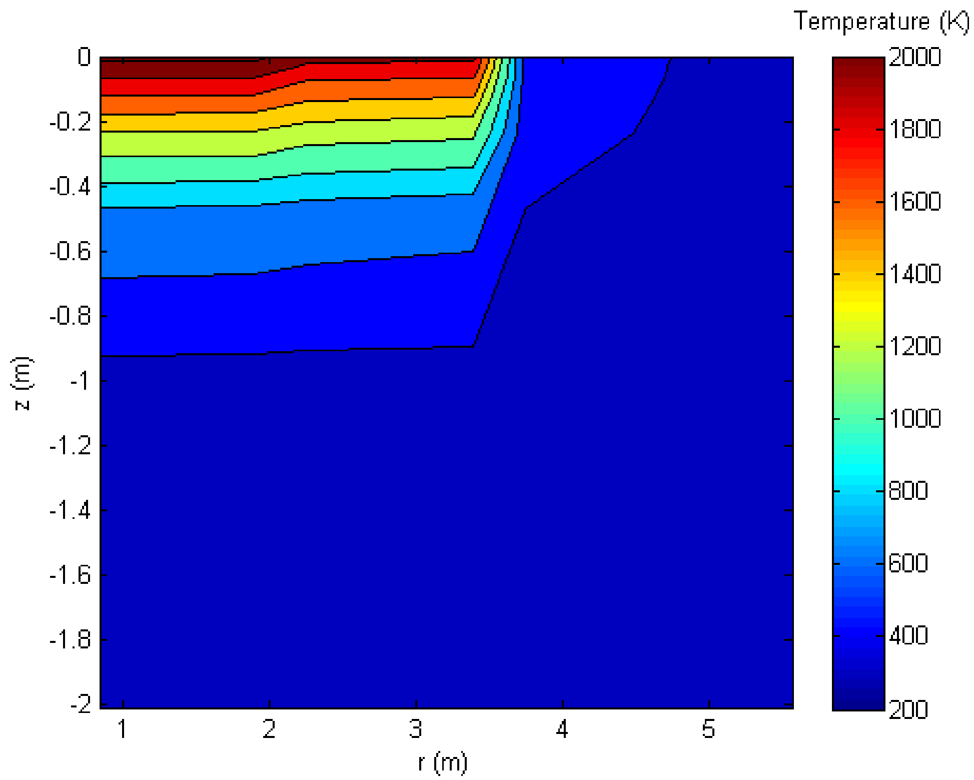


Figure 134 SFP concrete floor temperature for unmitigated high-density moderate leak (OCP3; late reactor building failure)

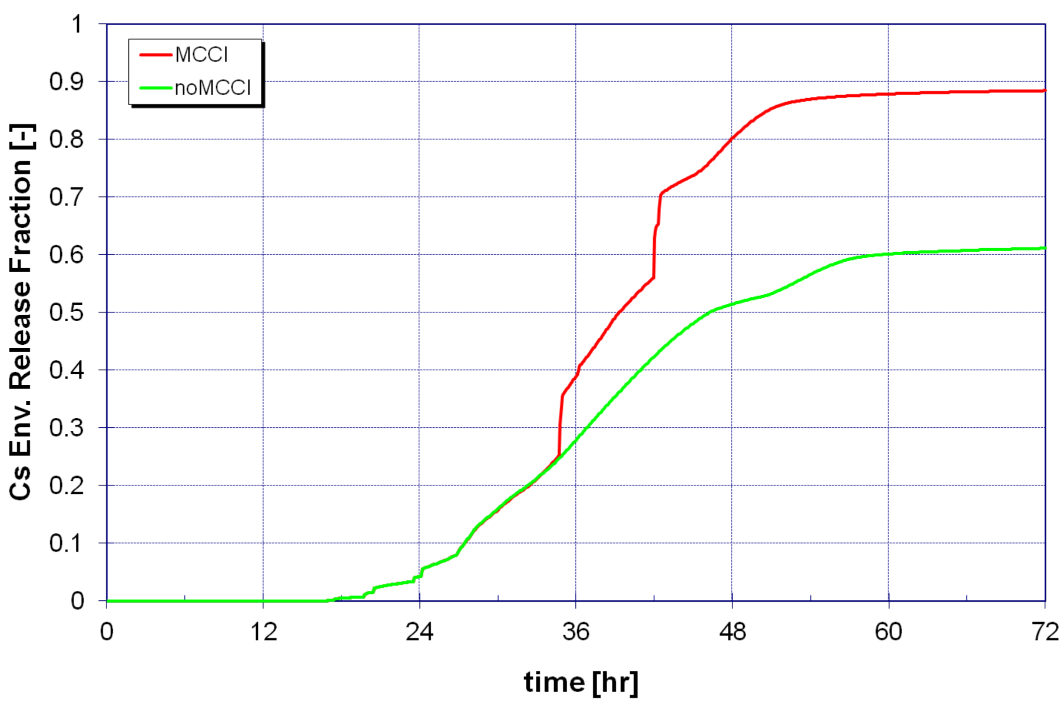


Figure 135 Cesium release fraction for unmitigated high-density moderate leak (OCP3; late reactor building failure) with and without MCCI

Table 57 Comparison of Release Fractions with and without MCCI.

Environmental release fraction	Without MCCI	With MCCI
Xe/Kr	0.92	0.92
Cs	0.61	0.88
Ba	0.01	0.07
I	0.83	0.91
Te	0.80	0.74
Ru	0.01	0.003
Mo	0.15	0.11
Ce	1.7E-07	0.007
La	1.6E-07	0.0002

Offsite Consequence Analysis (MACCS2)

The sequence used in the accident progression analysis was analyzed with MACCS2 to understand how MCCI affects offsite consequences. The sequence analyzed was the OCP3 moderate leak scenario, with hydrogen combustion in the refueling bay at 16.9 hours as predicted from the SOARCA short-term SBO with RCIC blackstart scenario.

The focus of this study was specifically on the SFP, and therefore, this sequence is not part of the main results. Rather, these are part of a different sensitivity investigating the effects of concurrent reactor events. Therefore both sequences with and without MCCI were calculated with MACCS2 in order to focus on MCCI. The individual consequence results of these sequences are not reported; however the effect of MCCI on the offsite results is shown in Table 58.

Table 58 Consequence Comparison – Molten Core Concrete Interaction

Molten Core Concrete Interaction Sensitivity (Weather-Averaged; OCP3 Moderate Leak sequence with reactor building failure at 16.9 hours)	
Type of Consequence	Conditional* Consequences
	Percent Increase
Individual Latent Cancer Fatality Risk** for 0-10 Miles	-17%
Collective Dose	9%
Land Interdiction (mi^2)	33%
Displaced Individuals (Persons)	17%

* Conditional on a release occurring (frequency of 1E-7 per year, or lower)
** Linear-No Threshold, Population-Weighted

No early fatalities are predicted for these sequences. These sequences have considerable release fractions, including increased contributions from some of the typically non-volatile chemical groups. However, these characteristics were not enough to reach the dose thresholds associated with early fatalities in OCP3, for which the last fuel offload has cooled for 37 days since shutdown.

Although the release is larger with MCCI, the individual LCF risk for 0-10 miles non-intuitively decreases. The reason for this is likely due to the significant level of protective actions and the type of radionuclides in the different source terms. The radionuclides in the MCCI sensitivity likely have different dose contributions relative to their LCF risk contribution, and therefore are more likely to cause protective actions despite having relatively lower risk factors, which in turn causes lower LCF risk.

To verify this phenomenon, the risk and dose contributions of different radionuclides to offsite consequence could be investigated. This is not done here; however, when farther distances are included (which are areas where protective actions and this phenomenon are less likely to occur), this reduction in the results no longer exists. This can be seen in the effect of MCCI on the collective dose in Table 58. Similarly, land interdiction and displaced individuals have higher consequences, as one may expect with relatively larger source terms.

9.6 Sensitivity to Radiative Heat Transfer (MELCOR)

For this sensitivity calculation, the surface area between Rings 2 and 4, Rings 4 and 6, and Rings 6 and 7 are modified by plus or minus 25 percent. Although Rings 2, 4 and 6 are empty, they still contain rack components and can impact the heatup in Rings 1, 3, and 5. Table 59 reports the results of the calculation for the unmitigated small leak (OCP2), low-density configuration. In general, the highest differences are observed for the reduction in the area (approximately 30-percent reduction in the release of volatiles), while increasing the surface area only has a modest effect on the release. The fuel heatup, radiation between rack components, initiation of air cooling, and interaction between different assemblies and racks are all complex phenomena that contribute to fission product release in a nonlinear fashion. Nevertheless, the releases for the low-density case are still small, and uncertainties in radiation modeling do not seem to significantly change the results

Table 59 Low-density OCP2 Release Fraction Sensitivity to Ring-Ring Radiation

	Base Case	-25% surface area	+25% surface area
Xe/Kr	4.41E-02	4.44E-02	4.40E-02
Cs	1.71E-02	1.23E-02	1.66E-02
Ba	5.19E-03	3.69E-03	5.06E-03
I	3.31E-02	2.50E-02	3.22E-02
Te	3.54E-02	2.53E-02	3.45E-02
Ru	9.27E-07	9.27E-07	8.75E-07
Mo	9.95E-05	9.93E-05	9.39E-05
Ce	3.54E-11	4.03E-11	3.16E-11
La	3.58E-11	4.06E-11	3.19E-11

9.7 Sensitivity to Land Contamination (MACCS2)

The measure of contaminated land area can vary significantly with the criterion used to measure or estimate the level of contamination. This study calculates the land that exceeds 500 mrem in the first year after the accident as an indicator for land contamination, based on the Pennsylvania 500 mrem annual dose limit for habitability. However, other protective action levels exist that can also be used as an indicator for measuring land contamination. These protective actions tend to be related to dose levels associated with either land interdiction or decontamination, but not necessarily.

Radioactivity levels are not typically used as a basis for protective actions. Instead, activity levels are usually measurements for estimating different dose levels, which are in turn used as the basis for protective actions. A range of typical activity levels for Cs-137 is included in the table below. These particular levels have been widely reported as the zoning criteria for the Chernobyl nuclear disaster.

The EPA intermediate phase PAG levels are 2 rem in the first year, and 500 mrem annually thereafter. Previous studies have typically used one criterion (4 rem in 5 years) to represent these PAG levels. How well this represents the actual EPA PAG levels was not analyzed here, although this criterion is included here.

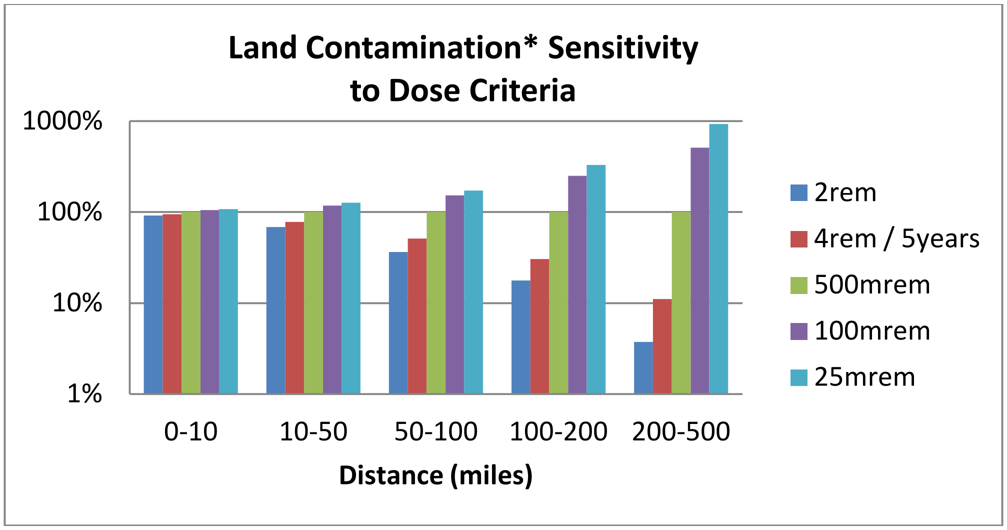
For simplicity, this sensitivity was based on the weather-average results of a single accident sequence. This is unlike the results of the production analyses, which are frequency-weighted averages of all the release sequences. The sequence chosen was the OCP3 small leak from a high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment, which has a cesium release fraction of 42% at 72 hours. Since the consequence results of individual sequences are not reported, the results of this sensitivity have been normalized.

Table 60 Consequence Comparison – Land Contamination Sensitivity

Total Land Area Sensitivity to Dose/Activity Criteria (Weather-Averaged; OCP3 Small Leak from high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment)		
Dose		
Protective Action Basis	Dose Level[1]	Land Area
EPA intermediate phase PAG[2]: 1st year	2 rem	21%
EPA intermediate phase PAGs[2] (as commonly represented in previous studies)	4 rem / 5 years	32%
Pennsylvania dose limit to the public	500 mrem	100%
ICRP recommendation[3]	100 mrem	361%
10CFR Part20 Subpart D		
10CFR Part20 Subpart E[4]	25 mrem	731%
Activity		
Protective Action Basis	Activity Level (Cs-137 Bq/m2)	Land Area
-	1.48E+06	73%
-	5.55E+05	181%
-	1.85E+05	346%
-	3.70E+04	557%

[1]Annual doses, unless otherwise noted
[2] EPA intermediate phase PAGs are: 2 rem in the first year, and 500mrem annually thereafter.
[3] ICRP recommends using the lower portion of a band that spans 1-20 mSv as a reference level for protective measures, and past experience demonstrates 1mSv is typical.
[4] 10CFR Part20 Subpart E also includes ALARA, which is not considered here.

As seen from the table above, different dose or activity levels can significantly change the amount of land area that exceeds a given limit. In addition to the total land area, a range of different distances were also analyzed in the graph below.



*Weather-averaged; OCP3 Small Leak from high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment

Figure 136 Land Contamination Sensitivity to Dose Criteria

At shorter distances, the change in the land area that exceeds a given dose limit is not significant, while at far distances, the change can be more than a factor of 10. The distance where the amount of land contamination becomes sensitive to different dose criteria is expected to depend on the initial concentration and the deposition rate. For this release magnitude, most of the plume exceeds all of the dose criteria at close distances. However, irrespective of the release magnitude, the affected area will increase and the concentrations will decrease as the plume spreads. Therefore, for all releases, land contamination is more dependent on the dose criteria at far distances.

9.8 Sensitivity to Time Truncation (MELCOR/MACCS2)

Project staff judged that a reasonable approach for the project is to consider radionuclide releases only if the fuel has become uncovered by 48 hours and to assume that any potential radiological release is stopped at 72 hours (Section 5.3). However, the use of a time truncation is uncertain, and is capable of significantly affecting the consequences. This assumption could be pessimistic since many resources are available at the State, regional, and national level that could be available to potentially truncate the accident more aggressively. On the other hand, this time truncation could be optimistic, as it assumes that an ongoing spent fuel pool release is capable of being truncated.

Given the uncertainty, this sensitivity considers the effects of both a more aggressive and a less aggressive time truncation. For simplicity, the sensitivity of longer time truncation was based on the weather-average results of a single accident sequence. This is unlike the results of the production analyses, which are frequency-weighted averages of all the release sequences. The sequence chosen was the OCP3 small leak from a high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment, which has a cesium release fraction of 42% at 72 hours. Since the consequence results of individual sequences are not reported, the results of this sensitivity are reported as the fractional increase in consequences over the original results. The sensitivity of a shorter time truncation discusses in which sequences releases would be averted.

A period of 96 hours was chosen to represent a less aggressive time truncation. MELCOR and MACCS2 calculations were extended to 96 hours from the original 72 hours. The effect on the release fractions and the relative effect on offsite consequence can also be seen in the Figure 137 and Table 61 below.

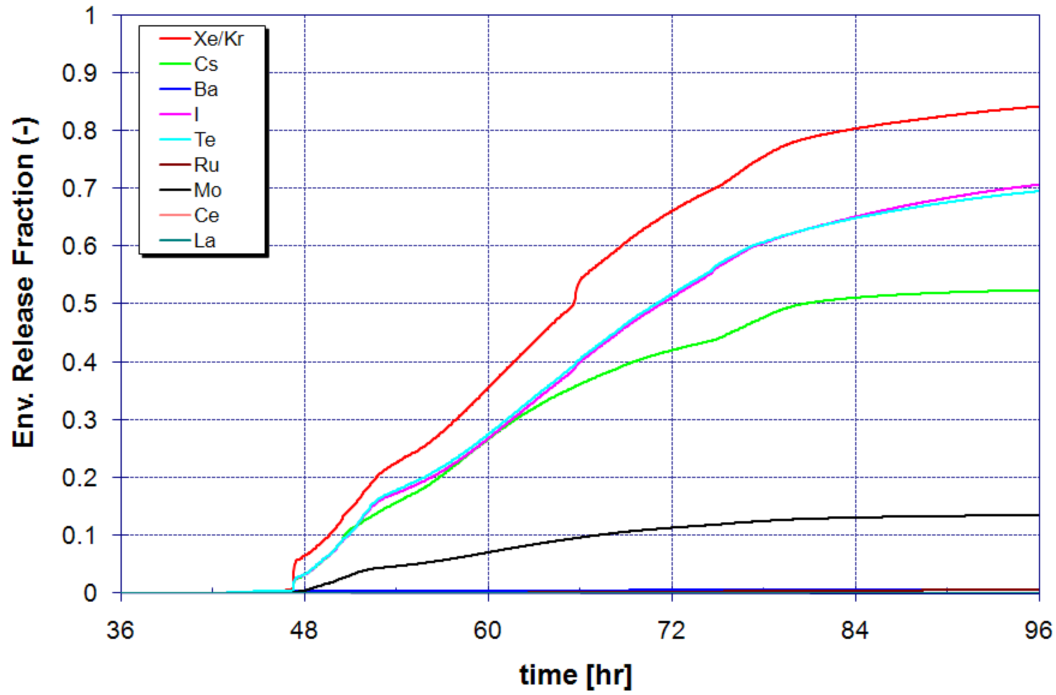


Figure 137 Atmospheric release fractions for unmitigated high density small leak (OCP3) with a 96 hour time truncation

Table 61 Consequence Comparison – Time Truncation Sensitivity

Time Truncation Sensitivity: 72 Hour vs. 96 Hour (Weather-Averaged; OCP3 Small Leak from high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment)	
Type of Consequence	Conditional* Consequences
	Percent Increase
Individual Latent Cancer Fatality Risk** for 0-10 Miles	38%
Collective Dose (Person-Sv)	27%
Land Interdiction (mi^2)	28%
Displaced Individuals (Persons)	27%

* Conditional on a release occurring (frequency of 1E-7 per year, or lower)
** Linear-No Threshold, Population-Weighted

A longer time truncation, however, can also significantly affect the results. For the sequences involving a small leak from a high density SFP during OCP1 and OCP2 with unsuccessful deployment of 50.54(hh)(2) equipment, fuel uncovery occurs around 40 and 43 hours (compared to a baseline time truncation of 48 hours). Using a time truncation less than 40 and 43 hours respectively, would avoid releases for these sequences. In other scenarios, fuel uncovery and release occur much sooner than the baseline time truncation.

These results highlight that some releases are expected to be prolonged and therefore a choice in a time truncation can affect offsite consequence predictions.

9.9 Sensitivity to Reactor Building Leakage (MELCOR)

Four sensitivity calculations were performed to examine the impact of the reactor building leakage on hydrogen combustion and accident progression. These covered the small leak scenarios in OCP2 and OCP3 without successful deployment of mitigation. Two larger leak sizes were considered, (1) an increase in the nominal leakage area by a factor of 10, and (2) an increase in the nominal leakage area corresponding to area of a blowout panel. In general, while an increase in area by a factor of 10 increases the leakage, any further increase in area has no effect since the building pressure adjusts to limit the leakage. The leakage area has no significant impact on accident progression, and since the hydrogen is produced over a relatively short time, the hydrogen mole fraction quickly reaches the 10% threshold for ignition. The Cs release fractions are not significantly altered. In OCP2, Cs release fraction is reduced by ~12% while it is increased by ~2% in OCP3 owing to slight variations in the course of the accident.

10. ASSESSMENT OF PREVIOUS STUDIES OF SAFETY CONSEQUENCES ASSOCIATED WITH LOADING, TRANSFER, AND LONG-TERM DRY STORAGE

10.1 Introduction

Staff has performed an assessment to 1) identify previous studies of safety consequences of spent fuel accidents in both wet and dry storage, 2) determine the extent to which those previous studies are comparable to results from the SFPS, and 3) to the extent practicable, update the results of the previous studies to facilitate a comparative assessment. The SFPS discusses off-site consequences of a spent fuel pool accident in Chapter 7, and provides limited discussion of several similar previous spent fuel pool studies. This Chapter provides a more detailed comparison between off-site consequences calculated for the SFPS and those calculated from previous studies. This Chapter also provides a comparative assessment of SFPS results against previous studies of the safety consequences associated with loading, transfer, and long-term storage in dry cask storage systems (DCSS). In these assessments, staff limited its focus to offsite consequences of accidental releases at commercial nuclear power plants. Specifically, these assessments compare the direct impacts due to offsite radiological exposure and the indirect (e.g., economic or land use) impacts of protective measures taken to avert offsite radiological exposure, of the various studies considered. Offsite impacts from routine operations, doses to workers from routine or accidental exposures, or non-safety related impacts such as costs of spent fuel management, were not considered. Furthermore, staff focused on studies associated with accidents, rather than studies of safety consequences associated with deliberate human actions such as sabotage or terrorism.

10.2 Previous Spent Fuel Pool Studies

There have been several previous studies of the consequences of spent fuel pool accidents. These include those in support of Generic Safety Issue 82 and of consequences from spent fuel pool accidents at shutdown nuclear power plants:

- "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82" (NUREG/CR-4982, 1987)
- "Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools," (NUREG/CR-5281, 1989)
- "Regulatory Analysis for the Resolution of Generic Issue 82 'Beyond Design Basis Accidents in Spent Fuel Pools'" (NUREG-1353, 1989)
- “A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants” (NUREG/CR-6451, 1997)
- “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (NUREG-1738, 2001)

The studies conducted to evaluate beyond design basis accidents in spent fuel pools in the late 1980’s (NUREG/CR-4982, NUREG/CR-5281, and NUREG-1353) report a variety of impacts related to both radiological doses (e.g., collective doses), as well as the potential impacts associated with limiting radiological doses, such as costs and extent of land condemnation. NUREG-1353 reports collective doses within a 50 mile radius of 8 to 26 million person-rem per event based on MACCS calculations documented in NUREG/CR-5281. It also reports an interdiction area (“*area with such a high level of radiation that it is assumed that it cannot be decontaminated”*), based on CRAC2 calculations from NUREG/CR-4982, of 0 to 244 square miles (within a 50 mile radius) and offsite property damages of $3 to $26 billion in 1983 dollars (also within a 50 mile radius), based on MACCS calculations documented in NUREG/CR-5281. The more recent studies conducted to examine the risks from spent fuel pool accidents at shutdown nuclear power plants (NUREG/CR-6451 and NUREG-1738) also report a variety of both radiological and non-radiological impacts. NUREG-1738 reports radiological impacts of 0 to 200 early fatalities and potential latent cancer fatalities (out to 500 miles) in the hundreds of thousands. These studies used a variety of assumptions regarding pool inventory, release fraction, population density, and emergency response. The results of previous spent fuel pool studies are compared to the SFPS results for various consequence metrics in Table 62.

10.2.1 Quantitative Comparison of Spent Fuel Pool Analytical Results

The following table presents selected consequence results from previous studies of spent fuel pool accidents, and the SFPS.

Table 62 Comparison of consequence results from current and previous spent fuel pool analyses

<table>
<tr><th>Metric</th><th>NUREG/CR-4982, Table 4.7</th><th>NUREG/ CR-5281, Table 3.2</th><th>NUREG/ CR-6451, Tables 4.1/4.2</th><th>NUREG-1738[1] Tables 3.7-1/3.7-2</th><th>SFPS Results[2,3]</th></tr>
<tr><td>Early fatalities (0 to 500 miles)</td><td>Not reported</td><td>Not reported</td><td>0 to 101</td><td>0 to 200</td><td>0</td></tr>
<tr><td>Individual LCF risk within 10 miles (conditional)</td><td>Not reported</td><td>Not reported</td><td>Not reported</td><td>7.7e-4 to 8.2e-2</td><td>2.0e-4 to 4.4e-4</td></tr>
<tr><td>Collective dose within 50 miles in Person-Sv</td><td>11,000 to 26,000[6]</td><td rowspan="2">80,000 to 256,000</td><td>30,000 to 810,000</td><td>37,000 to 240,000</td><td>7,400 to 39,000</td></tr>
<tr><td>Collective dose within 500 miles in Person-Sv</td><td>710,000</td><td>40,000 to 3,400,000</td><td>450,000 to 600,000</td><td>27,000 to 350,000</td></tr>
<tr><td>Interdicted land (square miles)</td><td>Not reported</td><td>Not reported</td><td>Not reported</td><td>Not reported</td><td>170 to 9,400</td></tr>
<tr><td>Condemned land (square miles)</td><td>4 to 224[4,6,7]</td><td>Not reported</td><td>1 to 2,800</td><td>Not reported</td><td><1 to 83</td></tr>
</table>

1. Results presented in Section 3.7 are taken from there; otherwise values are from Appendix 4, Note that the upper end of these values is generally driven by high ruthenium source terms with late evacuation.
2. 0 to 500 mi results are actually 0 to 1000 mi results, which is likely analogous to past study modeling assumptions; uniform pattern results are not included at this time; only LNT results are presented
3. The range of results is not bounding, as it does not represent ranges due to many uncertainties such as weather, operating cycle phase, or pool damage states. Direct consideration of these uncertainties would increase the range, as it likely would for the previous studies as well.
4. Note that the definition of interdicted land is not consistent with the definition used in the SFPS report. The text in NUREG/CR-4982 clarifies that what is reported is permanently uninhabitable land, which is analogous to condemned land.
5. These values use the annualized release frequency, combined with the conditional consequences, thereby over-estimating the average risk.
6. This range is for fire scenarios. For the non-fire scenario, the values were 4 person-rem and 0.0 sq. mi interdiction area.
7. Note that this metric does not change between cases 1A (50 miles) and 1C (500 miles), indicated that there is no additional condemned land beyond 50 miles in this analysis
8. The range at which this metric computed is not specified in NUREG/CR-5281

10.2.2 Comparison of SFPS Results to previous Spent Fuel Pool Studies

Comparison of SFPS results to past spent fuel pool studies is not straight-forward, because those studies reported a variety of consequence metrics and used a range of assumptions regarding pool inventory, release fraction, population density, and emergency response. These ranges present a variety of approaches to represent uncertainties from select input parameters, depending on the study. For instance, the range of NUREG-1738 results represents a range due to evacuation times, ruthenium release modeling, time since reactor shutdown, and two competing seismic hazard models. The range of the SFPS, on the other hand, represents a range due to uncertainty in deployment of mitigation equipment and variations of potential pool loading density. In addition, the SFPS results are expected to be sensitive to uncertainties in hydrogen combustion ignition criteria and the time truncation value (and these uncertainties are not reflected in the range of results), as well as uncertainties in weather, decay power, and pool damage states (which are not explicit in the range of results since average results are presented). It is also important to remember that past studies generally used generic assumptions intended to envelope the situation, as opposed to the focus on site-specificity with the SFPS. Nevertheless, these ranges of consequence metrics are often cited by external stakeholders, and thus comparison is informative.

A comparison of the release characteristics from previous spent fuel pool studies demonstrates that releases of cesium are generally less in the current study than in previous studies, and the time from accident initiation to release to the offsite environment is generally longer:

Table 63 Comparison of Source Terms from Current and Previous SFP analyses

Resolution of GI-82 (NUREG-1353, NUREG/CR-4982, NUREG/CR-5281):	NUREG-1738	SFPS (preliminary results):
• 10 to 100% Cs release (100% assumed for cases 1 and 2) • Release over 8 hours for a propagating SFP zirc fire (assumed) • 0.25 (BWR) or 1.0 (PWR) conditional probability if fuel becomes uncovered	• 75% Cs release (assumed from NUREG-1465) • Instantaneous draindown for large seismic • 2 to 14 hour heatup depending on fuel age (see Table 1A-1)	• Cs release = < 1% to 49% • Draindown to uncovery – 2.5 to 43 hours (when leak exists) • Start of release = 8 hours to > 72 hours

The lack of any early fatalities attributable to acute radiation exposure in this study is consistent with results of some past SFP studies, and much lower than others (e.g., up to 200 early fatalities from NUREG-1738). The range of latent fatalities predicted in this study is consistent with the lower end of the range reported in past SFP studies. The conditional individual latent cancer fatality risk from 0 to 10 miles for the scenarios studied in this report is several orders of magnitude below that reported in NUREG-1738, which was the only other study to report this metric. Even when the early evacuation scenario from NUREG-1738 is used for comparison (average individual risk is in the range of 2.6E-3 to 4.8E-3), the results from the current SFPS study are significantly lower. The collective dose values predicted in this study are consistent with the lower end of the range reported in past SFP studies. The SFPS reports temporarily interdicted land (uninhabitable land during the first year following the postulated accident), in order to remove uncertainty in longer-term effects and policies related to weathering and decontamination decisions. Reporting interdicted land makes the results incomparable to the past SFP studies which have presented condemned land. The SFPS does not report other aspects of offsite property damage.

10.3 Previous Dry Cask Storage Studies

The number of studies of the consequences from dry cask handling and storage accidents are more limited than those for spent fuel pools. Safety analysis reports for dry cask storage systems, submitted in support of applications or renewals for site-specific independent spent fuel storage installation (ISFSI) licenses or for DCSS certificates, include some information on offsite consequences of potential accidents (e.g., tornado missile impacts, earthquakes, floods). However, such accidents are generally shown by analysis not to result in a release, and the likelihood of more severe accidents is sufficiently low that the consequences need not be explicitly evaluated. Staff identified one previous NRC analysis on the offsite safety consequences of accidents from dry cask storage systems. The report, "A Pilot Probabilistic Risk Assessment of a Dry Storage System at a Nuclear Power Plant" (NUREG-1864, ML071340012), documents a pilot PRA for a specific dry cask system (Holtec International HI-STORM 100) at a specific boiling-water reactor (BWR) site. The study included an assessment of potential offsite consequences from the drop and failure of a cask. It provides estimates of the annual risk for one cask in terms of the individual probability of a latent cancer fatality within 16 km (10 miles) of the site, and also reports that there are no prompt fatalities. The assessment was performed using MACCS2 for a representative site and is described in detail in Appendix E to NUREG-1864. Site-specific data important to modeling a HI-STORM dry cask 30.5 meter (100-foot) drop accident scenario in the MACCS2 consequence calculation were collected and used. The important parameters/variables required to model the site are the population density/distribution and the site meteorology. The radionuclide inventory, source term (i.e., release fraction, release start time, and release duration), initial plume dimensions (related to the system geometry), and plume heat content were described. Other settings and models necessary for a MACCS2 calculation (e.g., food chain model) were taken from the NUREG-1150 study MACCS2 input file prepared for the Surry Power Station. The input file is documented in Appendix C to the MACCS2 code manual and is referred to there as Sample Problem-A.

10.3.1 Supplemental Analyses

In order to provide quantitative estimates of safety consequences for accidents during dry cask handling and storage that are directly comparable with the results of the SFPS, and to provide additional output metrics for comparison, staff performed limited MACCS2 supplemental dry cask storage analyses. These supplemental analyses used the source term characteristics from NUREG-1864 coupled with the site-specific characteristics reflected in the MACCS input decks used in the SFPS analyses. The analyses conducted in NUREG-1864 were conducted at a different geographic location than the site selected for the SFPS and evaluated impacts only in terms of selected human health metrics (the individual probability of a prompt fatality within 1.6 km (1 mile) and of latent cancer fatality within 10 miles, and the individual lifetime dose commitment). These metrics can be affected by site-specific characteristics such as meteorology and population distributions surrounding the site. To perform this analysis, staff modified the MACCS input files used for the analyses in the SFPS (described in detail in Chapter 7 of the SFPS) with the revisions discussed below.

<u>Changes related to meteorology, site characteristics, and dosimetry:</u>
No changes were made to the SFPS input deck related to meteorology, site demographic and economic characteristics, or dosimetry. Site data, including weather, population, and land values are therefore consistent with SFPS results. The dosimetry files used are consistent with FGR-13, whereas NUREG-1864 used the dose conversion factors used in NUREG-1150. This is a potential source of difference from the results reported in NUREG-1864.

<u>Changes related to source term and release:</u>

The radiological inventory was changed to be consistent with Table E.1 of NUREG-1864. In addition, a limited set of radionuclides present in the SFPS input deck that are expected to be in secular equilibrium with the nuclides listed in Table E.1 (Ba-137m, Pr-144, and Rh-106) were added with an activity equal to that of their parent radionuclide. However, a limited set of nuclides (Pm-147, Eu-154, Am-242m, Am-243, and Cm-243) reported in NUREG-1864 Table E.1 were not used in the SFPS MACCS2 input deck. Because the dosimetric data for these nuclides was not developed in the SFPS input deck, these radionuclides were not included in the modeled inventory for the supplemental analysis. Based on the much larger inventory of fission products such as Cs-137 and Sr-90, and of actinides such as Pu-241, the omission of these nuclides is not expected to significantly affect the results; however, this is a potential source of difference from the results reported in NUREG-1864. The number of chemical groups was changed to three to represent noble gases (NG), activation products (CRUD) and particulates (PART) to be consistent with the NUREG-1864 source term. Consistent with the NUREG-1864 source term, the only nuclide in the noble gas chemical group was Kr-85, and the only nuclide in the activation product chemical group was Co-60. For consistency with NUREG-1864, all other nuclides were assigned to the particulate group in view of the fact that releases from dry casks are likely to result from impacts at a sufficiently low temperature that radionuclides would be released by mechanical means rather than because of different volatilities. The inventory modeled in this supplemental analysis is provided below:

Table 64 Modeled Inventory for Supplemental Reanalysis

Nuclide	Bq	Chemical Group
Co-60	1.61E+14	CRUD
Kr-85	2.77E+15	NG
Sr-90	3.40E+16	PART
Y-90	3.40E+16	PART
Ru-106	2.92E+14	PART
Rh-106	2.92E+14	PART
Cs-134	5.13E+15	PART
Cs-137	5.54E+16	PART
Ba-137m	5.54E+16**	PART
Ce-144	5.08E+13	PART
Pr-144	5.08E+13**	PART
Pm-147	0* (*3.37E+15*)	PART
Eu-154	0* (*4.15E+15*)	PART
Pu-238	3.98E+15	PART
Pu-239	1.87E+14	PART
Pu-240	3.47E+14	PART
Pu-241	5.23E+16	PART

Am-241	1.20E+15	PART
Am-242m	0* (*1.97E+13*)	PART
Am-243	0* (*3.07E+13*)	PART
Cm-243	0* (*3.02E+13*)	PART
Cm-244	5.66E+15	PART

*These nuclides were not included in the supplemental analysis, as discussed above. The values from NUREG-1864 are provided in parentheses to allow comparison of source terms.
**These short-lived progeny were not in Table E.1 of NUREG-1864 but are included in the SFPS input deck. These were included in this table to represent the fact that these are likely to be in secular equilibrium with their parent radionuclides.

The particle size distribution assumed for NUREG-1864 was not identified in Appendix E. For purposes of this supplemental analysis, the particle size distribution for the particulate and activation product chemical group was assigned to be equal to the particle size distribution of the lanthanide chemical group in the SFPS, as lanthanides are presumed to be released due to mechanical measures rather than by volatility. Although this distribution is not the most appropriate for a dry cask storage scenario, using a cask-specific distribution would likely not change the conclusions of this Chapter. Specifically, the larger particle sizes expected to be associated with such a scenario would result in more deposition closer to the site, resulting in fewer exposed individuals within ten miles. The values used are given below:

Table 65 Particle Size Information

Particle Size Group	Particle Size Distribution	Dry Deposition Velocity (m/s)
1	3.2%	0.0011
2	15%	0.001
3	29%	0.0014
4	21%	0.0023
6	10%	0.0045
6	3.0%	0.0092
7	1.50%	0.0177
8	0.60%	0.0291
9	0.20%	0.0367
10	16%	0.0367

The release height and release fractions were varied to be consistent with NUREG-1864, Table E.1, as given in Table 66 below. To simulate the short duration release modeled in NUREG-1864, the number of plume segments was reduced to one with release starting at time zero, with a two minute (120 second) release duration. Reflecting the primarily mechanical rather than thermal nature of the release, the plume rise model was changed to a heat only option with a power of 18 kW to be consistent with NUREG-1864. However, parameters associated with building wake effects (e.g., building height, initial plume dimensions) were chosen to be consistent with SFPS values, as these values would be site specific. This represents another potential source of difference with NUREG-1864 values.

<u>Changes related to emergency response and long-term protective actions</u>

Consistent with the immediate release model and no evacuation assumption in NUREG-1864, the supplemental analysis eliminated all evacuating cohorts by changing the evacuation model to “No Evacuation”. However, sheltering and relocation parameters remained consistent with SFPS estimates. No changes were made to SFPS parameters for long-term protective actions such as decontamination levels and costs, as these were selected to be consistent with the SFPS site-specific values to allow for comparability. The application of SFPS emergency-phase sheltering, relocation, and long-term protective action parameters represent a source of difference between the results of NUREG-1864 and the supplemental analysis.

The results, and their comparability to the results provided in NUREG-1864, are provided in Table 66 and Table 67. Results are provided for a range of release fractions and release heights to facilitate comparison with the results reported in NUREG-1864. Staff considers the upper end of the release fraction for particulates in NUREG-1864 (0.12%) to represent a very conservative estimate of the potential respirable particulate release from a breached cask, as it assumes essentially complete fragmentation and entrainment of the high-burnup rim region and very limited filtration (10% released) within the cladding-fuel gap during entrainment flow. Reporting the full range of results, consistent with the results presented in Table E.1 of NUREG-1864, allows a more informed comparison of results including the effects of potential conservatisms in the analyses.

Results are reported for a variety of output metrics. These include both direct measures of health impacts (doses and probabilities of early and latent fatalities) as well as indirect measures such as the amount of land that is either temporarily interdicted or permanently condemned) or the numbers of temporarily or permanently displaced individuals.

Table 66 Parameters and Results from NUREG-1864, Table E.3

Release Fraction						
Noble Gases	Particles	CRUD	Release Height (m)	Ind. Risk of Prompt Fatality within 10 mi	Ind. Risk of LCF within 10 mi	Ind. Peak Dose at 1.2-1.6 km (Sv)
0.12	1.2E-03	1.5E-03	50	0	3.6E-04	1.85
0.12	1.2E-04	1.5E-04	50	0	5.2E-05	0.22
0.12	7.0E-06	1.5E-03	50	0	4.3E-06	2.6E-02
0.12	7.0E-07	1.5E-04	50	0	4.3E-07	2.7E-03
0.12	1.2E-03	1.5E-03	120	0	2.1E-04	0.14
0.12	7.0E-06	1.5E-03	120	0	2.6E-06	3.2E-03

Table 67 Supplemental Reanalysis with SFPS Input Deck

Release Fraction				Reanalysis with SFPS Input Deck						
NG	PART	CRUD	Release Height (m)	Prompt Fatality within 10 mi	Ind. Risk of LCF within 10 mi	Ind. Peak Dose at 1.2-1.6 km (Sv)	Collective Dose (0-50 mi) Person-Sv	Interdicted land in first year after accident (square miles)	Condemned land (square miles	Displaced Persons
0.12	1.2E-03	1.5E-03	50	0	7.1E-05	0.33	740	20	1.6E-03	5,800
0.12	1.2E-04	1.5E-04	50	0	8.9E-06	4.0E-02	86	1.4	0	150
0.12	7.E-06	1.5E-03	50	0	7.3E-07	6.6E-03	5.7	1.2E-02	0	1.8
0.12	7.E-07	1.5E-04	50	0	7.5E-08	6.8E-04	0.57	4.1E-06	0	-
0.12	1.2E-03	1.5E-03	120	0	5.1E-05	7.4E-02	780	24	3.9E-05	7,400
0.12	7.E-06	1.5E-03	120	0	7.0E-07	1.7E-03	6.2	3.2E-04	0	0.01

10.3.2 Quantitative Comparison of Dry Cask Storage and SFPS Analytical Results

The following table presents selected consequence results from the previous study of dry cask storage accidents, the supplemental dry cask storage study described above, and the SFPS.

Table 68 Comparison of consequence results from SFPS, NUREG-1864, and DCSS supplemental analyses

Metric	SFPS Results	NUREG-1864	DCSS Suppl. Analyses
Early fatalities (0 to 500 miles)	0	0	0
Individual LCF risk within 10 miles (conditional)	2.0e-4 to 4.4e-4	4.3e-7 to 3.6e-4	7.5e-8 to 7.1e-5
Collective dose within 50 miles in Person-Sv	7,400 to 39,000	Not reported	0.6 to 780
Collective dose within 500 miles in Person-Sv	27,000 to 350,000	Not reported	Not reported
Interdicted land (square miles)	170 to 9,400	Not reported	<<1 to 24
Condemned land (square miles)	<1 to 83	Not reported	<<1

10.3.3 Comparison of SFPS Results to Previous and Supplemental Cask Studies

Comparison of SFPS results to past dry cask studies is not straight-forward. This is because the type of information reported is different, the assumptions related to fuel and canister/cask damage are different, and the risks of dry cask handling, while low, are generally driven by design features that can vary significantly between different DCSS designs. For example, the NUREG-1864 study is based on a welded canister-in-overpack design, whereas the site selected for the SFPS study uses directly loaded bolted casks.

Nevertheless, meaningful comparisons can be made. An examination of the conditional individual latent cancer fatality probability metric demonstrates the effectiveness of emergency response and long-term protective actions at mitigating dose, consistent with the observations made in previous studies such as NUREG/CR-4982 and NUREG/CR-6451. The maximum consequences (in terms of latent cancer fatality probability) for both a pool accident and a dry cask accident, although involving substantially different amount of released material, are both limited to a range of 1E-4 to 1E-3 per event. The contrast to the much higher conditional consequence reported in NUREG-1738 (8.2E-2) is due to the assumption of a late evacuation coupled with a high source term in this study. The difference between impacts from pool and cask accidents is more clearly highlighted in measures related to the areal extent of contamination rather than in measures of peak individual risk. Inspection of Table 67 and Table 68 demonstrates that even in the case of very high release fractions from dry cask accidents, conditional results for metrics such as population dose or condemned or interdicted lands are several orders of magnitude lower than the low end of consequences of pool accidents. This comparison is significantly exaggerated if a less conservative estimate of the DCSS release fraction is used. The results suggest that a DCSS accident is unlikely to result in the need for extensive offsite protective action such as land interdiction or population displacement, in contrast to a pool accident that may require significant offsite protective action. Furthermore, for the risks (expressed as a frequency-weighted consequence) of a DCSS accident to be comparable to the risks of a pool accident, the frequency of a DCSS accident would have to be several orders of magnitude higher than that of a pool accident.

10.4 Summary of Assessment of Previous Studies

This assessment demonstrates that past SFP accident consequence estimates from large seismic events are similar to this study for most metrics. Comparison of this study to dry cask storage studies (NUREG-1864 and supplemental analyses from this Chapter), indicates that in some circumstances, the conditional individual LCF risk within 0 to 10 miles would be similar due primarily to the conservative upper bound estimate of the dry cask release as well as the expected effectiveness of protective actions in response to an SFP release. However, conditional results for metrics such as population dose or condemned or interdicted lands are several orders of magnitude lower for dry cask scenarios than the low end of consequences of pool accidents, due to the substantially smaller amount of released material.

11. REGULATORY ANALYSIS SCREENING SUMMARY

Based on past studies, the NRC has concluded that both spent fuel pools and dry casks provide adequate protection of public health and safety and the environment, and that the likelihood of an accident involving a radiological release from the spent fuel remains extremely small. While the staff believes that public health and safety is adequately protected for both spent fuel pool and dry cask storage, the Spent Fuel Pool Study (SFPS) provides one part of a technical analysis to confirm, using insights from Fukushima, that both spent fuel pools and dry cask storage continue to provide adequate protection. As indicated by its title, this study looks at the storage of spent nuclear fuel in spent fuel pools. The study also assesses whether any significant safety benefits (or detriments) would occur from expedited transfer of spent fuel to dry casks, and the potential costs associated with such expedited transfer.

The study establishes that both high and low density spent fuel pool arrangements provide reasonable assurance of adequate protection. The analysis in Appendix D, which is summarized here, assesses the benefits and costs of this action relative to the baseline of existing requirements, including current regulations and relevant orders.

11.1 <u>Decision Rationale</u>

11.1.1 Comparison to the Safety Goal Policy Statement

The Safety Goals for the Operation of Nuclear Power Plants: Policy Statement (51 FR 28044) (safety goal policy statement) was used to evaluate the impacts resulting from a severe spent fuel pool accident. The frequency of damage to the spent fuel pool is estimated to be approximately between $7.11\text{x}10^{-7}$ and $5.39\text{x}10^{-6}$ per year when considering all initiators that could challenge spent fuel pool cooling or integrity. This value, when compared to a target core damage frequency value of $1\text{x}10^{-4}$ per reactor-year in the Safety Goal Policy Statement, represents 0.71 to 5.39% percent of the overall frequency of core damage.

As described in Appendix D it is difficult to compare the estimated $7.11\text{x}10^{-7}$ to $5.39\text{x}10^{-6}$ per reactor-year release frequencies for the postulated spent fuel pool accident when considering all initiators to a target value of $1\text{x}10^{-5}$ per reactor year for a large early release frequency (LERF). The spent fuel pool source term is not similar to the core damage (or melt) source term. The consequences of a spent fuel pool accident are predicted to have no early fatalities and public health risk is dominated by latent cancer risks resulting from long-term exposures. Because the analyzed spent fuel accident is a slow progression with at least eight hours before an environmental release occurs and the resultant release is not expected to result in any offsite early fatalities, the analysis suggests that the spent fuel pool release does not fall within the definition of a large early release. Although this analyzed accident is different from a reactor accident, the spent fuel pool estimated release frequencies of $7.11\text{x}10^{-7}$ to $5.39\text{x}10^{-6}$ per reactor-year meet the $1\text{x}10^{-5}$ LERF guidelines.

Collective risk is based on the statistically expected number of early and latent cancer fatalities. The safety goal policy statement defines the early fatality area calculation as that within one mile from the site boundary. A ten-mile radius is defined for calculating latent cancer fatalities. The quantitative objective of the Policy Statement is for the risk to the population in the vicinity of a nuclear power plant from an accident at a nuclear power plant to not exceed 0.1 percent of the sum of cancer fatality risks resulting from all other causes. Based on recent data, the total fatality rate from cancer in the U.S. is 580,350 per 315,747,500 persons (http://www.census.gov/popclock/) or a risk of 1.84×10^{-3} per year, which results in a safety goal of 1.84×10^{-6} per year. Using the bounding frequency of damage to the spent fuel pool of 5.39×10^{-6} per year, which considers all initiators that could challenge spent fuel pool cooling or integrity, and the conditional individual latent cancer fatality risk within a 10-mile radius of 4.4×10^{-4} yields a latent cancer fatality risk of 2.37×10^{-9} per year. This calculated value of 2.37×10^{-9} latent cancer fatalities per reactor-year associated with a spent fuel pool accident represents a 0.13 percent fraction of the 1.84×10^{-6} per year societal risk goal.

Therefore, the risk and consequences of a spent fuel pool accident at the reference plant meet the Safety Goal Policy Statement public health objectives. They also meet the 1×10^{-5} per reactor-year LERF guideline. Therefore, the NRC concludes that a regulatory requirement for expedited transfer of spent fuel from the spent fuel pool to storage casks is not needed for the reference plant In order to meet the Safety Goals.

11.1.2 Cost-Benefit Analysis

The key findings of the analysis are as follows:

- Total Cost to the Reference Plant. The proposal to expeditiously move older spent fuel assemblies from pool storage to dry cask storage beginning in year 2014 to achieve and maintain a low-density loading in the pool within five years will result in an estimated present value cost of $46.77 million (using a 7-percent discount rate) and $41.82 million (using a 3-percent discount rate) over the next 26 years. The earlier upfront and incremental dry storage cask capital and loading costs dominated these incremental costs. The reference plant routine occupational health costs will result in an estimated present value cost of $2,000 (using a 7 percent discount rate) and $6,000 (using a 3-percent discount rate). Sensitivity analyses result in an estimated present value cost that ranged from $15.7 million to $46.8 million.

- Value of Benefits to the Reference Plant. The benefits for expeditious movement of spent fuel to dry cask storage will result in an estimated present value benefit of $493,000 (using a 7-percent discount rate) and $711,000 (using a 3-percent discount rate). These benefits result from the monetized value for averted public and occupational radiation exposure and averted onsite impacts and offsite property damage. Sensitivity analyses result in an estimated present value benefit that ranged from $0.5 million to $7.5 million.

- Costs to NRC. The NRC costs to require the expeditious movement of spent fuel to dry cask storage were conservatively ignored to calculate the maximum potential benefit. Even though the NRC is not expected to incur substantial implementation or annual costs for this alternative, these costs would further reduce the calculated net benefit for the proposed expeditious movement of older spent fuel assemblies from pool storage to dry cask storage for the reference plant.

There are uncertainties in estimating the frequency of events for natural phenomena that are postulated to challenge spent fuel pool cooling or integrity. There are also uncertainties in the calculation of event consequences in terms of the dispersion and disposition of radioactive material into the site environs. This is due in part to uncertainties regarding the degree to which topographical features and other phenomena are modeled at distances away from the reference plant. Estimating economic consequences also includes large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies and the loss of infrastructure on the general U.S. economy. An example of this is the supply chain disruptions that followed the 2011 Tohoku earthquake and subsequent tsunami in Japan or the 2004 Indian Ocean earthquake and tsunami in Thailand.

The NRC recognizes that there are also costs and risks associated with the handling and movement of spent fuel casks in the reactor building. These impacts, if included in this analysis, would further reduce the overall net benefit in relation to the regulatory baseline. These effects (e.g., the added risks of handling and moving casks) were conservatively ignored in order to calculate the maximum potential benefit by only comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and its implementation costs without consideration of cask movement risk.

The release of fission products to the environment resulting from other events that cause the loss of spent fuel pool cooling or integrity (i.e., missiles, heavy load drops, loss of cooling or make-up, inadvertent drainage or siphoning and pneumatic seal failures) are estimated to occur approximately once in 2.7 million years or 3.7×10^{-7} per reactor-year. Operator diagnosis and recovery are important factors considered in the development of the event frequencies for these events and portions of this evaluation are premised on licensees having taken appropriate actions to understand the potential consequences of spent fuel pool accident events and develop appropriate procedures and mitigating strategies to respond and mitigate the consequences.

In section 9.2 of the SFPS, a sensitivity analysis is provided in which a more favorable fuel pattern is considered in which eight cold assemblies surround each hot assembly (i.e., 1x8 fuel assembly pattern). Although only a few sensitivity analyses were performed using this configuration, the results looked promising for inhibiting spent fuel pool releases. The sensitivity calculations for the high-density 1x8 fuel pattern showed a shorter time to air coolability (i.e. no releases in OCP3). Even for the cases that led to the release of radioactive materials in OCP2, the release magnitude was much smaller than for the 1x4 fuel pattern, and comparable to the low density cases. The fuel thermal response has a slower heatup when compared to a fuel pattern in which four cold assemblies surround each hot assembly (i.e., 1x4 fuel assembly pattern) because there is more mass to absorb heat. Furthermore, the loading configuration may result in similar reductions in risk to the low-density storage option evaluated without the significant capital costs for implementation. Further evaluation of this alternative and possibly other loading configurations for all operating cycle phases is recommended as part of the regulatory analysis for expedited fuel movement as part of the program plan described in SECY-12-0095 to evaluate the transfer of spent fuel to dry cask storage.

Sensitivity analyses that extend the analyses beyond 50 miles show that the low-density spent fuel storage alternative was not cost-justified for any of the discounted sensitivity cases, and was only marginally justified if discounting was not applied. Therefore, the expedited transfer of spent fuel from pools to dry cask storage containers at the reference plant does not meet the cost-justified substantial safety enhancement criterion.

11.2 Further Actions

The NRC plans to use the insights from this study along with other analyses to inform a broader regulatory analysis, which will help decisionmakers determine whether operating or future nuclear power reactor licensees should be required to maintain a low-density configuration in their spent fuel pools.

The analysis for the reference plant and the longer-term generic regulatory analysis address the questions of what can go wrong; how likely is it; and what are the consequences. Although this approach is well established at the NRC and other government agencies, it is often difficult to explain following rare disasters such as the accident at Fukushima Dai-ichi, or in presenting the results of studies such as this one. It is not enough to look at only the estimates of the low probabilities for failing spent fuel pools or only at the worst-case consequences in the unlikely event of failures of spent fuel pool integrity and existing mitigating systems. One needs to look at the totality of information presented in this report, previous studies, operating experience, and assess both the potential advantages and disadvantages of regulatory actions regarding the movement of spent fuel from storage pools to dry cask storage containers.

12. SUMMARY AND CONCLUSIONS

12.1 Summary

This study sought to investigate the relative consequences between low and high-density loading situations for a selected site following a seismic event greater than the maximum earthquake reasonably expected to occur at the reference plant location. The NRC expects that the ground motion used in this study is more challenging for the spent fuel pool structure than that experienced at the Fukushima Daiichi nuclear power plant from the earthquake that occurred off the coast of Japan on March 11, 2011. That earthquake did not result in any spent fuel pool leaks. Chapter 1 discussed some of the considerations that are raised by stakeholders with respect to these differences. These are re-visited here to set the stage for presenting the study's findings.

- Expedited movement of fuel from the SFP 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)
OCP1	54	17	0.31
OCP2	59	22	0.37
OCP3	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 health effects, potential land contamination, and economic impacts

  This point is covered in the findings below.

- Removal of older fuel slightly 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)
OCP1	2,951	2,526	0.86
OCP2	3,567	3,143	0.88
OCP3	2,571	2,149	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.

<u>The results of the study are as follows:</u>

1. A beyond design basis event with a frequency of occurrence of 1 in 60,000 per year was used in this study, and more likely earthquakes are not expected to challenge the SFP structure.

2. Past studies have indicated that large seismic events could lead to the loss of structural integrity of the spent fuel pool liner. This study's results confirm that such a condition is unlikely. For the low probability seismic event described above, the study estimated a conditional probability of failure of 0.1. The specific conditions under which a failure might occur are site-specific.

3. NUREG-1353 (1989) predicted generic seismically-induced SFP liner failure likelihoods of $2x10^{-6}$ to $6x10^{-6}$ per year, generally associated with events greater than 0.5g peak ground acceleration. NUREG-1738 (2001) predicted generic seismically-induced SFP liner failure likelihoods of $2x10^{-7}$ to $2x10^{-6}$ per year, generally associated with events around 1.2 g. The current study looks at a seismic event in the range of 0.5 to 1 g, and estimates a site-specific SFP liner failure likelihood of $2x10^{-6}$ per year (based on the informed expectation that this seismic range has the greatest contribution to frequency-weighted consequences). Since the updated *initiating event* frequency estimate (based on the 2008 U.S. Geological Survey model) for the reference plant for events greater than 1 g is $6x10^{-6}$ per year, this portion of the seismic hazard (i.e., > 1 g) may contribute more significantly to the overall frequency-weighted consequences for the reference plant than previously anticipated, depending on the conditional structural SFP liner failure probability associated with these larger events. The effect of this scope limitation may be offset by potential conservatisms in the structural analysis described in Section 4 of this report.

4. In this study, no set of conditions short of a liner failure led to a radiological release in less than 3 days, which is consistent with past studies. In most cases, the available time to prevent a radiological release was much greater than 3 days.

5. In this study, without mitigative action, fuel is estimated to be air coolable for all but roughly 10% of the operating cycle[45]. Past studies estimated this time to be a greater fraction of the operating cycle, when hotter fuel was contiguously stored. In other words, use of the 1x4 pattern has a positive effect in promoting natural circulation air coolability and reducing the likelihood of a release should the SFP become completely drained. An even shorter time was predicted for the 1x8 pattern currently employed at PBAPS. While variability in SFP loading configurations was not a focus of this study, this report consistently shows the advantages associated with dispersed fuel loading patterns.

6. In the cases studied, which in general did not account for multiple or concurrent reactor and SFP accidents, the precise time to diagnose the need for SFP mitigation did not have an effect on the course of most scenarios.

    Nevertheless, the improved reliable and available SFP indication required by the NRC Order of March 12, 2012 (EA-12-051) is important to ensure that plant personnel can effectively prioritize emergency actions. The availability of such instrumentation may have changed the mitigation mode (makeup versus sprays) deployed to mitigate events that resulted in a release.

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[45] The actual time is between 37 days (not air coolable) and 107 days (air coolable), with 60 days representing the demarcation point between these two Operating Cycle Phases. The citation of 60 days as a representative value is reasonable based on other separate effects analyses not documented in this report. The actual time to air coolability could be more or less, depending on specific conditions.

7. This study considered variations in both pool loading and the effective deployment (or lack thereof) of 10 CFR 50.54(hh)(2) mitigation capabilities (i.e., water makeup or spray using portable equipment). Of these, effective deployment of mitigation had the largest impact on preventing a release of radioactive material, reducing the release frequency by a factor of about twenty (from $1x10^{-7}$/yr to $6x10^{-9}$/yr).

   Note that ongoing regulatory actions under Order EA-12-049 dated March 12, 2012 (and related correction dated March 13, 2012) increase the capability of nuclear power plants to mitigate beyond-design-basis external events, such as the seismic event studied here.

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

   - In terms of the likelihood of release within 3 days, no difference was seen.
   - In terms of consequences, the low density cases resulted in a smaller release due to the smaller inventory of radioactive material and the lower potential for hydrogen combustion. For high-density loading, the rapid draindown cases in general had smaller releases mainly because the reactor building remained intact (hydrogen combustions not predicted). For slow draindown events, longer times are available for deployment of mitigation. Without successful deployment of mitigation, the releases could be up to two orders of magnitude larger (these cases are associated with hydrogen combustion events).

9. For all scenarios, no offsite early fatalities attributable to acute radiation exposure are predicted to occur. Due to radioactive decay, spent fuel pools tend to have significantly less shorter-lived radionuclides (e.g. I-131) than reactors. Partly because of this, the release is not predicted to be fast and large enough to significantly exceed offsite dose levels necessary to induce early fatalities. . When necessary, emergency response as treated in this study effectively prevents early fatalities from acute radiation exposure.

10. In both high and low density loading without successful deployment of mitigation, the individual latent cancer fatality risk within 10 miles for the studied scenarios is predicted to be on the order of 10^{-10} to 10^{-11} per year, based on the linear no threshold dose response model. While this risk is scenario-specific and related to a single spent fuel pool, it is several orders of magnitude lower than the 2e-6 per year individual latent cancer fatality risk corresponding to the quantitative health objective for latent cancer fatalities and therefore unlikely to contribute significantly to a risk that would challenge the Commission's safety goal policy (NRC 1986). In addition, there is uncertainty in the risk calculations because it is dominated by low doses. As a perspective on uncertainty, excluding the uncertain effects of low doses significantly reduced the quantified individual latent cancer fatality risk within 10 miles. Average individual latent cancer fatality risk is low because of low release frequencies and the expected protective actions.

11. Average individual latent cancer fatality risk is low and decreases slowly as a function of distance from the plant. For scenarios with large releases, significant collective doses are estimated; however, risk of cancer fatalities from these doses would be a small fraction of the risk of cancer fatalities from all causes. Additionally, these individual risks are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough to be considered habitable. In comparing pool configurations, collective dose (and latent cancer fatalities) for the studied scenarios could be an order

of magnitude higher for the high density loading situation as compared to the low density loading situation

12. The amount of land interdiction for the studied scenarios could be up to two orders of magnitude greater for certain high density loading situations as compared to the low density loading situations. Also, like releases in the low density loading situation, successfully deployed mitigation in the high density loading situation is predicted to reduce the amount of land interdiction to a similar extent. For both situations, the major difference is driven by hydrogen combustion events and associated large releases, which are only predicted to occur in scenarios with unsuccessful deployment of mitigation.

13. While the likelihood of release is very low, offsite protective measures in the form of population relocation and land interdication may be extensive. High-density loading releases without 10 CFR 50.54(hh)(2) mitigation measures are calculated to result in release frequency-weighted land interdiction values of 0.001 mi^2 per year and 0.5 displaced individuals per year which are arrived at by multiplying the estimated frequency and the estimated consequence. While the amount of land interdiction can be large, the fraction expected to be permanently interdicted is small if a release were to occur. For low-density loading or with successful deployment of 10 CFR 50.54(hh)(2) mitigation measures, considerably less land interdiction and displaced individuals are predicted.

14. A comparison of the risks of different fuel handling strategies, such as current practice and expedited transfer, depends on several factors including the relative, site-specific risks, and the time spent in each stage of spent fuel storage. Other risks, such as the risk from cask drop events damaging fuel in the cask or the SFP, may at least partially offset the benefit of lower spent fuel pool risk from low density loading.

15. The human reliability study shows that in most situations SFP mitigation can be deployed in time to prevent release given the assumptions that sufficient plant staff and equipment is available for SFP mitigation and the work area is accessible to perform mitigation. There are two exceptions where mitigation will be ineffective under the moderate leak scenarios: (1) the earthquake occurs at the beginning of a refueling outage when the spent fuel is too hot for the assumed mitigation; and (2) the earthquake occurs when spent fuel is relatively hot and the reactor and spent fuel pool are hydraulically disconnected resulting in insufficient time to deploy mitigation and natural cooling mechanisms cannot prevent fuel damage. This study identified that possible improvements in mitigation flow and nozzle placement in low-dose locations could improve mitigation success likelihood, but this would require further verification.

16. This study demonstrates that past SFP risk estimates from large seismic events are similar to this study for most consequence metrics (see Chapter 10). Comparison of this study to dry cask storage studies (NUREG-1864 and supplemental analyses from Chapter 10) indicates that in some circumstances, the conditional individual LCF risk within 10 miles would be similar due primarily to the conservative upper bound estimate of the dry cask release as well as the expected effectiveness of protective actions in response to an SFP release. However, conditional results for metrics such as temporary or permanently interdicted land or population dose are several orders of magnitude lower for dry cask scenarios than the low end of consequences of pool accidents, due to the substantially smaller amount of released material.

17. The application of this study's results to the NRC's regulatory analysis guidelines indicates that requiring the low-density spent fuel pool storage alternative is not justified for the reference plant given the analysis assumptions. The risk due to beyond design basis accidents in the spent fuel pool analyzed in this study is sufficiently low that the added costs involved with expediting the movement of spent fuel from the pool to achieve the low-density fuel pool storage alternative are not warranted. Sensitivity analyses that extend the analyses beyond the primary area considered also show that the low-density spent fuel storage alternative was not cost justified for any of the discounted sensitivity cases.

12.2 Conclusions

In conclusion, past SFP risk studies have shown that storage of spent fuel in a high-density configuration is safe and risk is low. This study is consistent with earlier research conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking. The study estimated that the likelihood of a radiological release from the spent fuel pool resulting from the selected severe seismic event analyzed in this study is on the order of one time in 10 million years or lower. For the hypothetical releases studied, no early fatalities attributable to acute radiation exposure were predicted and individual latent cancer fatality risks are projected to be low, but extensive protective actions may be needed.

The study results demonstrated that in a high-density loading configuration, a more favorable fuel pattern or successful mitigation generally prevented or reduced the size of potential releases. Low-density loading reduced the size of potential releases but did not affect the likelihood of a release. When a release is predicted to occur, individual early and latent fatality risks for individuals within 10 miles do not vary significantly between the scenarios studied because protective actions, including relocation of the public and land interdiction, were modeled to be effective in limiting exposure. The beneficial effects in the reduction of offsite consequences between a high-density loading scenario and a low-density loading scenario are primarily associated with the reduction in the potential extent of land contamination and associated protective actions. However, the risk due to beyond design basis accidents for the spent fuel pool studied is sufficiently low that the added costs involved with expediting the movement of spent fuel from the pool to achieve low-density fuel pool storage are not warranted. Therefore, the expedited transfer of spent fuel from pools to dry cask storage containers does not provide a substantial safety enhancement for the reference plant. The NRC plans to use the insights from this analysis to inform a broader regulatory analysis of the spent fuel pools at all US operating nuclear reactors.

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APPENDIX A: DETAILED EMERGENCY RESPONSE MODELS

The detailed evacuation timing and speeds for each cohort developed using the information and approach described in Section 7.1.4 are described in this appendix. 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, which are the beginning, middle, and late phases. Average evacuation speeds were derived from the reference plant's ETE report. Speed adjustment factors are 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. The remainder of the evacuation, following period defined by DURMID, is the late phase.

A.1 Evacuation Model 1: WinMACCS response parameters for sequences where PAGs are not exceeded beyond the EPZ.

The following cohorts were established for this evacuation model:

0 to 10 Miles, Early Evacuees: This population begins to evacuate before receiving an evacuation order. Focus group work conducted to support NUREG/CR-6953, Volume 2 (NRC, 2008c) 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 percent of residents of EPZs have packed a "go-bag" and are ready to leave. Because the accident is initiated by a severe earthquake, it is assumed 30 percent of the public evacuate.

10 to 20 Miles, Shadow: These residents evacuate from areas that are not under an official evacuation order. 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, a uniform fraction of the population is assumed to evacuate within the 10- to 20-mile region. Because the accident is initiated by a severe earthquake, it is assumed 30 percent of the public evacuate. This cohort will begin evacuating as they hear of the evacuation orders and observe EPZ evacuees traveling through the area.

0 to 10 Miles, Public: This population group evacuates over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is

modeled as a single cohort, while the rest of the group is captured as different cohorts, such as the tail.

<u>0 to 10 Miles, Special Facilities:</u> This is a small but unique population group within the 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.

<u>0 to 10 Miles, Tail:</u> The tail represents the last 10 percent of the EPZ population who typically take a longer time to begin to evacuate.

<u>0 to 10 Miles, Schools:</u> 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.

<u>Nonevacuating Public:</u> A portion of the public does not follow protective action orders. It is assumed that 0.5 percent of the general public within the EPZ refuse to evacuate.

Table 69 Evacuation Model 1: EPZ Evacuation

Population			Response Delays (hours)				Phase Duration (hr)		Evacuation Travel Speeds (mph)		
Cohort		Population Fraction	Siren (OALARM)	Delay to Shelter	Delay to Evacuation	Total (Depart time)	Early (DURBEG)	Middle (DURMID)	Early (ESPEED)	Middle (ESPEED)	Late (ESPEED)
1	0 to 10 miles Early Evacuees	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	0.006	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	20	15	20
6	0 to 10 miles Nonevacuating Public	0.005	1	-	-	-	-	-	-	-	-

For this sequence, hotspot relocation is 5 rem at 4 hours and normal relocation is 1 rem at 8 hours. The values were established specific for this evacuation model developed for sequences with relatively small releases.

A.2 Evacuation Model 2: WinMACCS response parameters for late release sequences where the PAG is exceeded beyond the EPZ.

Preliminary results suggest that emergency-phase doses of 1 rem may extend 30 to 40 miles from the plant for some of the larger postulated releases. The EPA PAG suggests 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 a 30-mile radius of the reference plant is approximately 1.4 million. The population within the 40-mile radius is approximately 3.4 million. Because of larger populations at longer distances, it is important to better understand the potential directions that the plume would travel. The reference plant's wind rose in the figure below suggests that the predominant wind direction is to the south and east, which is generally toward lower population areas. A secondary direction in terms of likelihood is to the northwest to north. This region is also low in population. Thus, if a release were to occur, it is more likely that a relatively small population would be affected than if the release occurred at a facility near a major city.

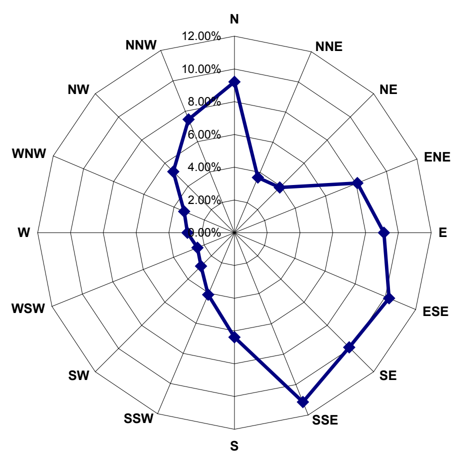

Figure 138 The Reference Plant's Wind Rose

It is assumed in this evacuation model that ORO's begin to order evacuations beyond the EPZ 24 hours after the start of the accident. This is based on preliminary results that indicates a large release beginning at 48 hours. For this sequence, the population within a 30-mile radius is evacuated after the EPZ has evacuated. The overall evacuation would be implemented as a staged evacuation, which is common for plume-related emergency response. In addition, a SIP is assumed to be ordered for the 30- to 40-mile radius area.

To develop an ETE and corresponding speeds for the areas beyond the EPZ, it was assumed 90 percent of the general public who reside between 10- and 30-miles from the plant can be evacuated 24 hours after ordered to evacuate. This is consistent with the lengthy travel times observed in hurricane evacuations of similar populations. The last 10 percent (evacuation tail) is estimated to take an additional 12 hours. Because of the lengthy time for this release to the atmosphere, this evacuation model effectively includes two separate evacuations, the first being within EPZ followed later by the 10- to 30-mile area.

The following cohorts were established for this evacuation model:

<u>0 to 10 Miles, Schools:</u> 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.

<u>0 to 10 Miles, Early Evacuees:</u> This population begins to evacuate before receiving an evacuation order. Focus group work suggested that some residents are prepared and ready to evacuate at the first indication of an accident at the nuclear power plant (NRC, 2008c). Because the accident is initiated by a severe earthquake, it is assumed 30 percent of the public evacuate.

<u>0 to 10 Miles, Public:</u> This population group would evacuate over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is modeled as a single cohort, while the rest of the group is captured as different cohorts, such as the tail.

<u>10 to 20 Miles, Shadow:</u> These residents evacuate from areas that are not under an official evacuation order. 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. For simplicity, this cohort is assumed to be distributed uniformly over the 10- to 20-mile area. This cohort begins evacuating as they observe EPZ evacuees traveling through the area.

<u>0 to 10 Miles, Special Facilities:</u> 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.

<u>0 to 10 Miles, Tail:</u> The tail represents the last 10 percent of the EPZ population who typically take a longer time to begin to evacuate.

<u>10 to 30 Miles, Public:</u> This population group evacuates over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is modeled as a single cohort that enters the roadway network while EPZ evacuees are travelling through. The rest of the group is captured as different cohorts, such as the tail.

<u>10 to 30 Miles, Special Facilities:</u> Special vehicles needed to evacuate these facilities require additional time to mobilize and support the evacuation.

<u>30 to 40 Miles, Shadow:</u> A shadow evacuation may be expected in the area beyond the evacuation area. For this analysis, it is assumed that 20 percent of the general public from the 30- to 40-mile area evacuate. This cohort begins evacuating as they hear the order to evacuate for the 10- to 30-mile area, or observe evacuees traveling through the area.

<u>10 to 30 Miles, Tail:</u> The tail represents the last 10 percent of the population of this area who typically take a longer time to begin to evacuate.

<u>30 to 40 Miles, Shelter in Place (SIP):</u> For this evacuation model, it is assumed that 80 percent of the public remaining after the shadow evacuation complies with the SIP order.

<u>Nonevacuating Public:</u> A small portion of the public does not follow protective action orders. It is assumed that 0.5 percent of the general public within the 0 to 40 mile area refuse to evacuate. This group, however, is subject to relocation and a portion of this cohort is relocated according to the relocation parameters discussed above.

Table 70 Evacuation Model 2: Evacuation for PAGs exceeded beyond the EPZ (SFP release after 40 hours.)

Population			Response Delays (hours)			Phase Duration (hr)		Evacuation Travel Speeds (mph)		
Cohort		Population Fraction	Delay to Shelter*	Delay to Evacuation	Total (Depart time)	Early (DURBEG)	Middle (DURMID)	Early (ESPEED)	Middle (ESPEED)	Late (ESPEED)
1	0 to 10 miles Schools	.172	0.25	1.25	1.5	0.25	2	20	15	20
2	0 to 10 miles Early Evacuees	.2	0.5	0.5	1	1	2	20	10	20
3	0 to 10 miles General Public	.517	1	2	3	0.25	3	5	2	20
4	10 to 20 miles Shadow	.3	2	2	4	0.25	6	20	15	20
5	0 to 10 miles Special Facilities	.006	0	5	5	3	2	2	5	20
6	0 to 10 miles Evacuation Tail	.1	3	3	6	2	2	2	5	20
7	10 to 20 miles General Public	.552	24	4	28	2	18	2	1	20
	20 to 30 miles General Public	.852								
8	10 to 30 miles Special Facilities	.043	15	15	30	1	10	1	1	20
9	30 to 40 miles Shadow	.2	24	8	32	1	6	15	5	20
10	10 to 30 miles Evacuation Tail	.1	24	16	40	10	2	1	10	20
11	30 to 40 miles Shelter in Place	.795	NA	NA	NA	NA	NA	NA	NA	NA
12	0-40 miles Non-evacuating Public	.005	NA	NA	NA	NA	NA	NA	NA	NA

*Delay to shelter is from the start of the accident (i.e. OALARM set to zero)

For this sequence, hotspot relocation is 5 rem at 4 hours and normal relocation is 1 rem at 16 hours. The releases for these sequences do not begin until about 40 hours or thereafter, and hotspot relocation does not begin until 4 hours after the plume reaches the location. OROs would be able to assemble considerable resources to monitor radiological conditions and could be expected to relocate people relatively rapidly should it be necessary.

A.3 Evacuation Model 3: WinMACCS response parameters for early release sequences where the PAG is exceeded beyond the EPZ.

This evacuation is similar to Evacuation Model 2. Preliminary results suggest that certain sequences that have large releases that begin between 8 and 18 hours and are capable of emergency-phase doses that exceed the PAGs beyond the EPZ. It is expected that dose projections would indicate protective actions beyond the EPZ are necessary. It is assumed that evacuation of the area beyond the EPZ would begin at 10 hours after the start of the accident.

Because the evacuation of the 10- to 30-mile area begins at 10 hours, this response is typical of a staged evacuation that would be employed in the case of a chemical release. The EPZ evacuation impacts the evacuation speeds of the 10- to 30-mile area. The following cohorts were established for this evacuation model:

0 to 10 Miles, 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 Miles, Early Evacuees: This population begins to evacuate before receiving an evacuation order. Focus group work suggested that some residents are prepared and ready to evacuate at the first indication of an accident at the nuclear power plant (NRC, 2008c). Because the accident is initiated by a severe earthquake, it is assumed 30 percent of the public evacuate.

0 to 10 Miles, Public: This population group would evacuate over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is modeled as a single cohort, while the rest of the group is captured as different cohorts, such as the tail.

10 to 20 Miles, Shadow: These residents evacuate from areas that are not under an official evacuation order. 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. For simplicity, this cohort is assumed to be distributed uniformly over the 10- to 20-mile area. This cohort begins evacuating as they observe EPZ evacuees traveling through the area.

0 to 10 Miles, 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 Miles, Tail: The tail represents the last 10 percent of the EPZ population who typically take a longer time to begin to evacuate.

10 to 30 Miles, Public: This population group evacuates over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is modeled as a single cohort that enters the roadway network while EPZ evacuees are travelling through. The rest of the group is captured as different cohorts, such as the tail.

10 to 30 Miles, Special Facilities: Special vehicles needed to evacuate these facilities require additional time to mobilize and support the evacuation.

30 to 40 Miles, Shadow: A shadow evacuation may be expected in the area beyond the evacuation area. For this analysis, it is assumed that 20 percent of the general public from the 30- to 40-mile area evacuate. This cohort begins evacuating as they hear the order to evacuate for the 10- to 30-mile area, or observe evacuees traveling through the area.

10 to 30 Miles, Tail: The tail represents the last 10 percent of the population of this area who typically take a longer time to begin to evacuate.

30 to 40 Miles, Shelter in Place (SIP): For this evacuation model, it is assumed that 80 percent of the public remaining after the shadow evacuation complies with the SIP order.

Nonevacuating Public: A small portion of the public does not follow protective action orders. It is assumed that 0.5 percent of the general public within the 0 to 40 mile area refuse to evacuate. This group, however, is subject to relocation and a portion of this cohort is relocated according to the relocation parameters discussed above.

Table 71 Evacuation Model 3: Evacuation for PAGs exceeded beyond the EPZ (SFP release after 8 hours).

Population			Response Delays (hours)			Phase Duration (hr)		Evacuation Travel Speeds (mph)		
Cohort		Population Fraction	Delay to Shelter*	Delay to Evacuation	Total (Depart time)	Early (DURBEG)	Middle (DURMID)	Early (ESPEED)	Middle (ESPEED)	Late (ESPEED)
1	0 to 10 miles Schools	.172	0.25	1.25	1.5	0.25	2	20	15	20
2	0 to 10 miles Early Evacuees	.2	0.5	0.5	1	1	2	20	15	20
3	0 to 10 miles General Public	.517	1	2	3	0.25	3	5	2	20
4	10 to 20 miles Shadow	.3	2	2	4	0.25	6	20	15	20
5	0 to 10 miles Special Facilities	.006	0	5	5	3	2	2	5	20
6	0 to 10 miles Evacuation Tail	.1	3	3	6	2	2	2	5	20
	10 to 20 miles General Public	.552								
7	20 to 30 miles General Public	.852	6	4	10	2	18	2	1	20
8	10 to 30 miles Special Facilities	.043	0	20	20	1	10	1	1	20
9	30 to 40 miles Shadow	.2	6	6	12	1	6	15	5	20
10	10 to 30 miles Evacuation Tail	.1	10	20	30	10	2	1	10	20
11	30 to 40 miles Shelter in Place	.795	NA	NA	NA	NA	NA	NA	NA	NA
12	0-40 miles Non-evacuating Public	.005	NA	NA	NA	NA	NA	NA	NA	NA

*Delay to shelter is from the start of the accident (i.e. OALARM set to zero)

For this evacuation model, the hotspot relocation is 5 rem at 26 hours and normal relocation is 1 rem at 38 hours. The plume's initial release for these sequences begins between 8 and 18 hours after accident initiation. The assumed inability of the licensee to halt this release is the basis for expecting decision makers to expand the evacuation. Sequences that develop rapidly could challenge ORO resources to assess radiological conditions beyond the evacuated areas and delay relocation of affected people. In this evacuation model, hotspot relocation does not begin until 26 hours after the release arrives, in order to account for the relatively earlier release and the evacuation of the public from the 10-30 mile area (which is expected to take as long as 24 hours).

APPENDIX B: A QUALITATIVE RISK COMPARISON OF SPENT FUEL STORAGE STRATEGIES

B.1 Introduction

In Staff Requirements Memorandum (SRM) M120607C, dated July 16, 2012, the Commission directed the staff to conduct a comparative assessment of the results of the Spent Fuel Pool Study (SFPS) against previous studies of the safety consequences associated with loading, transfer, and long-term storage in dry cask storage systems (DCSS). Since the SFPS only includes a consequence study of certain seismic events, it is necessary to create a step-by-step model that can be used to compare safety consequences associated with the various stages of onsite spent fuel management. As part of the response to this SRM, this analysis (1) defines several fuel storage strategies to be compared, (2) develops a structure for calculating the difference in risks between these strategies, (3) identifies what relevant information exists, and (4) identifies what new information may be needed.

B.2 Spent Fuel Storage Strategies

For the purpose of studying this issue, two distinct spent fuel storage strategies commonly considered are defined: (1) current practice and (2) expedited transfer of spent fuel into dry storage. A large amount of variation exists in current spent fuel storage practices at various sites. Expedited transfer strategies, if implemented, would also be expected to vary considerably from site to site. Rather than attempting to bound all of the practices that are or may be implemented at various sites, this appendix will focus on the key elements of spent fuel storage strategies covered by existing risk analyses. Current practice generally consists of loading casks only when the pool, in a high density configuration, is nearly full. Just enough casks are loaded to maintain the capability to unload one full core into the pool. Expedited transfer of spent fuel into dry storage involves loading casks at a faster rate for a period of time to achieve a low density configuration in the spent fuel pool (SFP). The expedited process maintains a low density pool by moving all fuel cooled longer than 5 years out of the pool.

B.3 Spent Fuel Storage Stages

The risks associated with spent fuel storage will vary throughout the lifetime of a plant site and will depend on how the fuel is stored, and in what quantities. To analyze the lifecycle risk of spent fuel storage at a plant site, this appendix defines five fuel storage stages, beginning with a low-density pool approaching high density and ending with the final core offload being loaded into casks. The current practice strategy will not include the expedited transfer stage defined below.

Stage 1 consists of the fuel being offloaded into a low-density pool that eventually reaches high density. It is assumed that no casks are loaded during this stage as is generally industry practice. Stage 2 is when the pool is full in a high-density configuration and only as many casks are loaded as necessary. Stage 3 is during the expedited transfer period when the amount of cask loading is increased so as to decrease the inventory in the SFP. Stage 4 begins when expedited loading has been completed and the pool has returned to a low-density configuration and a lower rate of cask loading. Stage 5 begins when the reactor is permanently shut down and the last core is offloaded to the SFP. Stage 5 ends when all fuel has been placed in dry cask storage.

B.4 Risk of Spent Fuel Storage

This section presents generalized equations for the risk of spent fuel storage. These equations will serve as a guide to a subsequent discussion of the relative risk between storage stages and what drives the changes in risk.

The total annual risk of storing spent fuel during any stage can be expressed as the sum of the risk from the SFP and the dry casks. This can be expressed as,

$$R = R_{casks} + R_{sfp}$$

where: R = annual risk of spent fuel
R_{casks} = annual risk of loading and storing fuel in dry casks
R_{sfp} = annual risk of the spent fuel pool

The risk from loading and storing each dry cask is assumed to be constant and only dependent on the number of casks loaded or stored. The total risk of loading and storing casks is given by,

$$R_{casks} = r_{cask,load} * N_{load} + r_{cask,store} * N_{store}$$

where: $r_{cask,load}$ = risk per cask loaded
N_{load} = number of casks loaded per year
$r_{cask,store}$ = risk per cask in storage
N_{store} = number of casks being stored

Section 1.5 of the SFPS report provides an overview of contributors to SFP risk. The majority of SFP risk is thought to emanate from a loss of water from a leak or a boiloff. The risk from the SFP can then be characterized as the frequency of fuel uncovery multiplied by the consequences of the accident. The uncovery frequency is the sum of the frequency of uncovery from cask drops, seismic events, and other initiators. The frequency of a cask drop damaging the pool and leading to uncovery is the product of the number of casks loaded, the probability of a drop, and the probability of pool damage and uncovery given a drop. This value is given by,

$$R_{sfp} = (N_{load} * P_{drop} * P_{damage} + F_{seismic} + F_{other}) * C_{uncovery}$$

where: P_{drop} = probability of a cask drop per cask loaded
P_{damage} = probability of a dropped cask leading to fuel uncovery
$F_{seismic}$ = frequency of uncovery from seismic events
F_{other} = frequency of uncovery from sources other than cask drops and seismic
$C_{uncovery}$ = consequences of fuel uncovery

The SFPS provided a detailed analysis of the consequences, $C_{uncovery}$, for a particular site and a calculation of $F_{seismic}$ for seismic bin 3. To fully calculate $F_{seismic}$, seismic bin 4 would need to be analyzed as well. The SFPS did not analyze other initiators for pool accidents that contribute to SFP risk.

B.5 Risk during Each Stage

Figure 139 is an illustration of the spent fuel risks during each stage for both the current practice and expedited transfer strategies. Though the “current practice” strategy does not include expedited loading, it is divided into the same stages (time periods) for comparison purposes. The figure depicts the SFP risk, dry cask loading risk and dry cask storage risk. The SFP risk includes the risk to the pool from dropped casks.

Figure 139 includes the following major assumptions and limitations:

- The figure is only intended to show trends, not absolute differences in risk. No specific numbers were used to generate the figure.

- The type of risk used will significantly affect the relative values of different portions of the figure. Table 37 gives the ratio of consequences between a high- and low- density pool for several types of risk, with the risk reduction from a low-density pool varying from a factor of 2.1 for individual latent cancer fatality risk for 0–10 miles to 56 for land interdiction.

- The amount of time spent in each stage will affect a calculation of the total risk.

- Changes in N_{load}, N_{store}, and $C_{uncovery}$ are the drivers for the change in risk between stages. Other terms in the above equations are assumed to be constant.

- The specific shape of the figure will depend on site specific parameters such as the pool’s susceptibility to be drained from a cask drop event (P_{damage}).

- Risks are averaged over the operating cycle to demonstrate general trends rather than short-term changes in risk.

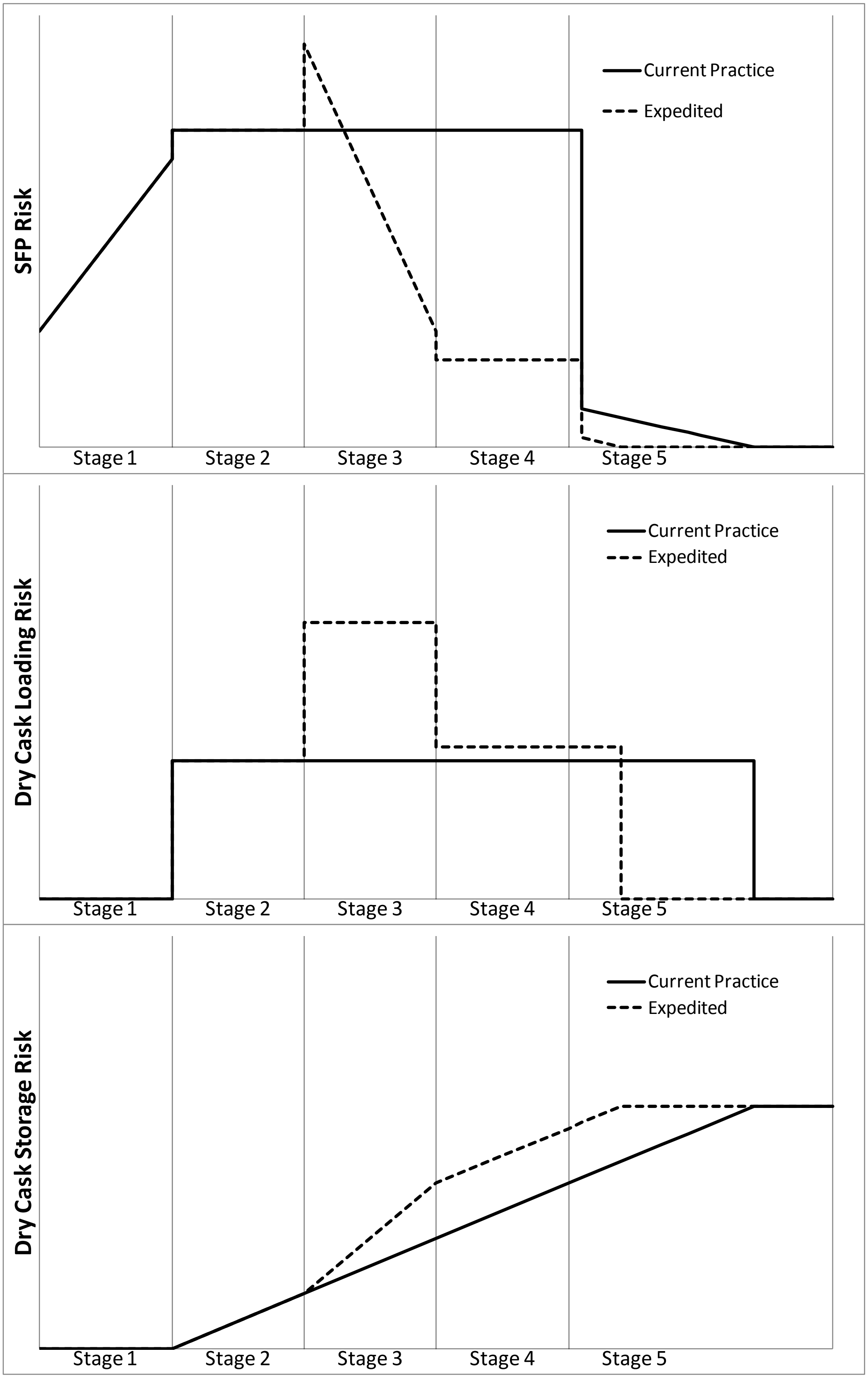

Figure 139 Graphical representation of spent fuel risks

During Stage 1, for both the current practice and expedited transfer scenarios, the amount of spent fuel being stored in the SFP is increasing. As the pool moves from low-density loading to high-density loading, the consequences of fuel uncovery, $C_{uncovery}$, and thus SFP risk, R_{sfp}, increase. Dry cask loading and storage risk is zero since no casks are loaded during this stage (N_{load} and N_{store} = 0).

At the beginning of Stage 2, the pool reaches a high-density configuration and cask loading begins. SFP risk is greater than at the end of Stage 1 because of the possibility of cask drops ($N_{load} > 0$). It is assumed that the rate of dry cask loading is constant throughout this stage, leading to a constant loading risk and a gradually increasing storage risk as more casks are stored (N_{store} is increasing). For the current practice spent fuel storage strategy, the same loading rate is maintained in Stages 2, 3 and 4 and the pool is maintained at maximum loading.

For the expedited transfer strategy, Stage 3 is the beginning of an increased cask loading rate (N_{load} increases). The SFP risk undergoes another step increase (from the increased frequency of cask drop events) and then declines as the pool approaches a low-density configuration and the consequence of fuel uncover, $C_{uncovery}$, decreases. Cask loading risk increases as the rate of loading, N_{load}, increases. Storage risk increases at a faster rate while more casks are being loaded.

Stage 4 marks the end of the expedited transfer phase when the pool has reached a low-density configuration. The cask loading rate and risk decrease to levels slightly higher than in Stage 2. The hotter fuel being loaded requires more lower capacity casks. The decrease in cask loading rate, N_{load}, and consequences of uncovery, $C_{uncovery}$, decrease the SFP risk which remains at a constant, lower level for the rest of the stage. Cask storage risk continues increasing at a slower rate.

At the beginning of Stage 5, the reactor ends its final operating cycle and fuel in the reactor core is offloaded to the SFP. After several months, the fuel in the SFP is generally capable of being air cooled, and the risk decreases for both the current practice and expedited transfer strategies. The risk is nonzero because of the possibility of events which may impede air cooling of the fuel. It is assumed that casks continue being loaded at a constant rate until the pool is empty. The SFP risk continues decreasing gradually as the fuel cools and is removed from the pool. When the cask loading is complete, the pool risk and the cask loading risk go to zero, and the cask storage risk stabilizes. The low-density pool in the expedited transfer case contains less fuel and is emptied sooner since much of the fuel was removed in Stage 3.

B.6 Total Risk over Time

Two components compromise the total risk over a period of time, (1) the amount of time spent in each stage and (2) the risk in each stage. The time spent in each stage will vary depending on how soon expedited transfer is initiated (if at all), how long it takes, and how long the reactor continues to operate with a low-density pool. The risks during each stage will depend on the relative hazards at each site.

The recent EPRI report analyzing spent fuel transfer (EPRI, 2012) estimates that expedited transfer for fuel cooled longer than five years would take between 8-15 years at most sites. The amount of fuel in the SFP, whether the site has multiple units sharing a single cask handling crane, and the availability of trained personnel, and equipment affect this estimate. The amount of time then spent in a low-density configuration depends on how much longer the reactor is operated. It is expected that most reactors will apply for and receive extensions of their operating licenses to 60 years.

The site-specific risks during each stage will drive whether expedited transfer decreases risk, and over what timeframe. An accounting of the risk increases and decreases of expedited transfer compared to current practice will illustrate this point.

For expedited transfer, risk increases relative to current practice are seen in the following stages:

- SFP risk in the beginning of Stage 3 from cask drop events,
- Cask loading risk during Stage 3, and to a lesser extent in Stages 4 and 5,
- Storage risk in Stages 3 and 4 and the beginning of Stage 5

Risk decreases occur in:

- SFP risk later in Stage 3 and in Stage 4,
- cask loading risk in Stage 5

Since the total number of casks loaded is likely to be only slightly higher for the expedited strategy, the increase in cask loading risk during Stage 3 is expected to be mostly offset by the decrease in risk in Stage 5. Furthermore, previous studies such as NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System At a Nuclear Power Plant," issued March 2007, and an EPRI report entitled, "Probabilistic Risk Assessment (PRA) of Bolted Storage Casks," indicate that dry cask storage risk is likely lower than SFP risk. Hence, this leaves a comparison of the increase in SFP risk from cask drops to the decrease in risk from a low-density pool.

The risk increase will depend on the pool's susceptibility to drops (discussed below). For example, if the SFP risk at a particular site were dominated by cask drop risk, it could take many years of operating with a low-density pool to "pay back" the temporary risk increase seen at the beginning of Stage 3. This increase in risk could be mitigated by only loading casks during operating cycle phases 4 or 5 when the SFP is typically air coolable for a complete draindown. In contrast, a pool with low susceptibility to cask drops and high seismic risk will see a greater risk benefit sooner.

One aspect not included in the above figure is the potential need to repackage casks that have already been loaded, before interim storage or permanent disposal. Given the uncertainty in the national strategy for spent fuel, the specifications for disposal at a long-term repository or interim storage site are currently unclear but may be developed in the future. Earlier movement of fuel into current cask designs increases the probability that fuel will have to be repackaged to meet these specifications.

B.7 Availability of Information

The equations defined earlier identify the variables needed to calculate the risk of spent fuel storage. The general shape of
Figure 139 is believed to generally apply to operating reactors. However, no analysis has attempted to quantify the horizontal or vertical axes. The discussion below points to existing

information that could be useful in quantifying these variables as well as what further information could be useful.

B.7.1 Cask Risks ($r_{cask,load}$ and $r_{cask,store}$)

Two major studies have addressed the risk of dry cask loading and storage, NUREG-1864 and EPRI's probabilistic risk assessment of bolted storage casks. NUREG-1864 analyzed a welded cask at a particular boiling-water reactor site. In a complementary effort, the EPRI study analyzed a bolted cask at a generic pressurized-water reactor site. NUREG-1864 identified cask drops and aircraft strikes as the major contributors to risk during cask loading and cask storage, respectively. The EPRI study found the major contributors to risk during cask loading to be drops, failure of the refueling building, and high temperature fires. During storage, major risk contributors were high temperature and forces (e.g., explosions) or heavy loads (e.g., high winds) leading to cask tipover. The difference in major contributors to risk is likely because of differences in the methods used in the analysis as well as differences in the analyzed systems. Regardless, both studies found the risk of dry cask loading and storage to be extremely small. Key factors contributing to this result include the robustness of the analyzed casks and the availability of the refueling building ventilation system, which is capable of significantly decreasing the source term for many accident sequences that result in a cask release.

Several additional factors may affect a calculation of dry cask risk. Considerable uncertainty exists in the source term expected from cask accident sequences resulting in a significant range in consequences, as discussed in Chapter 10. Different cask designs will vary in their ability to resist hazards and may have failure modes not considered in previous studies. Since no standard for performing a dry cask PRA exists, these issues will have to be addressed on a case-by-case basis. The applicability of the assumptions and limitations of previous studies to any future analysis will have to be carefully considered.

B.7.2 Number of Casks (N_{load} and N_{store})

The number of casks loaded, N_{load}, and stored, N_{store}, affects the total cask risk, R_{cask}. The number of casks loaded also affects the SFP risk, R_{sfp}, because of the potential for cask drops. As discussed above, cask loading is assumed to begin in Stage 2, increase during Stage 3 (expedited transfer), and, in Stage 4, return to a lower level necessary to maintain a low-density configuration in the pool.

NUREG-1353 (NRC, 1989a) and the EPRI report on spent fuel transfer (EPRI, 2012) include estimates for the number of casks loaded. NUREG-1353 initially assumes two casks are loaded per week resulting in 104 loads per reactor year. Using assumptions based on more updated information regarding the number of assemblies discharged per reload, the length of the fuel cycle, and the capacity of storage casks in use at the time, the analysis revised this estimate downward by a factor of 10. The EPRI report contains a more detailed analysis considering multi-unit sites and possible expedited loading scenarios. Dependent on these factors, the number of casks loaded annually was estimated to average 3 to 4 annually for current sites and up to 15 to 19 annually for some sites during expedited loading.

At the end of 2011, more than 1,500 casks had been loaded (EPRI). For a comparative risk calculation, site-specific information would have to be collected or estimated. Given the uncertainties in the calculation of risk per cask, and the fact that the risk of loading and storing casks has been estimated to be lower than the risk associated with SFPs, a precise number of casks loaded and stored is not expected to drive the results.

As a first approximation, one might assume that the total number of casks loaded from the SFP would be the same no matter the fuel management strategy. However, expedited fuel transfer requires loading fuel with a higher heat load into casks. Therefore, expedited fuel transfer may result in more casks being loaded with different accident consequences than the current package. The EPRI report estimates the increased number of casks required.

B.7.3 Pool Uncovery Frequency from Cask Drop Events (N_{load}, P_{drop} and P_{damage})

Heavy load drops have the potential to damage the SFP, possibly leading to uncovery of the fuel. In general, casks are considered to be the only loads handled over the pool heavy enough to have the potential to cause structural damage. Other heavy loads are usually prevented from traveling directly over the pool.

SFPs can have a variety of configurations affecting their susceptibility to cask drop events. Some pools contain cask loading pits with floors at a higher elevation than the bottom of the pool. Damage from a cask drop event would only drain the pool to a certain level, potentially giving operators sufficient time to align a makeup water source and continue keeping the fuel covered. The cask loading pit may be separated from the pool by a gate. In other pools, casks are loaded directly in the SFPs in a section which may or may not be reinforced to reduce the risk of damage in a cask drop event.

The total frequency of uncovery will be a function of how many casks are loaded, the estimated probability of a drop per loading, and the probability of damage given a drop. Expedited transfer of spent fuel will lead to increased cask loading for a number of years, increasing the risk of a dropped cask damaging the pool. The number of casks loaded is discussed above.

Several studies have addressed the issue of heavy load drops and the anticipated effect on the SFP. NUREG-1353 estimated a drop rate of 3.1×10^{-4} per reactor year assuming two lifts per week without consideration of Generic Safety Issue (GSI) A-36, “Control of Heavy Loads Near Spent Fuel.” The reduction in the probability of a cask drop for a plant which complies with the resolution of GSI A-36 was estimated to be a factor of 0.001 for a revised probability of $3.1\text{x}10^{-7}$ per reactor year. A LLNL analysis reported in NUREG/CR-5176 in support of NUREG-1353 considered worst-case cask drops on the pool wall for a boiling-water reactor and a pressurized-water reactor (Prassinos, 1989). The analysis concluded that it is likely that the liner would be severely damaged, so a value of 1 was used for P_{damage}. Based on updated information, NUREG-1353 judged two lifts per week (104 per year) to be an overestimate by about a factor of 10. The final estimate of the frequency of a cask being dropped and damaging the SFP is 3.1×10^{-8} with an upper bound estimate of 3.1×10^{-7}. This analysis only considered casks dropped on the SFP wall.

NUREG-1738 considered drops that would catastrophically fail the pool, leading to a rapid draindown and failure modes other than drops onto the SFP wall (NRC, 2001). The analysis assumed that only casks are heavy enough to cause catastrophic damage to the pool. Data from NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36,” issued July 1980, and NUREG/CR-5176 were combined with a calculation of the fraction of the load path spent over the pool and the fraction of the total path the load is lifted high enough to damage the pool to estimate the probability of a drop that damages the pool. For a nonsingle-failure-proof crane, the drop frequency was estimated, based on NUREG-0612 information, to have a mean value of 2.1×10^{-5} per year (using 100 lifts per year). For single-failure-proof cranes or plants that conform to the NUREG-0612 guidelines, the drop frequency was estimated to have a mean value of 2×10^{-7} per year (for 100 lifts per year). The analysis assumed that licensees with a non-single-failure-proof crane took appropriate mitigative actions to reduce the expected frequency of catastrophic damage to the same range as for facilities with a single-failure-proof crane.

NUREG-1864 used empirical drop data reported in NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," issued July 2003, to estimate the probability of dropping a cask. Three load drop events were identified from an estimated 54,000 lifts in the 1968–2002 time period, giving a probability of 5.6×10^{-5} per lift. This probability was considered conservative given that, of the three events, only one was a freefall while the other two were uncontrolled descents. The probability of pool damage was not estimated.

The EPRI dry cask PRA constructed a fault tree of a crane to address a range of factors and to account for specific crane features. Data from NUREG-0612 and other sources were used to estimate failure probabilities for basic components as well as human error probabilities. The final probability of a cask drop given a lift was estimated to be 5.3×10^{-6}. The probability of pool damage was not estimated.

Data cataloguing the susceptibility of SFPs to cask drops for the reactor fleet is not readily available. To address this issue, a risk analysis would need to either perform a site-specific analysis of cask drops, or conservatively assume that most (if not all) drops will damage the pool.

B.7.4 Pool Uncovery Frequency from Seismic Events ($F_{seismic}$)

The frequency of seismic events damaging the pool liner and leading to fuel uncovery depends on both the seismic hazard (i.e., the frequency of the initiating event) and the fragility of the SFP (i.e., the probability that the liner fails given that the event occurs). Chapter 3 of the SFPS report discusses the availability of seismic hazard information.

In contrast, seismic fragility data has not been characterized for most SFPs. NUREG/CR-5176 used a fragility analysis approach involving seismic loads derived from floor response spectra for the reactor building based on design response spectra. These loads were then combined with analytical methods for the calculation of the fundamental period of vibration of SFP floors and walls, as well as approximate methods for the calculation of the strength of these floors and walls. The approach used to derive the SFP fragility was generally consistent with methods used for seismic margin assessments at the time of that study. Since NUREG-1738 was not a site-specific analysis, an attempt was made to generalize this information. NUREG-1738 convolved a generic fragility (roughly corresponding to the fragilities calculated in NUREG/CR-5176) with EPRI and LLNL seismic hazard estimates to estimate the seismic risk. The study also developed a screening checklist such that a plant passing the checklist would have confidence of having a pool fragility of at least the assumed amount.

Finally, the SFPS performed a detailed analysis for the reference plant employing a combination of the approach used in NUREG/CR-5176 to estimate seismic loads with finite element analyses of the SFP structure to calculate hydrodynamic impulsive loads, nonlinear response mechanisms and strain concentrations in the liner. Chapter 4 of the SFPS report describes the structural analysis and estimated SFP performance. Chapter 10 provides a comparative assessment of the estimated performance for the SFP considered in the study with the performance of SFPs in recent major earthquakes in Japan including the SFPs at the Fukushima Daiichi nuclear power plant under the Tohoku earthquake of March 11, 2011.

The most robust way to estimate the seismic risk would be to utilize existing hazard estimates, and perform a site-specific fragility analysis. For some analyses, particularly those considering multiple sites, this may be time and cost prohibitive since the staff and licensees do not generally have fragility analyses of the pools. A generalized analysis for all plant sites would have to address the uncertainty in the variation of SFP responses to seismic events. One approach would be to use a conservative fragility estimate and to develop a checklist to ensure that the estimate is appropriate for the pools being considered.

B.7.5 Pool Uncovery Frequency from Other Events (F_{other})

As discussed in Section 1.5 of the SFPS report, the majority of SFP risk is believed to emanate from pool leakage events such as cask drops and seismic events discussed above, as well as events that preclude water injection for a long period of time (e.g. days). Table 1 in the SFPS report shows the frequency of fuel uncovery from various contributors calculated in NUREG-1353 and NUREG-1738.

B.7.6 Pool Consequences ($C_{uncovery}$)

The SFPS is the most detailed analysis to date of SFP consequences. As discussed in the SFPS report, the study was performed for a specific site for a specific initiating event. However, the consequence results will largely hold for other initiating events and may offer insights applicable to other sites. When applying the consequence results to one or several other sites, the assumptions used in the study, discussed in Chapter 2 of this report, must be considered along with which factors drove the results, discussed in Chapter 10 of this report. Some of these drivers and how they are expected to vary between plants are discussed below.

Once fuel in the pool has become uncovered, it may still be coolable from natural circulation of air, depending on the amount of decay heat and the amount of cooling. In the SFPS, the fuel is estimated to not be air coolable for 10 percent of the operating cycle. Factors affecting this include the amount of fuel in the pool, its configuration, burnup, geometry of the fuel racks, etc. A partial draindown event with channeled fuel could impede airflow and increase the time to coolability.

A significant driver of the amount of radioactivity released is whether a hydrogen combustion event occurs. The SFPS results predict these events in some high density loading situations, but not in any low density loading scenarios. It’s not clear whether this result will hold true for other reactor sites and what level of pool loading is sufficient to achieve this result. Furthermore, the SFPS did not consider hydrogen events from hydrogen originating from a concurrent reactor accident.

The consequence metric used will significantly affect the outcome of any comparative risk calculation. Comparisons of results using different consequence metrics are seen in Table 38 of the SFPS for high-density versus low-density fuel loading and are discussed in Chapter 10 for previous SFP and dry cask risk studies. These results demonstrate that the individual latent cancer fatality risk metric is relatively insensitive to changes in release magnitude for spent fuel accidents. Other metrics, such as land contamination, are much more affected by the amount of radioactivity released. Specific reasons for this are discussed in more detail in Sections 7.6 and Chapter 10.

Other site-specific factors that may affect the consequences of pool uncovery include SFP inventory, mitigative measures, and the surrounding population density and land value. The SFPS analysis for these aspects may have varying levels of applicability to other sites.

B.7.7 Other Spent Fuel Risk Considerations

Several additional events are not believed to significantly contribute to spent fuel risk. Dropping a single assembly is not expected to challenge the integrity of the pool, but may release some radiation. NUREG-1864 analyzes this event. Criticality events, which NUREG-1353 assumes not to be an issue, are considered in Section 3.6 of NUREG-1738. Although this report does not explicitly evaluate criticality events, Section 2.3 does discuss them.

B.8 Conclusions

This appendix discusses some of the information needed to perform a risk comparison of spent fuel storage strategies. NUREGs 1353, 1738 and 1864 provide much of the information for certain plants, and could be supplemented by the site-specific analyses described in the SFPS report. A complete comparison depends on several factors, including the relative, site-specific risks, and the time spent in each stage of fuel storage. The benefit of lower SFP risk from low density loading may be offset by increases in other risks, such as the risk from cask drop events damaging fuel in the cask or the SFP. However, the magnitude of that offset has not been completely calculated for any single plant. Additional work would be necessary to evaluate the applicability of existing information to a particular site or a group of sites.

APPENDIX C: COMMISSION AND ACRS CORRESPONDENCE

Letter from the Advisory Committee on Reactor Safeguards from April 25th 2012 (ML1208A216): In this letter the ACRS describes the results of a meeting between the Advisory Committee on Reactor Safeguards and the Office of Nuclear Regulatory Research, to the NRC Chairman.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

April 25, 2012

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SPENT FUEL POOL SCOPING STUDY

Dear Chairman Jaczko:

During the 593rd meeting of the Advisory Committee on Reactor Safeguards, April 12-14, 2012, we reviewed the methods and approaches being used in the Office of Nuclear Regulatory Research (RES) Spent Fuel Pool Scoping Study (SFPSS). Our Materials, Metallurgy, and Reactor Fuels and Reliability and PRA Subcommittees jointly reviewed the methods and approaches as well as preliminary results of this study on March 6, 2012. During these meetings, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS

1. The SFPSS is being performed in an organized and systematic manner, and is using modern NRC codes to evaluate the change in consequences from seismically induced spent fuel pool accidents with high and low-density loading.

2. The SFPSS consists of a detailed deterministic analysis of the consequences of a severe seismic event on a spent fuel pool at a single boiling water reactor (BWR) site.

3. The study will contribute to the technical basis for making decisions regarding expedited transfer of older irradiated spent nuclear fuel (SNF) from spent fuel pools.

BACKGROUND

The SFPSS was initiated in July 2011 and is planned for completion by June 2012. The staff was tasked with developing updated information on key aspects of potential spent fuel pool accident consequences on an aggressive one year schedule. The staff was directed to move expeditiously in a technically rigorous manner, using modern codes and methods.

To accomplish this task, the staff reviewed past consequence and risk assessments related to SNF storage, as well as other reports of relevance that have been developed by other organizations. The staff identified seismic hazard as the logical starting point to assess the continued applicability of past studies and to develop insights for the SFPSS. Depending on the results gained from the SPFSS, additional work may be appropriate to reach generally applicable conclusions for the U.S. BWR and pressurized water reactor (PWR) fleet.

Along with providing general updates to past information within the current operational and regulatory environment, the staff indicated that for the scenarios investigated, the study can address key questions and provide insights, such as:

- Do accident progression timelines for SNF pools proceed more slowly than previously thought?
- Do seismically induced station blackout scenarios contribute significantly to the overall consequences, or are these consequences dominated by seismically induced pool drain down?
- Do low-density loadings in spent fuel pools produce substantially different results in terms of public health effects and offsite consequences compared to high-density loadings?
- Do successful post event mitigation actions substantially reduce offsite consequences?

The staff indicated that answers to these questions are expected to be helpful in determining whether expedited transfers of SNF from pools to dry cask storage systems (DCSS) produce substantial safety benefits, thereby informing future regulatory decision making. Other ongoing efforts, such as planned Level 3 probabilistic risk analyses, will complement and build on this work.

DISCUSSION

The technical approach selected by the staff is focused on a detailed analysis of the spent fuel pool in a General Electric BWR-4 reactor at a single site during five phases of an operating cycle. Two conditions in the pool are considered: one representative of the current high-density loading in a relatively full SNF pool, and another representative of low-density loading in which older SNF has been removed to a dry cask storage facility. The elements of the study will include:

- seismic and structural assessments of the integrity of the pool and liner following seismic events with up to six times greater peak ground acceleration than the design basis safe shutdown earthquake (SSE) for the examined site;
- analysis of reactor building dose rates using the SCALE code package;

- accident progression analyses of fuel damage, fission product release, benefits of mitigation, and other effects using the MELCOR code modified to handle spent fuel pool accidents;
- emergency planning assessment;
- offsite consequences analyses of health effects and land contamination using the MACCS2 code; and
- probabilistic considerations.

The approach taken by the staff is capable of producing useful assessments of the consequences of severe seismic events on the structural integrity of the selected spent fuel pool design. The study is also capable of producing quantitative assessments of the safety benefits of low density fuel pool loading on the extent of fuel damage, land contamination, and off site health effects. However, since the study will not address the safety consequences of the same severe seismic events on cask loading, transportation, or long-term storage; the overall safety benefit will not be quantified. The possibility that there could be negative safety consequences associated the expedited loading, transfer, and long-term storage of possibly thousands of DCSS would need to be considered.

For the reasons noted above, the conclusions of the study may not be broadly applicable to the variety of reactor and pool designs in operation in the United States.

We look forward to a future review of the results and conclusions of the SFPSS.

Additional comments by ACRS Members J. Sam Armijo, Michael T. Ryan, Stephen P. Schultz, and Gordon R. Skillman are presented below.

Sincerely,

/RA/

J. Sam Armijo
Chairman

Additional Comments by ACRS Members J. Sam Armijo, Michael T. Ryan, Stephen P. Schultz, and Gordon R. Skillman

The staff's approach is rightly focused on the effects of severe seismic events on the structural integrity of U.S. spent fuel pools. Absent a failure of the pool structure and liner, there can be no rapid or uncontrollable draining of the pool, overheating and failure of the fuel cladding, release of radioactive fission products, and exposure to workers and the public. In the absence of pool failure and drain down, fuel cooling will be maintained in either the high density or low density loading scenarios to be studied in the SFPSS.

In view of the importance of pool structural integrity following seismic events, the SFPSS should be broadened to consider the performance of the spent fuel pools at the Fukushima Daiichi, Fukushima Daini, Onagawa, and Tokai sites following the severe Tohoku earthquake of March 2011, as well as the performance of the spent fuel pools at the Kashiwazaki-Kariwa site following the severe Chuetsu earthquake of July 2007. None of these pools suffered structural failure or drain down. The demonstrated robustness of the spent fuel pools at Fukushima Daiichi was noteworthy. These pools were subjected to the initial M9 earthquake, followed by several aftershocks greater than M7, and hundreds of lesser magnitude. In addition, the potentially weakened spent fuel pools in Units 1, 3 and 4 survived further structural loading from hydrogen explosions without significant damage or draining of the pool water. Although limited in scope, inspection of the fuel and sampling of the spent fuel pool water in the badly damaged Unit 4 revealed that the fuel had not suffered significant damage. By any reasonable standard, the performance of spent fuel pools protected the fuel from significant damage.

Since the spent fuel pools at Fukushima Daiichi were of the same design and vintage as the design chosen for the SFPSS, this broader approach could provide valuable data to confirm or correct the findings of the study.

REFERENCES

1. Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, NUREG-1738, February 2001(ML010430066)

2. RES Memorandum, Subject: Project Plan for Spent Fuel Pool Scoping Study, July 26, 2011 (ML111570370)

Letter from the NRC Staff (MLXXX): Letter was to the ACRS Chairman, Dr. J. Sam Armijo in response to his earlier letter to the Commission.

May 23, 2012

Dr. J. Sam Armijo, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS LETTER, DATED APRIL 25, 2012, ON THE SPENT FUEL POOL SCOPING STUDY

Dear Dr. Armijo:

I am responding to your letter of April 25, 2012, in which you provided the comments of the Advisory Committee on Reactor Safeguards (ACRS) on the staff's Spent Fuel Pool Scoping Study that was presented to the ACRS on April 12–14, 2012.

The staff agrees with ACRS' summary of the limitations of the study. The study results and the limitations will be considered with other factors when the staff evaluates the Fukushima Lessons Learned Tier 3 issue of whether spent fuel should be transferred to dry cask storage earlier than currently planned by the nuclear power plant licensees. As summarized in your letter, such factors include, but are not limited to, the risks of additional fuel handling when loading and transferring the spent fuel to the casks. The staff will also consider the relevant operating experience related to the integrity of spent fuel pools subjected to severe seismic events.

We appreciate the comments on the staff's plans for the study and look forward to further interactions on this topic.

Sincerely,

/RA Michael F. Weber for/

R. W. Borchardt
Executive Director
for Operations

cc: Chairman Jaczko
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
SECY

Staff Requirements Memo of July 16th 2012 (ML121980043): SRM directing staff to include additional scope to the Spent Fuel Pool Scoping Study report.

July 16, 2012

ML121980043

IN RESPONSE, PLEASE
REFER TO: M120607C

MEMORANDUM TO: Edwin M. Hackett, Executive Director
Advisory Committee on Reactor Safeguards

R. W. Borchardt
Executive Director for Operations

FROM: Andrew L. Bates, Acting Secretary ***/RA/***

SUBJECT: STAFF REQUIREMENTS – MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, 9:30 A.M., THURSDAY, JUNE 7, 2012, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS) to discuss the Committee's recent accomplishments and its ongoing and future activities. The ACRS presented updates on the following specific issues: 1. Spent Fuel Pool Scoping Study (SFPSS), 2. Implementation of Fukushima Recommendations, 3. State-of-the-Art Reactor Consequences Analyses (SOARCA), and 4. NRC Research Program.

As the ACRS noted in its April 25, 2012, letter on the SFPSS and reiterated during its meeting with the Commission, "since the study will not address the safety consequences of the same severe seismic events on cask loading, transportation, or long-term storage, the overall safety benefit will not be quantified. The possibility that there could be negative safety consequences associated with the expedited loading, transfer, and long-term storage of possibly thousands of DCSS [dry cask storage systems] would need to be considered."

The Office of Nuclear Regulatory Research should conduct a comparative assessment of SFPSS results against previous studies of safety consequences associated with loading, transfer, and long-term dry storage. These previous studies should be updated as necessary to conduct the comparative assessment.

The staff should also conduct a human reliability analysis focused on the capability to implement effective spent fuel pool cooling mitigating strategies, such as those required by 10 CFR 50.54(hh) or the recently issued Order EA-12-49, "Mitigation Strategies for Beyond-Design-Basis External Events."

In addition, the SFPSS should consider the evidence from the performance of the spent fuel pools during the real incidents identified in the additional comments by ACRS members in the April 25, 2012, letter.

The results of the SFPSS and the comparative assessment should be provided to the ACRS for its review, and subsequently provided to the Office of Nuclear Reactor Regulation for use in disposition of the Near-Term Task Force Tier 3 item on spent fuel storage, and sent to the Commission as an information paper after the staff has addressed the ACRS's comments.

cc: Chairman Macfarlane
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
OGC
CFO
OCA
OIG
OPA
Office Directors, Regions, ASLBP (via E-Mail)
PDR

APPENDIX D: REGULATORY ANALYSIS AND BACKFITTING DISCUSSION TO DETERMINE THE SAFETY BENEFIT OF EXPEDITED TRANSFER OF SPENT FUEL AT A REFERENCE PLANT

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

ABBREVIATIONS AND ACRONYMS

ADAMS	Agencywide Documents Access and Management System
Bq	Becquerel
BLS	Bureau of Labor Statistics
BWR	boiling-water reactor
CDF	Core Damage Frequency
CFR	*Code of Federal Regulations*
CoC	certificate of compliance
Cs	cesium
DOE	U.S. Department of Energy
DSC	dry storage cask systems
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
FR	*Federal Register*
FTE	Full-Time Equivalent
ISFSI	independent spent fuel storage installation
LCF	latent cancer fatality
LERF	Large Early Release Frequency
LNT	linear no-threshold
LOOP	loss of offsite power
MACCS2	MELCOR Accident Consequence Code System, Version 2
MELCOR	not an acronym
NPV	Net Present Value
NRC	Nuclear Regulatory Commission
NTTF	Near-Term Task Force
OCP	operating cycle phase
OMB	Office of Management and Budget
PAG	protective action guides
PGA	peak ground acceleration
PRM	petition for rulemaking
RA	Regulatory Analysis
SFP	spent fuel pool
SRM	Staff Requirements Memorandum
USGS	United States Geological Survey

D.1 INTRODUCTION

This appendix, which is organized into five sections, presents the regulatory analysis and backfitting discussion to determine the safety benefit of expedited transfer of spent fuel at a reference plant:

- Section D.1 describes the nature of the problem and a clear statement of the objective of the proposed action.
- Section D.2 describes and clearly explains the alternative approaches considered.
- Section D.3 describes the attributes affected, the methodology used to evaluate benefits and costs, the analysis model, key data and assumptions, and results for the alternatives evaluated.
- Section D.4 presents the analytical results and findings including discussion of supplemental considerations, uncertainties in estimates, and results of sensitivity analyses on the overall costs and benefits.
- Section D.5 presents the preferred alternative and the basis for selection, discusses any decision criteria used, identifies and discusses the regulatory instrument to be used (as applicable), and explains the statutory basis for the action.

D.1.1 Statement of the Problem

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 spent fuel pools is safe and that the risk is appropriately low. These studies used simplified and sometimes bounding assumptions and models to characterize the likelihood and consequences of beyond-design-basis spent fuel pool accidents[46]. As part of NRC's post-9/11 security assessments, spent fuel pool 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 issued a new regulation in 2009, 10 CFR 50.54(hh)(2), that requires reactor licensees to develop and implement guidance and strategies intended, in part, to maintain or restore spent fuel pool cooling capabilities following certain beyond design basis events.

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 (73 FR 46204, August 8, 2008) 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 Federal government's decision to stop work on a deep geologic repository at Yucca Mountain and the events in Japan following the March 2011 earthquake has rekindled public and industry interest in understanding the consequences from postulated accidents associated with high-density spent fuel pool storage and the relative benefits of low-

[46] An overview of previous studies is provided in section 10.2 to the Spent Fuel Pool Study.

density spent fuel pool storage. This study provides an analysis of the health and safety benefits, if any, from moving from high-density to low-density spent fuel pool storage. This appendix assesses the costs and benefits of this activity, and then assesses whether the benefits are cost justified and substantial enough to justify a backfit to impose these requirements on the reference plant.

In response to these recent events, the staff has determined that it should confirm that high-density spent fuel pool configurations continue to provide adequate protection, and assess whether any safety benefits (or detriments) would occur from expedited transfer of spent fuel to dry cask storage.

The purpose of this regulatory analysis is to help ensure that:

- Appropriate alternatives to regulatory objectives are identified and analyzed.

- No clearly preferable alternative is available to this action.

- The costs of implementation are justified by its effect on overall protection of the public health and safety.

D.1.2 Objective of Proposed Action

Following the March 2011 accident at the Fukushima Daiichi nuclear power plant in Japan that resulted after the Tohoku Earthquake and subsequent tsunami, several stakeholders submitted comments to the Commission and staff requesting that regulatory action be taken to require the expedited transfer of spent fuel stored in spent fuel pools to dry cask storage. The rationale was that transferring the spent fuel to dry storage would lessen the potential consequences associated with a loss of spent fuel pool coolant inventory by decreasing the amount of spent fuel stored in these pools and thereby decreasing the heat generation rate and radionuclide source term associated with the spent fuel in pool storage.

As directed by the Commission in SRM-SECY-12-0025, dated March 9, 2012, the staff has undertaken regulatory actions that originated from the NTTF recommendations to enhance reactor and spent fuel pool safety. On March 12, 2012, the staff issued Order EA-12-051, which requires that licensees install reliable means of remotely monitoring wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event. In addition, the staff issued Order EA-12-049 which requires that licensees develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities following a beyond design basis external event. Upon full implementation of these Orders, spent fuel pool safety will be significantly increased.

While the staff has concluded, based on previous studies without these enhancements, that both spent fuel pools and dry casks provide adequate protection of public health and safety, the staff has determined that it should confirm that both spent fuel pools and dry cask storage continue to provide adequate protection.

This analysis uses information contained within the “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor” (Spent Fuel Pool Study or main document), to evaluate whether there is a benefit at the reference plant in the study to change from high- to low-density storage configurations in the spent fuel pool.

This analysis calculates the potential benefit per reactor year resulting from expedited fuel transfer by comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage. The comparison uses the initiating frequency and consequences from the Spent Fuel Pool Study as an indicator of any changes in the NRC's understanding of safe storage of spent fuel. The staff also used calculated results from previous spent fuel pool studies (i.e., NUREG-1353 and NUREG-1738) to extend the applicability of this evaluation to include other initiators, which could challenge spent fuel pool cooling or integrity.

D.2 IDENTIFICATION AND PRELIMINARY ANALYSIS OF ALTERNATIVE APPROACHES

This section presents the analysis of the alternatives that the NRC considered to meet the regulatory goals identified in the previous section. The NRC considered the regulatory baseline and one alternative to change this baseline as discussed below.

D.2.1 Alternative 1 – Regulatory Baseline – Maintain the Existing Spent Fuel Storage Requirements

This alternative reflects a Commission decision not to expedite the storage of spent fuel to dry cask storage, but to continue with NRC's existing licensing requirements for spent fuel storage. Under this alternative, spent fuel is moved into dry storage only as necessary to accommodate fuel assemblies being removed from the core during refueling operations. It also assumes that all applicable requirements and guidance to date have been implemented, but no implementation is assumed for related generic issues or other staff requirements or guidance that is unresolved or still under review.

The condition represented by this alternative is the storage of spent fuel in high-density racks[47] in the spent fuel pool, a relatively full spent fuel pool, compliance with all current regulatory requirements including those requirements associated with the following:

- 10 CFR 50.54(hh)(2) with respect to spent fuel configuration, and spent fuel pool preventive and mitigative capabilities,
- Order EA-12-051 that requires licensees to install reliable means of remotely monitoring wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event, and
- Order EA-12-049 that requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities following a beyond design basis external event.

Furthermore, because spent fuel pools have a limited amount of available storage, even after licensees expanded their storage capacity using high-density storage racks, the current practice

[47] Most nuclear power plant spent fuel pools were originally designed for temporary storage of spent fuel. Starting in the 1980s, most pools were "re-racked" to utilize hardware that stores the assemblies in a more closely-spaced arrangement, thus allowing the storage of more assemblies in a high-density configuration.

of transferring spent fuel to dry storage in accordance with 10 CFR 72 is assumed to continue.[48] This alternative represents the baseline for estimating the incremental costs of alternative 2.

D.2.2 Alternative 2 – Low-density Spent Fuel Pool Storage

Under this alternative, older spent fuel assemblies[49] are expeditiously moved from spent fuel pool storage to dry cask storage beginning in year 2014 to achieve and maintain a low-density loading of spent fuel in the existing high-density racks within five years. The situation where the spent fuel pool is re-racked to a low-density rack configuration was not evaluated because such a situation would be inefficient in terms of regulatory benefit given that much of the benefit could be achieved by storing less fuel in the existing high-density racks. Because of the low-density spent fuel pool loading, this alternative has less longer-lived radionuclide inventory in the spent fuel pool, a lower overall heat load in the pool, and a slight increase in the initial water inventory that displaces the removed spent fuel assemblies. In certain situations, this additional water could delay the clearing of the baseplate, which would temporarily inhibit natural air circulation cooling under and up through the racks should the spent fuel pool completely drain.

Due to the uncertainty associated with the schedule for the availability of a spent fuel repository, the reference plant has a plan to have sufficient on-site storage capacity (in-pool capacity and dry storage) to store all of the spent fuel discharged over the operating life of the plant until sufficient repository capacity becomes available. As a result, the analyzed incremental increase in costs results primarily from the increase in net present value cost for the early transfer of spent fuel into dry storage resulting from the earlier capital costs for new casks and for a dry storage facility.

The staff recognizes that there are cost and risk impacts associated with the transfer of spent fuel from the spent fuel pool to cask storage after five years of cooling and during long-term cask storage[50]. These cost and risk impacts, if included, would reduce the overall net benefit in relation to the regulatory baseline. These effects (e.g., the added risks of handling and moving casks) were conservatively ignored to calculate the potential benefit per reactor year by only comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and its implementation costs.

D.3 <u>ESTIMATION AND EVALUATION OF VALUES AND IMPACTS</u>

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[48] Maintenance of the existing spent fuel pool storage requirements would not limit the Commission's authority to add new requirements or update regulatory guidelines, as necessary. These actions and activities are a part of the regulatory baseline. However, these activities would be pursued as separate regulatory actions to resolve particular technical issues. Under this alternative, the NRC would take no action to require facilities to expedite the movement of spent fuel to achieve low-density loading in the spent fuel pool.

[49] Older spent fuel assemblies are those that have been placed in the spent fuel pool to cool for at least five years after discharge from the reactor core.

[50] EPRI report TR-1021049 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.

This section discusses the benefits and costs of each action alternative relative to the baseline. Ideally, all costs and benefits are converted into monetary values. The total of benefits and costs are then algebraically summed to determine for which alternative the difference between the values and impacts was greatest. However, in some cases the assignment of monetary values to benefits is not provided because meaningful quantification is not possible.

D.3.1 Identification of Affected Attributes

This section identifies the factors within the public and private sectors that the regulatory alternatives (discussed in section D.2) are expected to affect. These factors are classified as attributes using the list of potential attributes provided by the NRC in Chapter 5 of its Regulatory Analysis Technical Evaluation Handbook. The basis for selecting each attribute is presented below.

Affected attributes are the following:

- <u>Public Health (Accident)</u>. This attribute measures expected changes in radiation exposure to the public due to changes in accident frequencies or accident consequences associated with the proposed action. The expected changes in radiation exposure are measured over a 50-mile radius from the plant site. The dose to the public is from reoccupation of the land and other activities following a severe accident. In addition, the dose to the public includes the occupational dose to workers for cleanup and decontamination of the contaminated land not onsite. The calculation for each alternative is made by subtracting the alternative from the regulatory baseline.

- <u>Occupational Health (Accident)</u>. This attribute measures occupational health effects, for both immediate and long-term, associated with site workers because of changes in accident frequency or accident mitigation. Within the regulatory baseline, the short-term occupational exposure related to the accident occurs at the time of the accident and during the immediate management of the emergency and during decontamination and decommissioning of the onsite property. The radiological occupational exposure resulting from cleanup and refurbishment or decommissioning activities of the damaged facility to occupational workers are found within the long-term occupational exposure.

- <u>Occupational Health (Routine)</u>. This attribute accounts for radiological exposures to workers during normal facility operations (i.e., non-accident situations). These occupational exposures occur during DSC loading and handling activities; ISFSI operations, maintenance, and surveillance activities; and preparing to ship the spent fuel offsite.

  This attribute represents an estimate of health effects incurred during normal facility operations so accident probabilities are not relevant. As is true of other types of exposures, a net decrease in worker exposures is taken as positive; a net increase in worker exposures is taken as negative. This exposure is also subject to the dollar per person-rem conversion factor.

- <u>Offsite Property</u>. This attribute measures the expected total monetary effects on offsite property resulting from the proposed action. Changes to offsite property can take various forms, both direct, (e.g. land, food, and water) and indirect (e.g. tourism). This attribute is typically the product of the change in accident frequency and the property consequences from the occurrence of an accident.

For the regulatory baseline, the offsite property costs are any property consequences resulting from any radiological release from the occurrence of an accident. Normal operational releases and those releases before severe accident are outside the scope of this regulatory analysis.

- Onsite Property. This attribute measures the expected monetary effects on onsite property, including replacement power costs, decontamination, and refurbishment costs, from the proposed action. There are two forms of onsite property costs that are evaluated. The first type is the cleanup and decontamination costs for the unit. The second type is the cost to replace the energy from the damaged or shutdown units.

- Industry Implementation. This attribute accounts for the projected net economic effect on the affected licensees to implement the mandated changes. Costs include procedural and administrative activities. Additional costs above the regulatory baseline are considered negative and cost savings are considered positive.

- Industry Operation. This attribute accounts for the projected net economic effect due to routine and recurring activities required by the proposed alternative on all affected licensees.

- NRC Implementation. This attribute accounts for the projected net economic effect on the NRC to place the proposed alternative into operation. NRC implementation costs and benefits incurred in addition to those expected under the regulatory baseline are included. Additional rulemaking, policy statements, new or expedited revision of guidance documents, and inspection procedures are examples of such costs.

- NRC Operation. This attribute accounts for the projected net economic effect on the NRC after the proposed action is implemented. Additional inspections, evaluations, or enforcement activities are examples of such costs.

Attributes that are not expected to be affected under any of the alternatives include the following: public health (routine), other government, general public, antitrust considerations, safeguards and security considerations, regulatory efficiency, improvements in knowledge, and environmental considerations.

D.3.2 Methodology Overview

This section describes the process used to evaluate benefits and costs associated with the proposed regulatory framework alternatives. The benefits (values) include desirable changes in affected attributes (e.g., monetary savings and improved security and safety). The costs (impacts or burdens) include undesirable changes in affected attributes (e.g., increased monetary costs, and decreased security and safety).

The regulatory analysis methodology is specified by various guidance documents. The two documents that govern the NRC's voluntary regulatory analysis process are NUREG/BR-0058, Revision 4, "Regulatory Analysis (RA) Guidelines of the U.S. Nuclear Regulatory Commission," dated September 2004 (RA Guidelines), and NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," dated January 1997 (RA Handbook). The regulatory analysis identifies all attributes impacted by the proposed alternative and analyzes them either quantitatively or qualitatively as described in the previous section.

For the quantified regulatory analysis, the NRC staff develops expected values for each cost and benefit. The expected value is the product of the probability of the cost or benefit occurring and the consequences that would occur assuming the event actually happens. For each alternative, the staff first determines the probabilities and consequences for each cost and benefit, including the year the consequence is incurred. The NRC staff then discounts the consequences in future years to the current year of the regulatory action. Finally, the NRC staff sums the costs and the benefits for each alternative and compares them.

After performing a quantitative regulatory analysis, the NRC staff adds attributes that could only be qualified[51]. Based on the qualification of each attribute, uncertainties, sensitivities, and the quantified costs and benefits, the staff makes a recommendation for each alternative. If the benefits, both quantified and qualified, are greater than the quantified and qualified costs, then the staff recommends the alternative should be implemented. If the benefits, both quantified and qualified, are less than the quantified and qualified costs, then the staff recommends the alternative should not be implemented.[52]

D.3.2.1 Analysis Model

This regulatory analysis measures the incremental impacts of the proposed regulatory framework alternative to the "continue with the existing regulatory framework" baseline, which reflects anticipated behavior in the event that the proposed alternatives are not adopted. Section D.4 presents the estimated incremental costs and savings of each alternative relative to continuing with NRC's existing regulatory framework (alternative 1).

Key inputs into the analysis model are discussed in the following subsections.

D.3.2.1.1 Baseline for the Analysis

The regulatory baseline used in the analysis is to continue with NRC's existing approach to spent fuel pool storage. This baseline assumes full compliance with existing NRC requirements, including current regulations and relevant orders. This is consistent with NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Rev. 4, which states that, "in evaluating a new requirement…, the staff should assume that all existing NRC and Agreement State requirements have been implemented."

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[51] See NRC's Regulatory Analysis Technical Evaluation Handbook, Section 4.3, "Estimation and Evaluation of Values and Impacts."

[52] See NRC's Regulatory Analysis Technical Evaluation Handbook, Section 4.5, "Decision Rationale." Non-quantifiable attributes can only be factored into the decision in a judgmental way; the experience of the decisionmaker will strongly influence the weight that they are given. Qualitative attributes may be significant factors in regulatory decisions and should be considered, if appropriate.

D.3.2.1.2 Discount Rates

In accordance with guidance from the Office of Management and Budget (OMB) and NUREG/BR-0058, Rev. 4, present-worth calculations are used to determine how much society would need to invest today to ensure that the designated dollar amount is available in a given year in the future. By using present-worth, costs and benefits, regardless of when averted in time, are valued equally. Based on OMB guidance Circular No. A-4, September 17, 2003, present-worth calculations are presented using both 3 percent and 7 percent real discount rates. The 3 percent rate approximates the real rate of return on long-term government debt, which serves as a proxy for the real rate of return on savings. This rate is appropriate when the primary effect of the regulation is on private consumption. Alternatively, the 7 percent rate approximates the marginal pretax real rate of return on an average investment in the private sector, and is the appropriate discount rate whenever the main effect of a regulation is to displace or alter the use of capital in the private sector.

Although the NRC is not bound to follow OMB guidance, the NRC has voluntarily complied with the present-worth calculations developed in OMB Circular No. A-4 and has stated so in the RA Guidelines and RA Handbook.

D.3.2.2 Data

The data and assumptions used in analyzing the quantifiable impacts associated with the proposed alternative are discussed in this subsection. Information on attributes affected by the proposed regulatory framework alternatives were obtained from experienced NRC staff and other sources as referenced. The NRC staff considered the potential differences between the new requirements and the current requirements and has incorporated the proposed incremental changes into this regulatory analysis.

Available cost information is included in the backfitting analysis of the reference plant. The NRC plans to use the insights from this analysis to inform a broader regulatory analysis to support decisionmakers in determining whether NRC's regulations should be changed to impose new generic requirements on all operating nuclear reactors.

D.3.2.2.1 Spent Fuel Pool Initiator Release Frequency

Section 1.5 of the Spent Fuel Pool Study provides an overview of contributors to spent fuel pool risk. The majority of spent fuel pool risk emanates from a loss of water from a sizeable leak in the spent fuel pool or a boil off in which operator action to inject water into the pool for an extended period is precluded. The release frequency from the spent fuel pool can then be characterized as the frequency of the initiator causing fuel uncovery multiplied by the probability of a release given fuel uncovery for the specific initiating event. The total release frequency is the sum of the frequency of releases from cask drops, seismic events, and other initiators. This value is given by:

Where $F_{initiator}$ includes

F_{drop}	=	frequency of spent fuel uncovery from cask drops
$F_{seismic\text{-}bin\ 3}$	=	frequency of spent fuel uncovery from seismic bin 3 event
$F_{seismic\text{-}bin\ 4}$	=	frequency of spent fuel uncovery from seismic bin 4 event

F_{other} = frequency of spent fuel uncovery from sources other than cask drops and seismic

$P_{release}$ = probability of release given spent fuel uncovery for specific initiators

Source: Derived from Spent Fuel Pool Study, section B.4.

The Spent Fuel Pool Study provides a detailed analysis of the consequences, for a particular site and a calculation of $F_{seismic}$ for seismic bin 3, a hazard exceedance frequency range provided in Table 4 of the Spent Fuel Pool Study and reproduced in Table 72.

Table 72 Seismic Bins and Initiating Event Frequencies

Bin No.	Bin Range (g)	Bin PGA (g)	Approximate Initiating Event Frequency (USGS 2008 model) (/yr)
1	0.05 - 0.3	0.12	5.2×10^{-4}
2	0.3 - 0.5	0.4	2.7×10^{-5}
3	0.5 - 1.0	0.7	1.7×10^{-5}
4	> 1.0	1.2	4.9×10^{-6}

The Spent Fuel Pool Study did not analyze initiators that contribute to spent fuel pool risk other than for seismic events defined by seismic bin no. 3. However past studies, such as NUREG-1353 and NUREG-1738, evaluated additional events that could contribute to risk and consequences from spent fuel pool fires. Table 74 summarizes these initiating-event-class fuel uncovery frequencies. Uncovery frequencies taken from past studies depend on the assumptions stated in those studies. Additionally, seismic bin no. 4 is included by extrapolating the results of this study. For seismic bin no. 3 and bin no. 4 events, the uncovery frequency is the product of the initiating event frequency, ac power fragility, and the liner fragility.

The main report uses an ac power fragility value of 0.84 taken from NUREG-1150 as a surrogate for the conditional probability of normal spent fuel pool cooling and makeup not being available following a 0.7g earthquake. This simplifying assumption was made in light of the fact that the main report is not a probabilistic risk assessment (but rather a consequence analysis with probabilistic considerations) and that this value already approximates the upper bound value of 1.00. For the seismic bin no. 4 event, ac power fragility upper bound value of 1.00 was used in this regulatory analysis. In reality, the availability of normal spent fuel pool cooling and makeup would be a combination of the ac power fragility, the fragility of the actual equipment and its support equipment, and operator actions to recover spent fuel pool cooling capabilities using additional mitigation equipment and strategies implemented in response to Order EA-12-049, which were not considered in the main report. The modeling and consideration of these guidance and strategies to maintain or restore spent fuel pool cooling capabilities following a beyond design basis external event could result in a smaller value for spent fuel pool cooling and makeup failure conditional probability than the values used here and a resulting smaller initiating event fuel uncovery frequency.

Section 4.1.5 of the main report describes the results from the nonlinear finite element analysis to estimate the likelihood of leakage from concrete cracking and related spent fuel pool liner failure for the 0.7g earthquake. Figure 27 shows that the maximum membrane effective strain is about 3.7 percent. Based on this calculated liner strain for the 0.7g earthquake, a structural analysis of the pool estimates that the spent fuel pool in this study has a 90% probability of surviving the 0.7g earthquake with no liner leakage (or conversely, a 10% probability of damaging the liner such that leakage will occur). As a result, a liner fragility value of 0.1 is used for the seismic bin no. 3 initiating event. For the seismic bin no. 4 initiating event (i.e., 1.2g earthquake), a comparable structural analysis was not performed to determine the liner fragility value. As detailed in section 4.1.1 of the main report, the specific conditions for liner failure vary according to site conditions and spent fuel pool design. NUREG-1353 predicted the likelihood of liner failure from all potential earthquakes to be between about two and six times in a million years. NUREG-1738 predicted the likelihood of liner failure from all potential earthquakes to be between about two times in a million years and two times in 10 million years. Because a documented liner fragility value for a 1.2g earthquake for the reference plant is not readily available, a conservative bounding approach was used. A liner fragility value of 1.00 is used in this regulatory analysis for the best estimate, even though a realistic analysis may be able to justify a value a factor of 2 or more lower.

Past studies have reached generally similar conclusions about the relative contribution to risk from the seismic initiating events considered. Table 73 Frequency of Spent Fuel Pool Fuel Uncovery for Seismic Events summarizes the impact of the above modeling assumptions when comparing the seismic initiating event fuel uncovery frequencies from previous spent fuel pool accident regulatory analyses.

Table 73 Frequency of Spent Fuel Pool Fuel Uncovery for Seismic Events

Reference	Spent Fuel Pool Fuel Uncovery (per reactor-year)	Percent Increase in Fuel Uncovery Frequency Value
NUREG-1353 (1989) (BWR, best estimate)[1]	$7x10^{-6}$	(10%)
NUREG-1738[2]	$2x10^{-6}$	315%
This regulatory analysis[3]	$6.3x10^{-6}$	100%

1. This number was not multiplied by the stated conditional probability of having a zirconium fire of 0.25.
2. NUREG-1738 presented results for the two different seismic hazard models in wide use at the time (the Electric Power Research Institute and Lawrence Livermore National Labs models). The larger of the two values is listed above.
3. The initiating event frequency values are from Table 72. The likelihood of fuel uncovery is a product of initiating event frequency (e.g., $1.6x10^{-5}$ for seismic bin no. 3), ac power fragility (0.84), and liner fragility (0.1). For seismic bin no. 4, the likelihood of fuel uncovery is a product of initiating event frequency ($4.9x10^{-6}$), ac power fragility of 1.0, and a liner fragility of 1.0 (i.e., 100-percent likelihood of ac power and pool liner failure).

As discussed in the SFPS report, the study was performed for a specific site and for a specific initiating event. Once fuel in the pool has become uncovered, it may still be coolable from natural circulation of air, depending on the amount of decay heat and the amount of cooling. In section 12.1 of the main report, the fuel is estimated to be air coolable for all but roughly 10 percent of the operating cycle. Factors affecting this value include the amount of fuel in the pool, its configuration, burnup, geometry of the fuel racks, etc. A partial draindown event with channeled fuel could impede airflow. In this case with no natural circulation of air through the racks, the cooling of the fuel by the spray flow would be the only effective cooling mechanism until the decay heat of the fuel is reduced.

For the seismic bin no. 4 event, the spent fuel is assumed to retain an air coolable geometry following this event that causes a moderate to large crack in the pool and results in full pool draindown. Information provided in NUREG/CR-5176 (Prassinos et al, 1989), which concludes that there is high confidence that spent fuel pool racks are sufficiently robust to remain generally intact with their fuel channels open supports this assumption. Furthermore, prior studies conclude that severe earthquakes are not expected to result in catastrophic failure of spent fuel pool structural walls and floor or fuel racks. Section 4.2 of this study cites median fragility for the reactor building of about 1.6g. However, the main report did not perform a structural analysis to verify that the reference plant spent fuel and racks retain their structural integrity and air-coolable geometry following a 1.2g peak ground acceleration seismic event. Given the uncertainties involved, a bounding approach was used to evaluate the sensitivity of assuming the spent fuel is not air-coolable following a seismic bin no. 4 earthquake that causes a rapid draindown of the spent fuel pool. This was done by assuming a value of 1.0 for the high estimate of the conditional probability of release for the seismic bin no. 4 unsuccessful mitigation event.

For the cask drop event, spent fuel is assumed to retain an air coolable geometry because a postulated cask drop accident would most likely affect the fuel pool floor in the cask loading area. The overhead crane used to move the casks is designed to meet single failure proof criteria, and has interlocks and administrative controls that limit the motion of the crane over the spent fuel pool to the cask loading area, where no fuel is stored. Although improbable, crane failure is more likely to occur during hoisting operations when many components contribute to holding the cask than during translational motion when the hoist holding brakes are set. The hoisting activities occur over the cask loading area, and, in that location, the cask, if dropped, could have sufficient potential energy to damage the spent fuel pool floor. . However, the main report did not perform a structural analysis to verify that the reference plant spent fuel and racks retain their structural integrity and air-coolable geometry following a cask drop event. Given the uncertainties involved, a bounding approach was used to evaluate the sensitivity of assuming the spent fuel is not air-coolable following a cask drop accident. This was done by assuming a value of 1.0 for the high estimate of the conditional probability of release for the cask drop unsuccessful mitigation event.

To calculate the total release frequency, the uncovery frequencies are multiplied by the conditional probability of release for each initiating event class. The conditional probability of release depends on the fraction of the operating cycle where the fuel is not air coolable. For the seismic bin no. 3 event analyzed in the SFPS, this was calculated to be the ratio of 60/730 days or 8.2% of the operating cycle. See Section 5.6.3 of the main document for further discussion. For the non-seismic and non-cask drop events taken from previous studies, the nature of the events may lead to a situation similar to a partial draindown where the rack baseplate is not cleared and airflow is impeded. For these events, the conditional release probability is assumed to be 100%.

When mitigation is credited, this study found that successful mitigation decreased the conditional probability by a factor of 19 for the seismic bin no. 3 event analyzed using mitigation measures required under 10 CFR 50.54 (hh)(2). The main report does not consider the post-Fukushima mitigation equipment and mitigation strategies for their use required under Order EA-12-049 and being implemented by the plants that are intended to increase the likelihood of restoring or maintaining power and mitigation capability during severe accidents. For the purposes of this regulatory analysis, it was assumed that successful mitigation decreased the conditional probability by a factor of 19 for all initiating events as determined in the main report. In reality, the effectiveness of post-Fukushima improvements to severe accident mitigation measures will depend on a variety of factors, which the SFPS did not consider, and which are expected to be more effective than what is assumed here. Although the likelihood of successful mitigation deployment is uncertain.

Table 74 summarizes the initiating event class fuel uncovery frequencies, the conditional probability of release, and the total release frequency with and without mitigation.

Table 74 Release Frequencies for Spent Fuel Pool Initiators

Spent fuel loading configuration		1x4		1x4	
Initiating Event Class	**Initiating Event Fuel Uncovery Frequency (per r-yr)**	**Conditional Probability of Release (Unsuccessful mitigation)**	**Release Frequency (Unsuccessful mitigation) (per r-yr)**	**Conditional Probability of Release (successful mitigation)**	**Release Frequency (successful mitigation) (per r-yr)**
Seismic bin no. 3 event	1.4×10^{-6} (3)	8.2%	1.18×10^{-7}	0.43% (4)	6.18×10^{-9}
Seismic bin no. 4 event	4.9×10^{-6} (3)	8.2% – 100%	4.03×10^{-7} – 4.9×10^{-6}	0.43% (4)	2.12×10^{-8} – 2.58×10^{-7}
Cask / heavy load drop	2×10^{-7} (2)	8.2% – 100%	1.64×10^{-8} – 2×10^{-7}	0.43% (4)	8.65×10^{-10} – 1.05×10^{-8}
LOOP – severe weather	1×10^{-7} (2)	100%	1.00×10^{-7}	0.43% (4)	5.26×10^{-9}
LOOP – other	3×10^{-8} (2)	100%	3.00×10^{-8}	0.43% (4)	1.58×10^{-9}
Internal fire	2×10^{-8} (2)	100%	2.00×10^{-8}	0.43% (4)	1.05×10^{-9}
Loss of pool cooling	1.5×10^{-8} (1)	100%	1.50×10^{-8}	0.43% (4)	7.89×10^{-10}
Loss of coolant inventory	3×10^{-9} (2)	100%	3.00×10^{-9}	0.43% (4)	1.58×10^{-10}
Inadvertent aircraft	3×10^{-9} (2)	100%	3.00×10^{-9}	0.43% (4)	1.58×10^{-10}
Missiles – general	2.5×10^{-9} (1)	100%	2.50×10^{-9}	0.43% (4)	1.32×10^{-10}
Missiles - tornado	1×10^{-9} (2)	100%	1.00×10^{-9}	0.43% (4)	5.26×10^{-11}
Pneumatic seal failures	n/a (5)				
Total			7.11×10^{-7} – 5.39×10^{-6}		3.74×10^{-8} – 2.84×10^{-7}

1. Values from NUREG-1353. These numbers were multiplied by the stated conditional probability of having a zirconium fire of 0.25.
2. Values from NUREG-1738
3. Initiating event frequency values from Spent Fuel Pool Study, Table 4. The likelihood of fuel uncovery is a product of initiating event frequency (e.g., 1.6×10^{-5} for seismic bin no. 3), ac power fragility (0.84), and liner fragility (0.1). For seismic bin no. 4, the likelihood of fuel uncovery is a product of initiating event frequency (4.9×10^{-6}), ac power fragility of 1.0, and a liner fragility of 1.0 (e.g., 100-percent likelihood of ac power and pool liner failure).
4. The conditional probability of release with successful mitigation with deployed 50.54(hh)(2) equipment is the quotient of OCP probability (60/730 or 8.2%) divided by the mitigation benefit in reducing the release likelihood (factor of 19). See Section 5.6.3 of the main document for further discussion. Additional mitigation equipment and mitigation strategies under Order EA-12-049 would further enhance the likelihood of successful mitigation, thereby further reducing the value for the conditional probability of release with successful mitigation.
5. As discussed in Table 3 of the main report, the reference plant has gates with mechanical seals to prevent leakage. These seals are kept under pressure by passive mechanical means (i.e., do not depend on air pressure, ac power, or dc power). Therefore, pneumatic seal failures are not applicable for the reference plant.

Based on this information, the values used in this regulatory analysis for $F_{release}$ is are summarized in Table 75.

Table 75 Spent Fuel Pool Release Frequency Estimates

Parameter	Unsuccessful mitigation			Successful mitigation		
	Low	Best	High	Low	Best	High
$F_{release}$	7.11×10^{-7}		5.39×10^{-6}	3.74×10^{-8}		2.84×10^{-7}

These release frequency values are subject to the assumption of unsuccessful deployment of mitigation and the other assumptions contained in this analysis and those stated in Table 3 of the main report. A comparison of the release frequencies (total and delta) used in this regulatory analysis to the release frequencies used for only seismic bin no. 3 in the Spent Fuel Pool Study is provided in Table 76.

Table 76 Release Frequency Comparison Between Inclusion of All Initiator Event Classes to the Seismic Bin No. 3 Event

Mitigation Case	Release Frequency for All Initiator Events Classes (per r-yr)	Release Frequency for the Seismic Bin No. 3 Event (per r-yr)	Percent Increase in Release Frequency
Unsuccessful Mitigation	7.11×10^{-7} – 5.39×10^{-6}	1.18×10^{-7}	505% – 4489%
Successful Mitigation	3.74×10^{-8} – 2.84×10^{-7}	6.18×10^{-9}	505% – 4489%
Delta change	6.74×10^{-7} – 5.11×10^{-6}	1.11×10^{-7}	505% – 4489%

D.3.2.2.2 Duration of On-site Spent Fuel Storage Risk

The reference plant operating license expires in 2034. For this analysis, it is assumed that the plant operates through the term of its operating license and that the licensee continues to store spent fuel in the pool following commercial operation[53] to allow the spent fuel to cool sufficiently before placing into dry storage. For all cases analyzed, it was assumed that spent fuel stored in the spent fuel pool is susceptible to the risk of spent fuel fires for up to one year after permanent cessation of operations.

D.3.2.2.3 Cost/Benefit Inflators

The consequences for some attributes are estimated based on the values published in the NRC Regulatory Analysis Handbook. Within the NRC Regulatory Analysis Handbook, the information in relation to severe reactor accident consequences is provided in previous year dollars. To evaluate the costs and benefits consistently, the consequences are inflated. The most common inflator is the Consumer-Price Index for all urban consumers (CPI-U), developed by the U.S. Department of Labor, Bureau of Labor Statistics. Using the CPI-U, the previous year dollars were converted to the year 2012. The formula to determine the amount in 2012 dollars is

.

Values of CPI-U used in this regulatory analysis are summarized in Table 77.

Table 77 Consumer Price Index – All Urban Consumers Inflator

Base Year	CPI-U Inflator for Year 2012
2005	1.1756
2006	1.1389
2007	1.1073
2008	1.0664

[53] Decommissioning of the unit must be completed within 60 years of permanent cessation of operations under 10 CFR 50.82, "Termination of License." Completion of decommissioning beyond 60 years will be approved by the Commission only when necessary to protect public health and safety.

Base Year	CPI-U Inflator for Year 2012
2009	1.0702
2010	1.0529
2011	1.0207

Source: www.bis.gov/data/inflation_calculator.htm

D.3.2.2.4 Dollar per Person-Rem Conversion Factor

Using the dollar value of the health detriment and a risk factor that establishes the nominal probability for stochastic health effects attributable to radiological exposure (fatal and non-fatal cancers and hereditary effects) provides a dollar per person-rem of $2,000, rounded to the nearest thousand, according to NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy," dated December 1995.

The NRC currently use a value of statistical life (VSL)[54] of $3 million based on NUREG-1530, and a cancer risk factor of 7.0 x 10^{-4}, which is a reduction to the closest significant digit of a recommendation by the International Commission on Radiation Protection (ICRP) in Publication No. 60. Therefore, the dollar per person-rem is equal to $3 million times 7.0 x 10^{-4} rounded to the nearest thousand (due to uncertainties) or $2,000.

D.3.2.2.5 Onsite Property Decontamination, Repair, and Refurbishment Costs

Spent fuel pool accident risks have significant contributions from onsite property monetary losses (e.g., repair and refurbishment) and plant decontamination. The risk dominant accident sequences involve the failure of the pool due to seismic or load drop events resulting in the loss of pool integrity. This scenario results in loss of spent fuel pool water inventory, zircaloy cladding fire initiation with propagation through the spent fuel assemblies stored in the pool, and an uncontrolled radiological release from the reactor building. The NRC assumes that, based on the current regulatory framework, with insights from the Fukushima Dai-ichi accident, that onsite property would be radiologically affected in the following way. The consequences of a spent fuel fire are expected to be similar to the Category II accident as defined in NUREG/CR-5281, section 3.2.4. Based on this reference, the cleanup and decontamination costs are estimated to be approximately $165 million (1983 dollars) and the cost for permanent disposal of the damaged fuel is $26 million (1983 dollars). Using Table C.95 from the RA Handbook, the pool repair to is expected to cost $72 million (1983 dollars). Adjusting these estimated costs using the CPI-U inflator formula and using a multiplier of three to model the high estimate and a divider of two to model the low estimate results in the values provided in Table 78.

54 The value of a statistical life (VSL) is the monetary value of a mortality risk reduction that would prevent one statistical (as opposed to an identified) death (Jones-Lee, 2004). The VSL is a key component in the calculation of the dollar per person-rem value, which is the product of the VSL multiplied by a risk coefficient.

Table 78 Onsite Property Decontamination, Repair, and Refurbishment Costs

Onsite Property Cost Element	1983 dollars			2013 dollars		
	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
Cleanup and decontamination	$165,000,000	$495,000,000	$82,500,000	$371,250,000	$1,113,750,000	$185,625,000
Repair Pool	$72,000,000	$216,000,000	$36,000,000	$162,000,000	$486,000,000	$81,000,000
Disposal of damaged fuel	$26,000,000	$78,000,000	$13,000,000	$58,500,000	$175,500,000	$29,250,000
Total	$263,000,000	$789,000,000	$131,500,000	$591,750,000	$1,775,250,000	$295,875,000

D.3.2.2.6 Replacement Energy Costs

Replacement energy costs are the costs for replacing the energy from the nuclear power plant due to a plant shutdown to install required equipment or due to an accident.[55] The NRC assumes that replacement energy costs would be required until onsite decontamination and repair efforts are completed or the unit is retired.

The NRC assumes that licensees engage in power purchase agreements (PPA)[56] to economically purchase replacement power. A PPA is a legal contract between an electricity generator (licensee) and a power purchaser. The NRC assumes that a licensee will not be able to replace the power through other generation for seven years and would have to buy power from the market. Although not all licensees may have PPAs, the licensee will still replace the lost energy any time that the nuclear power plant is not operating to meet its electrical power supply obligations. The NRC assumes that after 7 years, the onsite decontamination and repair efforts are completed or the unit is retired and other power sources will be developed to replace the unit’s lost electrical generation capability.

For the replacement energy cost calculation in this regulatory analysis, the NRC assumes that the reference plant is located on a multi-unit site. For the high estimate case, the NRC assumes that replacement energy is purchased for both the accident unit and the co-located unit at the site.

D.3.2.2.7 Occupational Worker Exposure (Accident)

There are two types of occupational exposure related to accidents: short-term and long-term. The first occurs at the time of the accident and during the immediate management of the emergency. The second is a long-term exposure, presumably at significantly lower individual rates, associated with the cleanup and refurbishment or decommissioning of the damaged facility. The value gained in the avoidance of both types of exposure is conditioned on the change in frequency of the accident's occurrence.

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55 The replacement energy cost is only the cost to buy the energy for production on the market. Therefore, the cost would be the cost of buying the cheapest energy. These estimates do not include transmission or distribution costs.

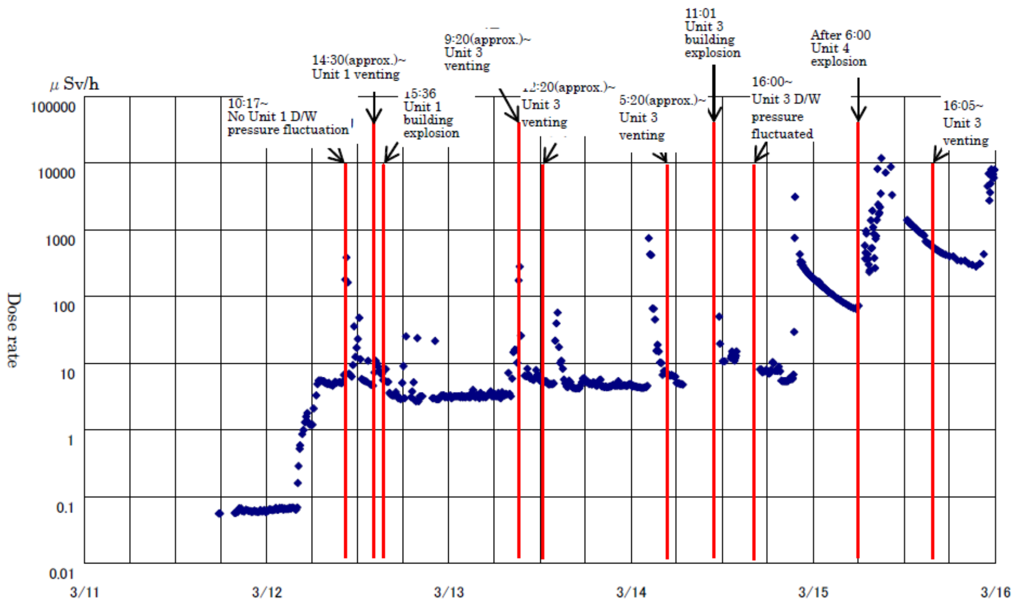
56 A power purchase agreement is a contract between two parties, one who generates electricity for the purpose of sale (the seller) and one who is looking to purchase electricity (the buyer). The PPA defines all of the commercial terms for the sale of electricity between the two parties, including when the project will begin commercial operation, schedule for delivery of electricity, penalties for under delivery, payment terms, and termination.

The experiences at the Three Mile Island Unit 2 (TMI-2), the Chernobyl, and the Fukushima nuclear power plants illustrated that significant occupational exposures could result from performing activities outside the control room during a power reactor accident. At TMI-2, the average occupational exposure related to the incident was approximately 1.0 rem, with a collective dose of 1,000 person-rem occurring over a 4-month span, after which time occupational exposure approached pre-accident levels. For Chernobyl, the average dose for persons closest to the plant was 3.3 person-rem (RA Handbook p. 5.30), yielding an average value of 3,300 person-rem.

After the Fukushima unit 1 building explosion on March 12, 2011, the unit 3 building explosion on March 14, and the unit 4 building explosion and the exposure of the unit 2 reactor fuel rods on March 15, radioactive materials were release into the environment and surrounding areas of the Fukushima Dai-ichi nuclear power plant. Measurement and evaluation of radiation exposure levels for workers engaged in emergency work at the Fukushima Daiichi NPS have been implemented continuously since the Tohoku Earthquake.

As shown in Figure 140, the dose rate in the vicinity of the main gate at the Fukushima Dai-ichi site near the time of the Unit 4 explosion varied between 20 mrem and 1.0 rem per hour (between 200 and 10,000 µSv per hour).

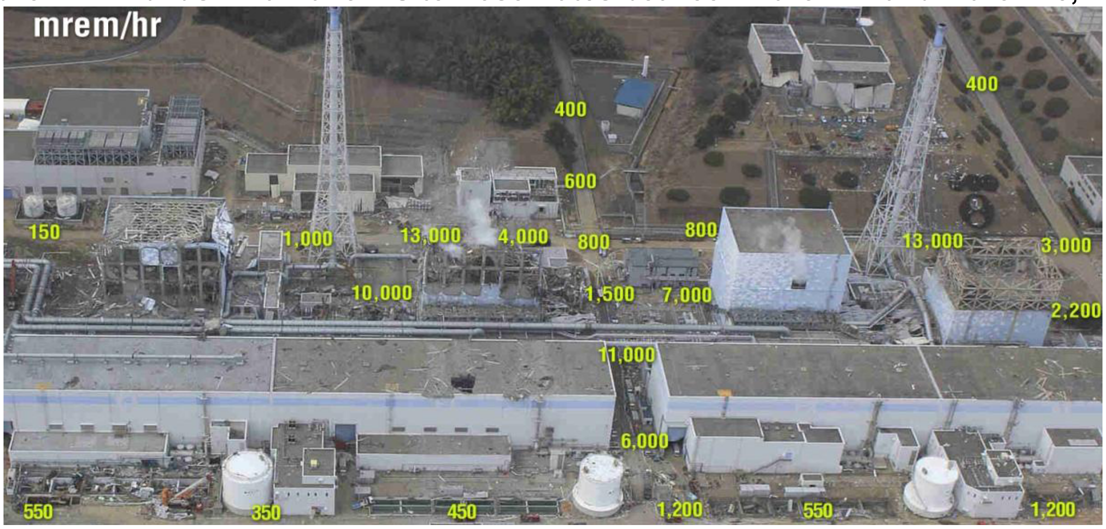
Figure 140: Dose Rate in Vicinity of Fukushima Daiichi Nuclear Plant Site Main Gate between March 11 and March 16, 2011



Source: Fukushima Nuclear Accident Analysis Report p. 371.

On March 22 and 23, surveys of the airborne radioactivity and dose rates around the Fukushima Daiichi site were collected and documented. The dose rates are shown on Figure 141.

Figure 141: Fukushima Daiichi Site Dose Rates between March 22 and March 23, 2011



Source: INPO 11-005, p 41

The distribution of total monthly exposure for workers engaged in radiation work at the Fukushima Daiichi NPS for the first three months following the March 2011 accident is provided in Table 79.

Table 79 Average Accident Occupational Exposure at Fukushima Dai-ichi Nuclear Power Plant from March to May 2011

Total Radiation Exposure (mSv)	Number of Plant Workers Exposed		
	March 2011[1]	April 2011[2]	May 2011[3]
≥ 250	6	0	0
200 - 249	2	0	0
150 - 199	14	0	0
100 - 149	77	0	0
50 - 99	309	3	0
20 - 49	859	81	19
10 - 19	1041	310	144
< 10	1434	3214	2854
Total number of workers	3742	3608	3017

1. Maximum March 2011 occupational exposure was 670.4 mSv.
2. Maximum April 2011 occupational exposure was 69.3 mSv.
3. Maximum May 2011 occupational exposure was 41.6 mSv.
4. One mSV is equal to 0.1 rem.

Source: Wada et al, Occupational and Environmental Medicine, 2012 August; 69(8): p. 600.

To estimate the monthly total occupational radiation exposure received by all workers, a high estimate, best estimate, and low estimate were calculated based on the maximum category value, the midpoint category value, and the first quartile category value. The results are tabulated in Table 80.

Table 80 Estimated Immediate Accident Occupational Monthly Exposure at Fukushima

Radiation Exposure (mSv)	Best Estimate Category Radiation Exposure (mSv)			High Estimate Category Radiation Exposure (mSv)			Low Estimate Category Radiation Exposure (mSv)		
	March 2011	April 2011	May 2011	March 2011	April 2011	May 2011	March 2011	April 2011	May 2011
≥ 250	460.2			670.4			355.1		
200 - 249	224.5			249			212.25		
150 - 199	174.5			199			162.25		
100 - 149	124.5			149			112.25		
50 - 99	74.5	69.3		99	69.3		62.25	62.25	
20 - 49	34.5	34.5	34.5	49	49	41.6	27.25	27.25	27.25
10 - 19	14.5	14.5	14.5	19	19	19	12.25	12.25	12.25
< 10	5	5	5	10	10	10	2.5	2.5	2.5
Total Monthly Dose	90,200	23,600	17,000	125,600	42,200	32,100	72,500	14,200	9,400
Avg Worker Dose	24.1	6.5	5.6	33.6	11.7	10.6	19.4	3.9	3.1

The immediate accident occupational exposure for a spent fuel pool accident shown in Table 81 is estimated based on the Fukushima data and the following assumptions:

- The immediate accident period lasts for one year,
- The workforce during the immediate accident period is 3,700 workers, and
- The average worker radiation exposure remains constant at the May 2011 value from May 2011 through February 2012.

Table 81 Immediate Accident Occupational Exposure for a Spent Fuel Pool Fire

Case	Immediate Accident Occupational Exposure (averted person-rem)
Low Estimate	18,070
Best Estimate	28,380
High Estimate	48,880

After the immediate response to a spent fuel pool fire, a long process of cleanup and refurbishment or decommissioning will follow. The Fukushima Nuclear Accident Analysis Report states, "The average value for 5,128 people in April of 2012 was 1.07 mSv per worker due to decreasing trends in environment dose rates (p 415). The NRC assumes that the process of cleanup and refurbishment or decommissioning will begin one year after the accident and will take seven years to complete. During those seven years, the NRC assumes that each occupational worker at the damaged reactor site will be exposed to 1.07 mSv per month (0.107 rem per month) for the duration of the cleanup and refurbishment or decommissioning. Assuming the average value for 5,128 workers would remain for the duration yields a cumulative long-term occupational dose of 46,000 person-rem.

In NUREG/CR-5281, Jo et al. (1989) conducted what essentially amounted to a regulatory analysis of a non-reactor nuclear fuel cycle facility using the 1983 Handbook (Heaberlin et al. 1983) as guidance. The accidental occupational exposure was assumed to be similar to that from TMI-2, which is 4,580 person-rem.

As described in the RA Handbook (p 5.30), the DOE (1987) summarized results on the collective dose received by the populace surrounding the Chernobyl accident. Average dose equivalents of 3.3 rem per person, 45 rem per person, and 5.3 rem per person were estimated for residents within 3 km, between 3 km and 15 km, and between 15 km and 30 km of Chernobyl, respectively (Mubayi et al. 1995, p. A-5). Assuming 1,000 workers and a 4.2 multiplier, an estimate radiation exposure of 14,000 person-rem results.

Site worker exposures following a spent fuel pool accident could be greater than that of a reactor core melt accident. This is because a spent fuel pool stores significantly more fuel assemblies than a reactor core. Additionally, radionuclides released during a spent fuel pool accident have longer half-lives (e.g., Cesium-137) than those that would be released during a reactor accident. Given the uncertainties in existing data and variability in severe accident parameters and worker response, Table 82 provides the long-term occupational dose used in this regulatory analysis to analyze spent fuel pool fires.

Table 82 Long-Term Accident Occupational Exposure for a Spent Fuel Pool Fire

Case	Immediate Accident Occupational Exposure (averted person-rem)
Low Estimate	4,580
Best Estimate	14,000
High Estimate	46,000

D.3.2.2.8 Long-Term Habitability Criteria

The long-term phase is the period following the seven-day emergency phase and is modeled for 50 years to calculate consequences from exposure of the average person. Radiation exposure during this phase is mainly from external radiation from trace contaminants that remain after the land is decontaminated, or in lightly contaminated areas where no decontamination was required. Internal radiation exposures may also occur during this period, including inhalation of resuspended radionuclides and ingestion of food and water with trace contaminants. Depending on the relevant protective action guides (PAGs) and the level of radiation, food, and water below a certain limit could be considered adequately safe for ingestion, and lightly contaminated areas could be considered habitable.

A long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be made by a number of local, state, and federal jurisdictions and would most likely 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 may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations.

Site-specific values are used to determine long-term habitability. For habitability, most states adhere to EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter. This habitability criterion was used in previous spent fuel pool studies, which used 4 rem in 5 years to represent these PAG levels (e.g., 2 rem in year one, followed by 0.5 rem each successive year). However, consistent with the location of the reference plant, the Spent Fuel Pool Study analysis utilizes the State of Pennsylvania habitability criterion of 500 mrem beginning in the first year (and each following year). The use of this long-term habitability criterion reduces the predicted long-term population doses and health effects and increases the costs associated with interdiction, decontamination, and condemnation.[57]

Given the uncertainties in which long-term habitability criterion would be used, Table 83 provides the long-term phase habitability criterion used in this analysis to analyze the consequences of spent fuel pool fires on public health (accident).

Table 83 Long-Term Habitability Criterion

Case	Long-Term Habitability Criterion	Protective Action Basis
Low Estimate	500 mrem annually	Pennsylvania dose limit to the public
Best Estimate	2 rem in the first year and 500 mrem each year thereafter	EPA intermediate phase PAGs
High Estimate	2 rem annually	EPA intermediate phase PAG: first year

Based on the average population dose for a release estimated using a sensitivity analysis, the public dose for the two EPA protective action bases was estimated by scaling population dose calculated using the Pennsylvania dose limit. The habitability criterion scaling factors used are provided in Table 84.

Table 84 Habitability Criterion Scaling Factors

	500 mrem	2 rem in the first year and 500 mrem each year thereafter	2 rem
Population Dose within 50 miles	100%	207%	278%
Total Population Dose	100%	165%	192%

The use of these habitability criteria also affects the values of offsite property damage used in this analysis. Certain metrics such as offsite property damage, the number of displaced individuals (either temporarily or permanently) and the extent to which such actions may be needed are inversely proportional to changes in collective dose resulting from changes in habitability criteria.

The impacts for alternate protective action levels were produced by examining the sensitivity analyses used to evaluate the effect of alternate protective action levels on land contamination, which were based on the results for a release from a high-density loading without credit for mitigation during OCP3. Scaling factors for different protective action levels were derived from this case. For a very large release that led to economic impacts beyond 50 miles, the sensitivity of the results within 50 miles to different protective action levels is less than the sensitivity of results beyond 50 miles. For significantly lower release magnitudes associated with the low density and successful mitigation cases, the scaling approach used can predict higher economic consequences within 50 miles than for the total. This implies that the economic impacts beyond 50 miles would be small relative to the economic impacts within 50 miles, and the total scaled

[57] Interdiction and condemnation refer to the relocation of people from contaminated areas according to the habitability criterion. Interdiction is the temporary relocation of the affected population while decontamination, natural weathering, and radioactive decay reduce the contamination levels. Condemnation is the permanent relocation of the affected population if decontamination, natural weathering, and radioactive decay cannot adequately reduce contamination levels to habitability limits within 30 years.

economic impact is therefore set equal to the scaled economic impact within 50 miles. The economic consequences scaling factors used are provided in Table 85.

Table 85 Economic Consequences Scaling Factors as a Function of Habitability Criteria

	500 mrem	2 rem in the first year and 500 mrem each year thereafter	2 rem
Economic Consequences within 50 miles	100%	67%	56%
Total Economic Consequences	100%	43%	31%

These criteria provide a benchmark for understanding the nature and the extent of the relationship between collective dose, economic consequences, and habitability criteria following a severe spent fuel pool accident. These measures are subject to large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies, the loss of infrastructure on the general U.S. economy, or the details of how long-term protective actions would be performed.

D.3.2.2.9 Other Key Data

All monetized costs are expressed in 2012 dollars. Ongoing costs of operation related to the alternatives are assumed to begin in 2014 unless otherwise stated, and are modeled on an annual cost basis.

Estimates were made for one-time implementation costs. The staff assumes that these costs will be incurred in the first year of the analysis unless otherwise noted.

Estimates were made for recurring annual operating expenses. The values for annual operating expenses are modeled as a constant expense for each year of the analysis horizon. An annuity calculation was performed to discount these annual expenses to 2012 dollar values.

Reference plant site population data was projected to year 2011 using the latest version of the computer code SECPOP2000. SECPOP2000 uses 2000 census data and applies a multiplier value of 1.1051 from the U.S. Census Bureau to account for the average population growth in the United States from 2000 to 2011 as discussed in section 7.1.3 of the main report. No further population growth was evaluated in this appendix.

D.3.2.3 Assumptions

The Spent Fuel Pool Study is used to inform this analysis is a consequence study based on the occurrence of a postulated beyond-design-basis earthquake (with an estimated frequency of occurrence of one event in 60,000 years) to a selected U.S. Mark I BWR spent fuel pool with a unit-specific spent fuel pool. The Spent Fuel Pool Study major assumptions are listed in section 2 of the main document. Additional assumptions used for this analysis are discussed below. The costs presented in this analysis are based on estimates by the authors or cited documents. It should be noted that this is a generic cost estimate and should be used accordingly. Site-specific features may result in higher or lower costs than those estimated.

D.3.2.3.1 Projected Number of Outages and Spent Fuel Assemblies

The reference plant is on a 24-month refueling cycle and is estimated to require eleven refueling outages between 2012 and the end of its operating license in 2034. It is assumed that

284 assemblies are offloaded to the spent fuel pool during each outage based on information in section 5.1 of the main document. The full core of 764 assemblies is offloaded to the spent fuel pool upon operating license expiration.

The analysis for the reference plant is based on a high-density spent fuel pool inventory of 3,055 assemblies in a high-density 1x4 loading configuration, a number based on the pool capacity of 3,819 assemblies, reduced by 764 assemblies to accommodate a full core offload capability using the existing high-density racking. In a low density 1x4 with empties configuration, the spent fuel pool stores 852 assemblies. The number of spent fuel assemblies required up to operating license expiration is calculated based on the existing high-density spent fuel pool inventory, the number added from refueling outages, and the full reactor core inventory and is provided in Table 86.

Table 86 Number of Spent Fuel Assemblies Remaining through Operating License Expiration

Category	Inventory	Number	No. of spent fuel assemblies
Current spent fuel pool inventory	3,055	1	3,055
refueling	284	11	3,124
reactor core	764	1	764
		Total	6,943

D.3.2.3.2 Dry Storage Capacity

Three companies supply most of the dry storage technologies to U.S. commercial nuclear power plants. These companies are Holtec International, Inc. (Holtec), NAC International, Inc. (NAC), and Transnuclear, Inc. (Transnuclear). The dry storage cask systems[58] (DSCs) for all three companies are certified by the NRC for storage of high burnup spent fuel (i.e., burnups greater than 45 GWd/MTU), using both regional and uniform loading of spent fuel in the packages. A summary of a representative sampling of dry storage canisters commercially available to the reference plant for BWR fuel storage is provided in Table 87.

Table 87 Representative Sampling of Commercially Available BWR Spent Fuel Dry Storage Technology

Vendor Package	Fuel Type	Canister Type	Capacity (Assemblies)	Maximum Decay Heat Per Package[1] (kW)
Holtec HI-STORM	BWR	MPC-68	68	34
Holtec HI-STORM FW	BWR	MPC-89	89	46.36
NAC MAGNASTOR	BWR	87B	87	33
Transnuclear NUHOMS	BWR	61BTH	61	31.2
Transnuclear TN-68	BWR	Bolted	68	30

The maximum decay heat per assembly for uniform loading is estimated by dividing the package decay heat by the number of assemblies. The maximum decay heat per assembly under regional loading schemes will generally be higher than the maximum decay heat per assembly assuming uniform loading for a smaller number of assemblies. Cask certificates of compliance provide the specific maximum assembly decay heat limits for each storage location in the basket.
Source: EPRI TR-1025206, p. 2-11.

[58] The term dry storage cask system (DSC) includes dual-purpose canister based systems, dual-purpose casks, and storage-only dry storage casks and canister systems.

D.3.2.3.3 Fuel Assembly Decay Heat as a Function of Burnup and Cooling Time

As fuel assembly burnups increase, the decay heat of the fuel assembly (watts per assembly) increases. Decay heat also can vary significantly with initial enrichment and assembly irradiation parameters. Spent fuel burnups have gradually increased since the 1990s with average BWR burnups about 43 GWd/MTU and range between 40 and 50 GWd/MTU. Spent fuel assembly average decay heat for a 40 GWd/MTU BWR assembly that has cooled for five years is approximately 360 watts/assembly. The average decay heat for a 50 GWd/MTU assembly that has cooled for five years is approximately 520 watts per assembly (EPRI TR-1021049, p. 2-3, Regulatory Guide 3.54). The average BWR spent fuel assembly that has cooled for five years is approximately 410 watts/assembly.

Table 88 Canister Storage Capacity Based on Heat Rate Limitations

Vendor Package	Capacity (Assemblies)	Maximum Decay Heat Per Package1 (kW)	Max. Capacity based on decay heat: 410w per assembly	520w per assembly	% Additional Canisters
Holtec HI-STORM	68	34	68.00	65.38	4.0%
Holtec HI-STORM FW	89	46.36	89.00	89.00	0.0%
NAC MAGNASTOR	87	33	80.49	63.46	37.1%
Transnuclear NUHOMS	61	31.2	61.00	60.00	1.7%
Transnuclear TN-68	68	30	68.00	57.69	17.9%

Based on the average BWR spent fuel assembly that emits 410 watts after it has cooled for five years, Table 88 shows that all of the dry storage canisters can be filled to capacity with the exception of the NAC MAGNASTOR, without exceeding the maximum decay heat per package rating, subject to restrictions on loading pattern. For 50 GWd/MTU assemblies that emit 520 watts after they have cooled for five years, fewer assemblies can be stored in a cask to ensure that it does not exceed the maximum decay heat rating. The number of additional dry storage casks required depends on the vendor package selected and range between no additional canisters to almost 40% additional canisters. Additional DSCs, which are required because of high heat load, are estimated in this appendix. For this regulatory analysis, the Transnuclear TN-68 dry casks are evaluated because the reference plant's ISFSI for dry cask storage utilizes the TN-68 cask design as discussed in section 1.3 of the main document. The currently approved minimum cooling time for fuel stored in the TN-68 dry casks is seven years (10 years for some fuel types), and Transnuclear would need to demonstrate, in an amendment request, that spent fuel that was cooled for a shorter period can be stored safely. The costs for Transnuclear to prepare such an amendment request and for the NRC review are not included in this regulatory analysis. The methodology used to estimate the capacity of the DSCs for spent fuel that has cooled for five years is subject to uncertainties resulting from decay heat and loading pattern restrictions. As a result, the actual DSC capacity may be higher or lower than those estimated.

D.3.2.3.4 Dry Storage Upfront Costs

Upfront costs include engineering, design, and licensing costs; equipment costs; construction costs; and start up and testing costs. Each of these cost components are further described in EPRI TR-1021048, "Industry Spent Fuel Storage Handbook." As noted in EPRI TR-1025206, "Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage after Five Years of Cooling, Rev. 1," the independent spent fuel storage installation

(ISFSI) upfront costs vary widely from site to site and the upfront costs for those in operation vary from several million to tens of millions of dollars. (EPRI TR-1025206, p. 2-23) Values for upfront costs were estimated based on two publically available cost estimates that identified the specified number of DSC to be stored. The estimate amortized upfront costs for each site is provided in Table 89.

Table 89 Amortized DSC Upfront Costs

ISFSI Facility	Upfront Cost Estimate (base year)	Upfront Cost Est. (2012 $)	DSC Storage Capacity	Attributed Upfront Cost per DSC (2012 $)
Monticello	$21.5 million (2005 $)	$25,275,400	30	$842,500
Pilgrim	$22 million (2006$)	$25,055,800	53	$472,800
Average (Best Estimate)		$25,165,600		$657,700

D.3.2.3.5 Incremental Costs Associated with Earlier DSC Purchase and Loading

Incremental costs are the costs associated with the purchase and loading of DSCs on a periodic basis. These costs include the capital costs for the DSC and the loading costs for the storage systems. The unit cost estimates used in this analysis are provided in Table 90. These cost estimates are based on the DSC unit costs that EPRI used for a Generic Interim Storage Facility (EPRI TR-1018722) and documented in EPRI TR-1025206. Operating nuclear power plants sites may experience incremental DSC purchase and loading costs that are higher or lower than the amount assumed in this analysis.

Table 90 Incremental Unit Cost Estimates

Item	Unit Cost (Constant $2012)
Canister	$780,000
Concrete overpack	$208,000
Loading of canister-based storage	$312,000
Total	$1,300,000

D.3.2.3.6 Incremental Annual ISFSI Operating Costs

Annual operating costs for an ISFSI during reactor operation include the costs associated with NRC inspections; security; radiation monitoring; ISFSI operational monitoring; technical specification and regulatory compliance, including implementation of new certificate of compliance (CoC) amendments; personnel cost and code maintenance associated with fuel selection for dry storage; personnel costs for spent fuel management and fabrication surveillance activities; electric power usage for lighting and security systems; road maintenance to the ISFSI site; and miscellaneous expenses associated with ISFSI maintenance. NRC license fees for dry storage are included as part of the 10 CFR 50 operating license fees.

Because the reference plant has already implemented dry storage, there are no incremental annual ISFSI operating costs expected to implement dry storage at an earlier date if a policy decision is made to accelerate the transfer of spent fuel stored in spent fuel pools to dry storage. Annual operating costs are a function of when a company begins dry storage.

Therefore, incremental costs associated with annual ISFSI operating costs are insignificant for this analysis.

D.3.2.3.7 Dry Storage Occupational Exposure (Routine)

Routine occupational exposure associated with dry storage of spent fuel includes worker dose associated with additional DSC loading, unloading and handling activities; additional ISFSI operations, maintenance, and surveillance activities; additional DSC storage at an ISFSI; and additional transportation cask loading, unloading, and handling activities.

Worker dose associated with DSC loading operations vary depending upon the cask technology being loaded, the characteristics of the fuel being loaded (e.g., fuel age and burnup), and fuel loading patterns in the DSC (e.g., the location of short-cooled, high burnup spent fuel or colder spent fuel within DSC baskets using regional loading). For the regulatory baseline, a worker dose of 400 person-mrem per DSC loaded was assumed. This radiation dose is consistent with that used in EPRI TR-1021049 and in EPRI TR-1018058, which analyzed worker impacts associated with loading spent fuel for transport to the proposed Yucca Mountain repository. Some sites achieve per package dose ranges in the range of 200 to 300 person-mrem per package loaded, while other sites experience higher per package dose rates. For the low-density storage case, each cask loaded in addition to the number required by the regulatory baseline is estimated to result in an incremental 400 person-mrem dose.

There is routine occupational dose associated with ISFSI annual operation and maintenance activities (i.e., inspection, surveillance, and security operations). The regulatory baseline assumes an annual dose of 120 person-mrem per site per year for inspection, surveillance, and security activities and 1,500 person-mrem per site per year for ISFSI operations and maintenance. These estimated radiation doses are consistent with assumptions used by EPRI in EPRI TR-1021049 and TR-1018058. Because additional shielding is assumed to be provided by concrete overpacks, the worker dose associated with ISFSI operations and maintenance is not expected to increase. Therefore, there is no incremental occupational dose predicted for performing annual ISFSI operation and maintenance.

There is routine occupational dose associated with the storage of each DSC at an operational ISFSI. The regulatory baseline assumes a worker dose of 170 person-mrem for each additional DSC loaded at an ISFSI site. This estimated radiation dose is consistent with assumptions used by EPRI in EPRI TR-1021049 and TR-1018058. Because additional shielding is assumed to be provided by concrete overpacks, the worker dose associated with each DSC stored at an operational ISFSI is not expected to increase. For the low-density spent fuel pool storage case, each cask stored in addition to the number required by the regulatory baseline is estimated to result in an incremental 170 person-mrem dose.

Table 91 summarizes the occupational dose estimates for each activity.

Table 91 Incremental Occupational Dose (Routine) Estimates

Activity	Incremental Occupational Dose (Routine) (person-mrem per activity)
Load a DSC	400
ISFSI Operation and maintenance	0
Loading a DSC at an ISFSI	170
Total	570

D.3.2.3.8 Number of Dry Storage Casks

In 2012, the reference plant has 3,819 fuel assemblies stored in the spent fuel pool in a high-density 1x4 loading configuration. During each refueling outage, 284 assemblies are offloaded from the reactor vessel to the spent fuel pool. For the regulatory baseline, the plant is expected to load the required number of Transnuclear TN-68 DSCs with a 68-assembly capacity each refueling outage to retain sufficient space in the spent fuel pool to discharge one full core of fuel (full core reserve). The estimated inventory for use by this regulatory analysis is shown in Table 92.

Table 92 Regulatory Baseline Loading of Dry Storage Casks

Year	Initial SFP inventory	Refueling	Placed into dry storage	Final SFP Inventory	No. of casks loaded	Cask Capacity
2012	3055	284	-340	2999	5	68
2014	2999	284	-272	3011	4	68
2016	3011	284	-272	3023	4	68
2018	3023	284	-272	3035	4	68
2020	3035	284	-272	3047	4	68
2022	3047	284	-340	2991	5	68
2024	2991	284	-272	3003	4	68
2026	3003	284	-272	3015	4	68
2028	3015	284	-272	3027	4	68
2030	3027	284	-272	3039	4	68
2032	3039	284	-272	3051	4	68
2034	3051	764	0	3815	0	68
2040	3815	0	-816	2999	12	68
2041	2999	0	-816	2183	12	68
2042	2183	0	-816	1367	12	68
2043	1367	0	-680	687	10	68
2044	687	0	-687	0	11	68
Total number of casks					103	

At the expiration of the operating license in 2034, the full core is offloaded into the spent fuel pool. The analysis further assumes that the entire spent fuel pool inventory will gradually be placed into dry storage beginning in 2040 and completed by 2044, 10 years after termination of unit commercial operation.

For the low-density spent fuel pool storage case, it is assumed that there is an NRC policy decision that requires licensees to offload the spent fuel inventory to dry storage to obtain a low-density 1x4 with empties configuration within five years (e.g., by end of 2019). In this configuration, the reference plant spent fuel pool stores 852 assemblies (Spent Fuel Pool Study, Table 15). Using the same initial conditions as above, and using the DSC with a 57-assembly derated capacity beginning in year 2019, the inventory model is provided in Table 93.

Table 93 Low-density Spent Fuel Pool Case Loading of Dry Storage Casks

Year	Initial SFP inventory	Refueling	Placed into dry storage	Final SFP Inventory	No. of casks loaded	Cask Capacity
2012	3055	284	-340	2999	5	68
2013	2999	0	0	2999		
2014	2999	284	-544	2739	8	68
2015	2739	0	-544	2195	8	68
2016	2195	284	-544	1935	8	68
2017	1935	0	-544	1391	8	68
2018	1391	284	-544	1131	8	68
2019	1131	0	-285	846	5	57
2020	846	284	-285	845	5	57
2022	845	284	-285	844	5	57
2024	844	284	-285	843	5	57
2026	843	284	-285	842	5	57
2028	842	284	-285	841	5	57
2030	841	285	-285	841	5	57
2032	841	286	-285	842	5	57
2034	842	764	-798	808	14	57
2043	808	0	-408	400	6	68
2044	400	0	-400	0	6	68
Total number of casks					111	

At the expiration of the operating license in 2034, the full core is offloaded into the spent fuel pool. The analysis further assumes that the entire spent fuel pool inventory will gradually be placed into dry storage beginning in 2043 and completed by 2044, taking only two years because of the smaller remaining inventory. Additionally, in years 2038 and 2039, the spent fuel has cooled for a sufficient length of time that the DSC is no longer derated.

D.3.3 Sensitivity Analysis

D.3.3.1 Present Value Calculations

Current trends in the marketplace have provided returns on investments well below the 3 percent and 7 percent discount rates, which OMB Circular No. A-4 is based. The NRC is providing a zero discount rate (e.g., undiscounted values) as a sensitivity analyses. Historically, regulatory analyses have provided the undiscounted values for the costs and benefits for information purposes, but have not provided them as a sensitivity analysis. However, the NRC is reporting the undiscounted costs and benefits as part of the sensitivity analysis based on current market trends and future predictions.

D.3.3.2 Dollar per Person-Rem Conversion Factor

The NRC is currently revising the dollar per person-rem averted conversion factor based on recent information regarding the value of a statistical life (VSL). However, until the NRC completes the update and publishes the appropriate guidance documents, the NRC will perform sensitivity analysis to estimate the impact on the calculated results when more current VSL and cancer risk factor are used. The NRC used the U.S. Environmental Protection Agency's (EPA) VSL as an interim value in the sensitivity analysis. The EPA's VSL was developed through a rigorous process, reviewing many published academic papers, and includes review from the Scientific Advisory Board, an independent review board.

The EPA's VSL in 2009 dollars is approximately \$7.2 million.[59] The VSL is derived from "using a mixed effects model (random intercept with fixed effects for study characteristics), the authors regressed the VSL estimates on average income, probability of death, and several study design variables" (EPA, page 41). Therefore, using the CPI-U based inflator to adjust from 2009 dollars to 2012 dollars yields a VSL of approximately \$7.7 million. The International Commission on Radiation Protection (ICRP) updated the mortality risk factor in ICRP Publication No. 103, the updated risk coefficient is 5×10^{-4}. Using the updated ICRP risk coefficient and escalated EPA-based VSL, the dollar per person-rem conversion, rounded to the nearest thousand, is \$4,000 per person-rem.

Therefore, the NRC will provide the \$2,000 per person-rem conversion value for the recommendation and the \$4,000 per person-rem conversion value as a sensitivity analysis for this regulatory analysis.

D.3.3.3 Replacement Energy Costs

The NRC is currently updating its estimates for replacement energy costs based on a U.S. competitive electricity market area model. The updated model provides the replacement energy costs by day, week, and year, based on market area, in 2010 dollars. For each U.S. power market area, a lowest cost and highest cost replacement energy cost estimate was calculated, normalizing for reactor megawatt rating differences. The estimated replacement energy cost per reactor per year ranges from a high estimate of \$54.4 million to a low estimate of \$692,000 across all U.S. power markets. The average estimated cost per reactor per year across all U.S. power markets is \$9.6 million and the median estimated cost is \$6.4 million in 2010 dollars. Using the CPI-U inflator formula and the 2010 CPI-U inflator value from Table 77, the estimated replacement energy costs range from \$57.3 million to \$729,000 in 2012 dollars. The average estimated cost per reactor per year across all US power markets is \$10.1 million and the median estimated cost is \$6.7 million in 2012 dollars.

D.3.3.4 Consequences Extending Beyond 50 Miles

NUREG/BR-0184 states that in the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile distance from the plant site. However, in this circumstance it is beneficial for the analysis to include supplemental information (e.g., analyses and results) that go beyond the guidance provided in this document. The Spent Fuel Pool Study uses a plume release model that predicts slow deposition of aerosols. This results in public health consequences that extend beyond 50 miles from the postulated accident site. While the accuracy of the model decreases with distance, the amount of public exposure beyond 50 miles in the event of a release is expected to be significant. To capture effects beyond 50 miles, this regulatory analysis

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59 Environmental Protection Agency, National Center for Environmental Economics, "Valuing Mortality Risk Reductions for Environmental Policy: A White Paper", dated December 2010.

evaluates the public health and safety and economic consequences estimated by the plume model beyond the 50-mile distance from the plant site as a sensitivity analysis.

D.3.4 Alternative – Low-Density Spent Fuel Pool Storage

D.3.4.1 Public Health (Accident)

This attribute measures expected changes in radiation exposure to the public due to change in accident frequencies or accident consequences associated with the proposed action. The expected changes in radiation exposure are predicted over a 50-mile radius from the plant site. The calculated radiation dose to the public is primarily from reoccupation of the land and other activities following the spent fuel pool accident. In addition, the calculated radiation dose to the public includes the occupational dose to workers for cleanup and decontamination of contaminated land not onsite. The incremental radiation doses are calculated by subtracting the values for the alternative from those of the regulatory baseline. The difference (delta) is the averted dose benefit of this alternative in units of person-rem. The quantitative results for public health (accident) considering the contribution of all initiators that could affect spent fuel pool risk is provided in Table 94. These values are based on the MACCS2 analyses and probabilistic considerations described in further detail in the Spent Fuel Pool Study and other referenced documents. The assumptions with regard to the release frequencies are discussed in section D.3.2.2.1 and with regard to the habitability criteria are found in section D.3.2.2.8 of this regulatory analysis.

Table 94 Summary of Public Health (Accident) for Low-density Spent Fuel Pool Storage [All Initiators]

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
	Low Est.	Best Est.	High Est.	Undiscounted	3% Net Present Value			7% Net Present Value		
				Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Low-density storage	60	124	1,260	$247,700	$86,700	$179,500	$1,825,500	$60,200	$124,600	$1,267,000

As Table 94 shows, the best estimate of the delta benefit for averted public health (accident) radiation exposure from a spent fuel pool accident, which results in spent fuel damage, is 124 person-rem. This dose represents the reduction of public health risk that results from a policy decision to transfer spent fuel from the spent fuel pool to dry storage in order to achieve low-density spent fuel loading in the pool at the reference plant. This value is based on a spent fuel pool accident that results in an averted delta dose exposure of approximately 5.6 person-rem per reactor-year over a remaining licensed lifetime of 22 years (until year 2034). The best estimate values are based on the reference site's population density of 722 people per square mile within a 50-mile radius from the site and result from the uncontrolled release of radionuclides from a full spent fuel pool. The low estimate case reflects the health benefit of a spent fuel pool with low-density storage compared to a pool with high-density storage if the more stringent Pennsylvania protective action guides are used following an event challenging spent fuel pool cooling. The high estimate case reflects the calculated health benefits that result if a less stringent 2 rem annual dose protective action guide is used.

A case to evaluate the sensitivity of the results to a change in the dollar per person-rem conversion value from $2,000 to $4,000 per person-rem averted was performed. The results of this case are provided in Table 95.

Table 95 Sensitivity Analyses of Public Health (Accident) Benefits for Low-density Spent Fuel Pool Storage for All Initiating Events (within 50 miles)

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
				Undiscounted	3% Net Present Value			7% Net Present Value		
	Low Est.	Best Est.	High Est.	Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Dollar per person-rem value	60	124	1,260	$495,500	$173,400	$358,900	$3,650,900	$120,400	$249,100	$2,534,000

Because a spent fuel pool fire under certain scenarios and environmental conditions could result in impacts to public health that extend beyond 50 miles, the next two cases evaluate the sensitivity of averted public health exposures extending beyond 50 miles from the site. The first sensitivity case extends the analysis beyond 50 miles from the plant site and uses the same low, best, and high estimate case assumptions for habitability described above and uses the standard $2000 per person-rem conversion factor. The second sensitivity case evaluates the sensitivity of extending the analysis beyond 50 miles and uses a $4,000 per person-rem conversion factor. Table 96 shows the sensitivity on public health (accident) benefits for these two cases.

Table 96 Sensitivity Analyses of Public Health (Accident) Benefits for Low-density Spent Fuel Pool Storage for All Initiating Events (extending beyond 50 miles)

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
				Undiscounted	3% Net Present Value			7% Net Present Value		
	Low Est.	Best Est.	High Est.		Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Base case extended beyond 50 miles	541	892	7,868	$1,783,450	$783,250	$1,291,900	$11,399,100	$543,650	$896,700	$7,911,700
Dollar per person-rem value	541	892	7,868	$3,566,900	$1,566,500	$2,583,800	$22,798,200	$1,087,300	$1,793,400	$15,823,400

D.3.4.2 Occupational Health (Accident)

Occupational health measures both short-term and long-term health effects associated with site workers as a result of changes in accident frequency or accident mitigation. Within the regulatory baseline, the short-term occupational exposure related to the accident occurs at the time of the accident and during the immediate management of the emergency and during decontamination and decommissioning of the onsite property. The radiological occupational exposure resulting from cleanup and refurbishment or decommissioning activities of the damaged facility to occupational workers are estimated within the long-term occupational exposure. The quantitative results for occupational health (accident) considering the contribution of all initiators that could affect spent fuel pool risk is provided in Table 97 and is based on the release frequencies discussed in section D.3.2.2.1 and the occupational health (accident) assumptions found in section D.3.2.2.7.

Table 97 Occupational Health (Accident) Benefits for Low-density Spent Fuel Pool Storage Considering All Initiating Events

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
				Undiscounted	3% Net Present Value			7% Net Present Value		
	Low Est.	Best Est.	High Est.	Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
short-term	0.3	0.4	5.5	$841	$388	$628	$7,959	$269	$453	$5,524
long-term	0.1	0.2	5.2	$415	$98	$310	$7,490	$68	$223	$5,198
Total	0.3	0.6	10.7	1,260	490	940	15,450	340	680	10,720

As Table 97 shows, the total delta benefit for short- and long-term occupational health (accident) is 0.6 person-rem averted per reactor. The estimated total benefit of the occupational health (accident) attribute for low-density spent fuel pool storage relative to the regulatory baseline, using the $2,000 per person-rem averted conversion factor, net present value ranges are insignificant for this analysis and do not warrant further sensitivity analysis.

D.3.4.3 Occupational Health (Routine)

Occupational health (routine) accounts for radiological exposures to workers during normal facility operations (i.e., non-accident situations). These occupational exposures occur during DSC loading and handling activities, ISFSI operations, and maintenance and surveillance activities. The assumptions in relation to the exposures for occupational health (routine) are found in section D.3.2.3.7 of this regulatory analysis.

Table 98 Occupational Health (Routine) Costs for Low-density Spent Fuel Pool Storage

	No. of DSCs			Dose (person-rem)		Costs (2012 dollars)		
Year	Low-Density SFP Loading	Regulatory Baseline	Difference	Exposure per DSC	Additional Dose	No Discount	3% NPV	7% NPV
2012	5	5	0	0.57	0	$0	$0	$0
2013	0	0	0	0.57	0	$0	$0	$0
2014	8	4	-4	0.57	-2.28	-$4,560	-$4,298	-$3,983
2015	8	0	-8	0.57	-4.56	-$9,120	-$8,346	-$7,445
2016	8	4	-4	0.57	-2.28	-$4,560	-$4,052	-$3,479
2017	8	0	-8	0.57	-4.56	-$9,120	-$7,867	-$6,502
2018	8	4	-4	0.57	-2.28	-$4,560	-$3,819	-$3,039
2019	5	0	-5	0.57	-2.85	-$5,700	-$4,635	-$3,550
2020	5	4	-1	0.57	-0.57	-$1,140	-$900	-$663
2022	5	5	0	0.57	0	$0	$0	$0
2024	5	4	-1	0.57	-0.57	-$1,140	-$800	-$506
2026	5	4	-1	0.57	-0.57	-$1,140	-$754	-$442
2028	5	4	-1	0.57	-0.57	-$1,140	-$710	-$386
2030	5	4	-1	0.57	-0.57	-$1,140	-$670	-$337
2032	5	4	-1	0.57	-0.57	-$1,140	-$631	-$295
2034	14	0	-14	0.57	-7.98	-$15,960	-$8,329	-$3,602
2040		12	12	0.57	6.84	$13,680	$5,979	$2,058
2041		12	12	0.57	6.84	$13,680	$5,805	$1,923
2042		12	12	0.57	6.84	$13,680	$5,636	$1,797
2043	6	10	4	0.57	2.28	$4,560	$1,824	$560
2044	6	11	5	0.57	2.85	$5,700	$2,214	$654
				Total:	**-4.56**	**-$9,000**	**-$24,000**	**-$27,000**

As Table 98 shows, the delta benefit for occupational health (routine) is an increase of 4.56 person-rem in worker exposure resulting from DSC loading and handling activities; ISFSI operations; and maintenance and surveillance activities. The estimated cost to the occupational health (routine) for low-density spent fuel storage relative to the regulatory baseline and calculated in accordance with the current regulatory framework, ranges from $24,000 (3 percent net present value) to $27,000 (7 percent net present value) using the $2,000 per person-rem averted conversion factor. These ranges are insignificant for this analysis and do not warrant further sensitivity analysis.

D.3.4.4 Offsite Property

The offsite property attribute measures the expected total monetary effects on offsite property resulting from the proposed action. Changes to offsite property can take various forms, both direct, (e.g. land, food, and water) and indirect (e.g. tourism). This attribute is the product of the change in accident frequency and the property consequences from the occurrence of a spent fuel pool accident at the reference plant.

For the regulatory baseline, the offsite property costs are any property consequences resulting from any radiological release from the occurrence of an accident. Normal operational releases and any plant releases not related to the severe accident analyzed are outside the scope of this regulatory analysis.

The cost offsets for the analyzed spent fuel pool accident are quantified relative to the regulatory baseline based on the MACCS2 calculation results and probabilistic considerations provided in the main document. The results for the consequences from a low-density spent pool accident are compared to those from the regulatory baseline spent fuel pool accident. The calculation is the difference between the calculated consequences resulting from a low-density and a high-density spent fuel pool accident and are provided in Table 99.

Table 99 Offsite Property Cost Offsets for Low-density Spent Fuel Pool Storage

Case	Offsite Property Cost Offsets (2012 dollars)						
	Undiscounted	3% Net Present Value			7% Net Present Value		
	Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Base case, consequences within 50 miles	$723,300	$777,500	$524,000	$3,323,400	$539,700	$363,700	$2,306,700
Sensitivity study, consequences extend beyond 50 miles	$2,139,300	$3,599,100	$1,549,700	$8,393,400	$2,498,000	$1,075,600	$5,825,500

As Table 99 shows the estimate of offsite property damage can vary significantly with the criterion used to measure or estimate the level of contamination. This regulatory analysis uses three protective action levels – the Pennsylvania PAG of 500 mrem annually for the low estimate, the EPA intermediate phase PAG level of 2 rem in the first year, and 500 mrem annually thereafter for the best estimate, and 2 rem annually for the high estimate – to evaluate post-accident collective dose and offsite property costs. As discussed in section D.3.2.2.8, offsite property costs are inversely proportional to changes in collective dose resulting from changes in habitability criteria (i.e., lower PAG guidelines result in lower collective dose value and higher offsite property costs). Furthermore, the high estimate is also affected by the bounding assumption used in establishing the high estimate spent fuel pool release frequency shown in Table 75. As shown in Table 99 the estimated total cost offsets for the low-density storage option relative to the regulatory baseline range from $0.5 to $3.3 million (3 percent net present value) and from $0.4 to $2.3 million (7 percent net present value) considering consequences within 50 miles from the site. As a sensitivity study, the analysis of potential consequences was extended beyond 50 miles from the site and were quantified based on the MACCS2 model. These estimate results are also shown in Table 99 and result in cost offsets approximately 2.5 to 4.6 times greater than those in the base case result.

This analysis does not address potential changes to current methodologies and tools to regulatory analysis guidance that may result from applying SOARCA insights and improving guidance and analysis tools (such as the MACCS2 computer code) based on up-to-date data in addition to advancements in accident consequence assessment knowledge as it relates to this attribute.

D.3.4.5 Onsite Property

This attribute measures the expected monetary effects on onsite property, including replacement power costs, decontamination, and refurbishment costs, from the proposed action. There are two forms of onsite property costs that each alternative must disposition. The first type of onsite property costs are the cleanup and decontamination costs for the unit. The second type of onsite property costs is the cost to replace the energy from the damaged or shutdown unit(s). The cost offsets for low-density spent fuel pool storage are quantified relative to the regulatory baseline based on the probabilistic considerations provided in the main document and the onsite property estimates described in section D.3.2.2.5.

As stated in section D.3.2.2.6, another unit is co-located on the reference plant's site. Therefore, both units may not operate (e.g., due to significant site damage or contamination resulting in high occupational exposure to the undamaged unit) due to the spent fuel pool accident. In modeling the replacement energy costs based on this scenario, it is assumed for the high estimate that replacement energy would be purchased for both units.

Based on these modeling assumptions, the onsite property results are provided in Table 100.

Table 100 Summary of Onsite Property Cost Offsets for Low-density Spent Fuel Pool Storage

Case	Onsite Property Cost Offsets (2012 dollars)						
	Undiscounted	3% Net Present Value			7% Net Present Value		
	Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Onsite Property - Replacement Energy	$1,639	$50	$1,091	$117,100	$30	$682	$73,200
Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$8,800	$2,900	$5,800	$132,500	$1,800	$3,600	$82,600
Total	$10,440	$2,950	$6,890	$249,600	$1,830	$4,280	$155,800

As Table 100 shows, based on these calculations, the delta cost offset for probability weighted onsite property best estimate ranges from $6,890 (3 percent net present value) to $4,280 (7 percent net present value). Low and high estimates are also provided in Table 100.

D.3.4.6 Industry Implementation

Industry implementation accounts for the projected net economic effect on the affected licensees to implement the mandated changes. Costs evaluated for dry storage include upfront and incremental DSC capital and loading costs. Additional costs above the regulatory baseline are considered negative and cost savings are considered positive. The quantitative results for industry implementation are given in terms of expected costs if a policy decision is made to accelerate the transfer of spent fuel stored in spent fuel pools to dry storage. These expected costs are not frequency weighted. Assumptions used for developing the industry implementation cost model are discussed in sections D.3.2.3.2, D.3.2.3.5, and D.3.2.3.6, with the results provided in Table 101.

Table 101 Industry Implementation Cost Model for Low-density Spent Fuel Pool Storage

	No. of DSCs			Unit Costs			Costs (2012 dollars)		
Year	Low-Density	Regulatory Baseline	Difference	One Time ISFSI Mod	Upfront costs per	DSC Purchase and Loading	No Discount	3% NPV	7% NPV
2012	5	5	0		$657,632	$1,300,000	$0	$0	$0
2013	0	0	0		$657,632	$1,300,000	$0	$0	$0
2014	8	4	-4		$657,632	$1,300,000	-$7,830,528	-$7,381,024	-$6,839,486
2015	8	0	-8		$657,632	$1,300,000	-$15,661,056	-$14,332,085	-$12,784,087
2016	8	4	-4		$657,632	$1,300,000	-$7,830,528	-$6,957,323	-$5,973,872
2017	8	0	-8		$657,632	$1,300,000	-$15,661,056	-$13,509,364	-$11,166,116
2018	8	4	-4		$657,632	$1,300,000	-$7,830,528	-$6,557,944	-$5,217,811
2019	5	0	-5		$657,632	$1,300,000	-$9,788,160	-$7,958,670	-$6,095,574
2020	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,545,373	-$1,139,360
2022	5	5	0		$657,632	$1,300,000	$0	$0	$0
2024	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,373,044	-$869,212
2026	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,294,225	-$759,203
2028	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,219,932	-$663,118
2030	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,149,902	-$579,193
2032	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,083,893	-$505,889
2034	14	0	-14		$657,632	$1,300,000	-$27,406,848	-$14,303,428	-$6,186,086
2040	0	12	12		$657,632	$1,300,000	$23,491,584	$10,267,625	$3,533,186
2041	0	12	12		$657,632	$1,300,000	$23,491,584	$9,968,568	$3,302,043
2042	0	12	12		$657,632	$1,300,000	$23,491,584	$9,678,222	$3,086,022
2043	6	10	4		$657,632	$1,300,000	$7,830,528	$3,132,111	$961,377
2044	6	11	5		$657,632	$1,300,000	$9,788,160	$3,801,105	$1,123,105
		Total:	**-8**			**Total:**	**-$15,660,000**	**-$41,820,000**	**-$46,770,000**

For this analysis, the Transnuclear TN-68 dry casks are evaluated for the best estimate because the reference plant's ISFSI for dry cask storage utilizes the TN-68 cask design as discussed in Section 1.3 of the main report. The results provided in Table 102 show that eight additional DSCs are needed to store the hotter spent fuel.

Table 102 Industry Implementation Costs for Low-density Spent Fuel Pool Storage

Case	Costs (2012 dollars)		
	No Discount	3% Net Present Value	7% Net Present Value
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000

Table 102 shows, the incremental costs associated with DSC upfront costs and the earlier purchasing and loading of DSCs on a periodic basis. The estimated industry implementation costs for low-density spent fuel storage relative to the regulatory baseline and calculated in accordance with the current regulatory framework, ranges from $41.8 million (3 percent net present value) to $46.8 million (7 percent net present value).

D.3.4.7 Industry Operation

Industry operation accounts for the projected net economic effect due to routine and recurring activities required by the proposed alternative. Annual operating costs for an ISFSI during reactor operation include the costs associated with NRC inspections; security; radiation monitoring; ISFSI operational monitoring; technical specification and regulatory compliance, including implementation of new certificate of compliance (CoC) amendments; personnel cost and code maintenance associated with fuel selection for dry storage; personnel costs for spent fuel management and fabrication surveillance activities; electric power usage for lighting and security systems; road maintenance to the ISFSI site; and miscellaneous expenses associated with ISFSI maintenance. NRC license fees for dry storage are included as part of the 10 CFR 50 operating license fees. As discussed in section D.3.2.3.6, incremental costs associated with annual ISFSI operating costs are insignificant for this analysis.

Industry operation also includes annual operating costs following reactor shutdown for decommissioning, which includes the costs associated with transporting spent fuel offsite. These costs were beyond the scope of the evaluation of expedited transfer of spent fuel to dry cask storage and are not included in this analysis.

D.3.4.8 NRC Implementation

These costs, if calculated, would further reduce the calculated net benefit for this reference plant regulatory and backfit analysis.

D.3.4.9 NRC Operation

These costs, if calculated, would further reduce the calculated net benefit for this reference plant regulatory and backfit analysis.

D.3.4.10 Other Considerations

D.3.4.10.1 Modeling Uncertainties

There remain significant uncertainties in estimating the frequency of events for natural phenomena, which are postulated to challenge spent fuel pool cooling or integrity. There are also significant uncertainties in the calculation of event consequences in terms of the dispersion and disposition of radioactive material into the site environs. This is due in part to significant uncertainties regarding the degree to which topographical features and other phenomena are modeled at distances away from the reference plant. Estimating economic consequences also includes large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies and the loss of infrastructure on the general U.S. economy. An example of this is the supply chain disruptions that followed the 2011 Tohoku earthquake and subsequent tsunami on Japan or the 2004 Indian Ocean earthquake and tsunami on Thailand.

D.3.4.10.2 Cask Handling Risk

The NRC recognizes that there are costs and risks associated with the handling and movement of spent fuel casks in the reactor building. These cost and risk impacts, if included in this analysis, would further reduce the overall net benefit in relation to the regulatory baseline. These effects (e.g., the added risks of handling and moving casks) were conservatively ignored in order to calculate the maximum potential benefit by only comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and its implementation costs without consideration of cask movement risk.

D.3.4.10.3 Mitigating Strategies

The release of fission products to the environment from events that may cause the loss of spent fuel pool cooling or integrity, such as seismic events, missiles, heavy load drops, loss of cooling or make-up, inadvertent drainage or siphoning and pneumatic seal failures, are estimated to be approximately 5.5×10^{-7} per reactor-year without successful mitigation. Operator diagnosis and recovery are important factors considered in the development of the event frequencies for these events and portions of this evaluation are premised on licensees having taken appropriate actions to understand the potential consequences of spent fuel pool accident events and develop appropriate procedures and mitigating strategies to respond and mitigate the consequences.

The main report evaluated the potential benefits of mitigation measures required under Title 10, Code of Federal Regulations (10 CFR), Part 50.54 (hh)(2), which were implemented following the September 11, 2001 attacks. These mitigation measures are intended to maintain spent fuel pool cooling in the event of a loss of large areas of the plant due to explosions or fire. The main report does not consider the post-Fukushima improvements required by NRC and being implemented by the plants. These improvements are intended to increase the likelihood of restoring or maintaining power and mitigation capability during severe accidents.

The new spent fuel pool level instrumentation required under Order EA-12-051 and the mitigation strategies now required under Order EA-12-049, significantly enhance the likelihood of successful mitigation beyond that considered in section 5.3 of the main report because of the following features:

- Portable equipment with redundant sets (e.g., N+1) that is sufficient to supply all functions, simultaneously for the entire site, including equipment for the spent fuel pool. This portable equipment provides reasonable protection from seismic events, which are a dominant contributor to spent fuel pool risk.

- The mission time for this equipment is indefinite, versus the 12-hour mission time for the 50.54(hh)(2) equipment.[60]

- The new EA-12-049 mitigating strategies are capable of being deployed in all modes, which means that the new strategies can address spent fuel pool cooling issues that could occur in any operating cycle phase.

- The new spent fuel pool level instrumentation required under Order EA-12-051, ensures a reliable indication of the water level in the spent fuel pool for identification of the following pool water level conditions:
    - A level that is adequate to support operation of the normal fuel pool cooling system
    - A level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and
    - A level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

[60] This section of the regulations deals with the development and implementation of guidance and strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with loss of large areas of the plant resulting from explosions or fire.

- The minimum spent fuel pool makeup flow rate under Order EA-12-049 is set to match the design basis heat load for the spent fuel pool, which is typically a full core offload in addition to the recently removed fuel from the last refueling outage. This results in a lower flow rate than that in NEI guidance for Part 50.54 (hh)(2) equipment and an earlier transition to spray, if necessary, due to leaks.

- The method of filling the spent fuel pool is via a connection to the normal spent fuel pool makeup system located away from the spent fuel pool floor, reducing the impacts on human performance due to potentially adverse environmental conditions (e.g., high temperature, humidity, and radiation) following an event.

This additional equipment, strategies, and features provided by Orders EA-12-049 and EA-12-051, provide additional accident mitigation capability and would further enhance the likelihood of successful mitigation, thereby further reducing the value for the conditional probability of release.

D.3.4.10.4 Other Favorable Spent Fuel Loading Configurations

In section 9.2 of the Spent Fuel Pool Study, a sensitivity analysis is provided in which a more favorable fuel pattern is considered. In this more favorable pattern, eight cold assemblies surround each hot assembly (i.e., 1x8 fuel assembly pattern). Although only a few sensitivity analysis were performed using this configuration, the results are promising. The sensitivity calculations for the high-density 1x8 fuel pattern showed a shorter time to air coolability (i.e. no releases in OCP3). Even for the cases that led to the release of radioactive materials in OCP2, the release magnitude was much smaller than for the 1x4 fuel pattern, and comparable to the low-density cases. Furthermore, the high-density loading configuration, which allows for 764 empty cells for a full core offload may result in similar reductions in risk to the low-density storage option evaluated without the significant capital costs for implementation. Further evaluation of this alternative and possibly other loading configurations for all operating cycle phases is recommended.

D.4 PRESENTATION OF RESULTS

This section presents the analytical results, including discussion of supplemental considerations, uncertainties in estimates, and results of sensitivity analyses on the overall benefits. The results are presented in two different ways, in order to address the differing decision criteria between regulatory analyses and backfit analyses (10 CFR 50.109).

D.4.1 Regulatory Analysis

D.4.1.1 Summary Table

Table 103 summarizes the quantified net benefits used to perform a safety goal screening.

Table 103 Summary of Net Benefits for Low-density Spent Fuel Pool Storage Considering All Initiator Events (within 50 miles)

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$247,700	$179,500	$124,600	$119,700	$86,700	$60,200	$2,520,000	$1,825,500	$1,267,000
Occupational Health (Accident)	$1,300	$900	$700	$700	$500	$300	$21,300	$15,400	$10,700
Offsite Property	$723,300	$524,000	$363,700	$1,073,300	$777,500	$539,700	$4,587,800	$3,323,400	$2,306,700
Onsite Property	$10,400	$6,900	$4,300	$4,480	$2,950	$1,830	$378,600	$249,600	$155,800
Total Benefits	***$982,700***	***$711,300***	***$493,300***	***$1,198,200***	***$867,700***	***$602,000***	***$7,507,700***	***$5,413,900***	***$3,740,200***
Occupational Health (Routine)	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***
Net Benefit	**-$15,416,000**	**-$41,385,000**	**-$46,368,000**	**-$15,200,800**	**-$41,228,300**	**-$46,259,000**	**-$8,891,300**	**-$36,682,100**	**-$43,120,800**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

As shown in Table 103, the calculated net benefits for requiring low-density spent fuel pool storage at the reference plant does not achieve a positive net benefit using the current regulatory framework. This means that the calculated licensee costs to implement a low-density spent fuel pool storage alternative at the referenced site outweighs the estimated benefits.

Furthermore, for the seismic event analyzed for the Spent Fuel Pool Study, no offsite early fatalities are calculated to occur. This result is expected for two main reasons:

1. In comparison to reactors, spent fuel pools have a larger proportion of longer-lived radionuclides, which are less likely to cause the significant doses required for acute health effects.

2. Despite the large releases for certain predicted spent fuel pool accident progressions, the release from the most recently discharged fuel (which contains the shorter-lived radionuclides) is predicted to be insufficiently fast and insufficiently large to reach the acute thresholds associated with offsite early fatalities. When doses do exceed minimum levels for early fatalities, emergency response, as treated in the main report, effectively prevents any early fatality risk, at least in part because the modeled accident progression results in releases that are long compared with the time needed for relocation.

In addition, the predicted long-term exposure of the population, which could result in latent cancer fatality risk, is also low for the following reasons:

1. The individual latent individual latent cancer fatality risk within 0-10 miles for the studied scenarios is predicted to be on the order of 10^{-10} to 10^{-11} per year, based on the linear no threshold (LNT) dose response model.

2. The risk within 10 miles of the analyzed accident is dominated by low dose received at a low dose rate. According to alternate dose response models, excluding the uncertain effects of low radiation dose could reduce the quantified individual latent cancer fatality risk within 10 miles to be approximately 10^{-14} per year, a reduction of approximately 3,000 times.

3. Average individual latent cancer fatality risk is low and decreases slowly as a function of distance from the plant. Additionally, the predicted individual risks latent cancer fatalities are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough to be considered habitable. Therefore, the use of alternate dose response models would significantly reduce the quantified latent cancer fatalities by at least an order of magnitude.

D.4.1.2 Implementation and Operation Costs

Table 104 Summary of Total Implementation and Operation Costs for Low-density Spent Fuel Pool Storage for All Initiator Events

Attribute	Costs (2012 dollars in millions)	
	3% Net Present Value	7% Net Present Value
Occupational Health (Routine)	$0.024	$0.027
Industry Implementation	$41.800	$46.770
Industry Operation	$0.252	$0.064
NRC Implementation	nc	nc
NRC Operation	nc	nc
Total	$42.096	$46.861

As shown in Table 104, the total estimated costs for the referenced plant unit to achieve and maintain a low-density spent fuel pool loading range from $42 million (3 percent net present value) to $47 million (7 percent net present value). These costs are dominated by the capital costs for the DSCs and the loading costs for the storage systems to achieve low-density storage in the spent fuel pool than that required for the regulatory baseline.

D.4.1.3 Total Benefits and Cost Offsets

Table 105 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage for All Initiator Events

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	Undiscounted	3% Net Present Value	7% Net Present Value
Public Health (Accident)	$0.12 to $2.52	$0.09 to $1.83	$0.06 to $1.27
Occupational Health (Accident)	$0.001 to $0.021	$0.0005 to $0.015	$0.0003 to $0.011
Offsite Property	$0.72 to $4.59	$0.52 to $3.32	$0.36 to $2.31
Onsite Property	$0.004 to $0.38	$0.003 to $0.25	$0.002 to $0.16
Total	$0.85 to $7.51	$0.61 to $5.42	$0.42 to $3.75

The total benefits, which include the public health (accident) and occupational health (accident) is summed with the cost offsets, which include offsite property and onsite property relative to the regulatory baseline, are shown in the Table 105. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

D.4.1.4 Sensitivity Analysis

This section summarizes the results of the sensitivity analyses that were performed as an additional consideration in performing safety goal screening for requiring low-density spent fuel pool storage at the reference plant.

D.4.1.4.1 Dollar per Person-Rem Conversion Factor

The NRC is currently revising the dollar per person-rem averted conversion factor based on recent information regarding the value of a statistical life. However, until the NRC completes the update and publishes the appropriate guidance documents, the NRC performs sensitivity analysis to estimate the impact on the calculated results when more current VSL and cancer risk factor are used. The NRC used the U.S. Environmental Protection Agency's (EPA) VSL as an interim value in the sensitivity analysis as described in section D.3.3.2. The affect of this variable on the calculated results are provided in Table 106.

Table 106 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-density Spent Fuel Pool Storage Considering All Initiating Events (within 50 miles)

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$495,400	$359,000	$249,200	$239,400	$173,400	$120,400	$5,040,000	$3,651,000	$2,534,000
Occupational Health (Accident)	$2,600	$1,800	$1,400	$1,400	$1,000	$600	$42,600	$30,800	$21,400
Offsite Property	$723,300	$524,000	$363,700	$1,073,300	$777,500	$539,700	$4,587,800	$3,323,400	$2,306,700
Onsite Property	$10,400	$6,900	$4,300	$4,480	$2,950	$1,830	$378,600	$249,600	$155,800
Total Benefits	***$1,231,700***	***$891,700***	***$618,600***	***$1,318,600***	***$954,900***	***$662,500***	***$10,049,000***	***$7,254,800***	***$5,017,900***
Occupational Health (Routine)	-$18,000	-$48,000	-$54,000	-$18,000	-$48,000	-$54,000	-$18,000	-$48,000	-$54,000
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,408,000***	***-$42,120,000***	***-$46,888,000***	***-$16,408,000***	***-$42,120,000***	***-$46,888,000***	***-$16,408,000***	***-$42,120,000***	***-$46,888,000***
Net Benefit	**-$15,176,000**	**-$41,228,000**	**-$46,269,000**	**-$15,089,400**	**-$41,165,100**	**-$46,225,500**	**-$6,359,000**	**-$34,865,200**	**-$41,870,100**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

As shown in Table 106, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530.

D.4.1.4.2 Consequences Extending Beyond 50 Miles

The RA Handbook states that in the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile distance from the plant site, although alternative distances from the plant may be used for sensitivity analyses. For this regulatory analysis, supplemental information (e.g., analyses and results) based on MACCS2 calculated results, which extends the analysis beyond 50 miles from the postulated accident site is provided in Table 107.

Table 107 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-density Spent Fuel Pool Storage Considering All Initiating Events

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$1,783,400	$1,291,900	$896,700	$1,081,200	$783,300	$543,600	$15,735,800	$11,399,100	$7,911,700
Occupational Health (Accident)	$1,300	$900	$700	$700	$500	$300	$21,300	$15,400	$10,700
Offsite Property	$2,139,300	$1,549,700	$1,075,600	$4,968,300	$3,599,100	$2,498,000	$11,586,600	$8,393,400	$5,825,500
Onsite Property	$10,400	$6,900	$4,300	$4,680	$3,150	$2,030	$378,600	$249,600	$155,800
Total Benefits	***$3,934,400***	***$2,849,400***	***$1,977,300***	***$6,054,900***	***$4,386,100***	***$3,043,900***	***$27,722,300***	***$20,057,500***	***$13,903,700***
Occupational Health (Routine)	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***
Net Benefit	**-$12,465,000**	**-$39,247,000**	**-$44,884,000**	**-$10,344,100**	**-$37,709,900**	**-$43,817,100**	**$11,323,300**	**-$22,038,500**	**-$32,957,300**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

As shown in Table 107, calculated net benefits for requiring low-density spent fuel pool storage at the reference plant do not achieve a positive net benefit for eight of the nine cases presented. One case, the undiscounted high estimate, shows a positive net benefit of about $11.3 million, which reflects the value and impacts at the time in which they are incurred with no present worth conversion. It is informative to compare this value to the other high estimate values of ($22.0 million) and ($33.0 million), which differ from this case by adjusting these future costs into year 2012 dollars using 3-percent and 7-percent discount rates as described in section D.3.2.1.2.

D.4.1.4.3 Combined Effect of Consequences Extending Beyond 50 Miles and Dollar per Person-Rem Conversion Factor

This sensitivity analysis considers all initiating events that can challenge the reference plant's spent fuel pool cooling or integrity while taking into account the combined effects of extending the analysis of consequences beyond 50 miles from the site and increasing the dollar per person-rem conversion value from $2,000 to $4,000 per person-rem averted. The combined effects of these two variables on the calculated net benefits are provided in Table 108.

Table 108 Combined Sensitivity Analysis that Analyzes Consequences Beyond 50 Miles using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-density Spent Fuel Pool Storage for All Initiator Events

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$3,566,900	$2,583,800	$1,793,400	$2,162,500	$1,566,500	$1,087,300	$31,471,600	$22,798,200	$15,823,400
Occupational Health (Accident)	$2,500	$1,900	$1,400	$1,300	$1,000	$700	$42,700	$30,900	$21,400
Offsite Property	$2,139,300	$1,549,700	$1,075,600	$4,968,300	$3,599,100	$2,498,000	$11,586,600	$8,393,400	$5,825,500
Onsite Property	$10,400	$6,900	$4,300	$4,680	$3,150	$2,030	$378,600	$249,600	$155,800
Total Benefits	***$5,719,100***	***$4,142,300***	***$2,874,700***	***$7,136,800***	***$5,169,800***	***$3,588,000***	***$43,479,500***	***$31,472,100***	***$21,826,100***
Occupational Health (Routine)	-$18,000	-$49,000	-$54,000	-$18,000	-$49,000	-$54,000	-$18,000	-$49,000	-$54,000
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,408,000***	***-$42,121,000***	***-$46,888,000***	***-$16,408,000***	***-$42,121,000***	***-$46,888,000***	***-$16,408,000***	***-$42,121,000***	***-$46,888,000***
Net Benefit	**-$10,689,000**	**-$37,979,000**	**-$44,013,000**	**-$9,271,200**	**-$36,951,200**	**-$43,300,000**	**$27,071,500**	**-$10,648,900**	**-$25,061,900**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

As shown in Table 108, calculated net benefits for requiring low-density spent fuel pool storage at the reference plant do not achieve a positive net benefit for eight of the nine cases presented. One case, the undiscounted high estimate, shows a positive net benefit of about $27.1 million, which reflects the value and impacts at the time in which they are incurred with no present worth conversion. It is informative to compare this value to the other high estimate values of ($10.6 million) and ($25.1 million), which differ from this case by adjusting these future costs into year 2012 dollars using 3-percent and 7-percent discount rates as described in section D.3.2.1.2.

D.4.2 Backfit Analysis

As discussed above, the NRC has determined that the reference plant would not achieve a substantial increase in the protection of public health and safety from a change to low-density spent-fuel-pool storage. The NRC has therefore determined that imposing a requirement to use only low-density spent fuel pool storage at the reference plant would not meet the requirements of the backfit rule. However, to ensure that there is a complete discussion of these issues, the NRC has drafted an analysis of the costs associated with imposing these requirements as a backfit. This analysis of the direct and indirect costs of implementing the new requirements provides an assessment of the costs associated with imposing these requirements and the relative safety benefits in terms of the NRC's backfit rule. This backfit analysis examines the impacts of requiring low-density spent fuel pool storage at the reference plant relative to the baseline used in the regulatory analysis, which consists of existing requirements including the recently issued orders.

This plant-specific backfit analysis differs from most NRC's backfit analyses in that the NRC is not imposing or proposing to impose any requirements on its licensees. Instead, the NRC is assessing the safety benefits and costs of hypothetical requirements that, if implemented, would result in the use of low-density spent fuel pool storage and a corresponding increase in on-site dry cask storage for the reference plant. An NRC rulemaking to impose requirements like the ones analyzed in this appendix would need to include a backfit analysis. This section of the appendix provides a discussion of some of the elements that would be analyzed as part of a backfit analysis of these requirements. Prior to imposing these requirements through a rulemaking the NRC would, at the very least, issue a separate regulatory bases for public comment. If it is determined that rulemaking is required, the NRC would issue a proposed rule for public comment.

Low-density Spent Fuel Pool Storage Alternative Requirements that Constitutes a Plant-Specific Backfit for the Reference Plant

- All spent fuel assemblies that have cooled for at least five years (older spent fuel assemblies) after discharge from the reactor core are expeditiously moved from spent fuel pool storage from spent fuel pool storage to dry cask storage.

- The completion of the initial movement of older spent fuel assemblies to dry cask storage is achieved within five years of the effective date of the requirement.

- Following each refueling outage, the older spent fuel assemblies stored in the pool shall be moved to dry cask storage in a timely manner.

In performing this analysis, the NRC considered the nine factors in 10 CFR 50.109, as described in the following subsections.

D.4.2.1 General Description of the Activity Required at the Reference Plant to Complete the Backfit

The alternative would require that the licensee of the reference plant incur upfront costs, including engineering, design, and licensing costs; equipment costs; construction costs; and start up and testing costs, as necessary for their independent spent fuel storage installation to accept the dry storage cask systems. The licensee would also need to purchase and load dry storage casks on a periodic basis in compliance with the regulatory requirement.

D.4.2.2 Potential Change in the Risk to the Public from the Accidental Offsite Release of Radioactive Material

Table 109 Public Health (Accident) Person-Rem Averted

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
	Low Est.	Best Est.	High Est.	Undiscounted	3% Net Present Value			7% Net Present Value		
				Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Low-density storage	60	124	1,260	$247,700	$86,700	$179,500	$1,825,500	$60,200	$124,600	$1,267,000

Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

If the NRC were to implement the low-density storage proposal, the storage of spent fuel in dry storage casks would decrease the accidental offsite release of radioactive material from a postulated spent fuel pool accident. As Table 109 shows, dry cask storage at the reference plant would decrease the radiation exposure to the public by between 60 and 1,260 person-rem. The dose to the public mostly comes from the reoccupation of land after decontamination and the exposure to the workers who are decontaminating the public land. This analysis also assumes that 0.5% of the public will not evacuate during the accident. This resultant radiation dose is included within the public health exposure. As shown in the regulatory analysis, the best estimate benefits range from $0.18 million (3 percent net present value) to $0.12 million (7 percent net present value). A more in-depth review of the person-rem exposure to the public is found in section D.3.4.2.

D.4.2.3 Potential Impact on Radiological Exposure of Facility Employees

Table 110 Facility Employee Exposure

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
	Low Est.	Best Est.	High Est.	Undiscounted	3% Net Present Value			7% Net Present Value		
				Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
accident short-term	0.268	0.421	5.493	$841	$388	$628	$7,959	$269	$453	$5,524
accident long-term	0.068	0.208	5.170	$415	$98	$310	$7,490	$68	$223	$5,198
routine	-4.560	-4.560	-4.560	-$9,000	-$24,000	-$24,000	-$24,000	-$27,000	-$27,000	-$27,000
Total	-4.224	-3.932	6.103	-$7,744	-$23,514	-$23,063	-$8,552	-$26,662	-$26,324	-$16,278

Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

If imposed on licensees, these requirements would provide added assurance that nuclear industry workers are not subjected to unnecessary radiological or hazardous chemical exposures as the result of mitigative and clean-up activities associated with a spent fuel pool accident that results in a radioactive release. Storage of spent fuel in dry storage casks would decrease the post-accidental offsite radiation exposure to facility employees from a postulated spent fuel pool accident. The exposure of facility employees comes from a short-term dose, based on the exposure during the accident, and a long-term dose, based on the exposure from the onsite cleanup costs. Facility employees, however, receive additional radiation exposure during DSC loading and handling activities, ISFSI operations, and maintenance and surveillance activities, resulting in a net increase in radiation exposure as shown in Table 110 for the low and best estimates. A more in-depth discussion of the person-rem exposure to facility employees can be found in sections D.3.4.2 and D.3.4.3.

D.4.2.4 Installation and Continuing Costs Associated with the Backfit, including the Cost of Facility Downtime or the Cost of Construction Delay

Table 111 Installation and Continuing Costs Associated with the Backfit

Case	Costs (2012 dollars)		
	Undiscounted	3% NPV	7% NPV
Implementation costs	-$15,660,000	-$41,820,000	-$46,770,000
Operation costs	-$730,000	-$252,000	-$64,000
Total	-$16,390,000	-$42,072,000	-$46,834,000

Implementation and continuing costs include the upfront costs, which include engineering, design, and licensing costs; equipment costs; construction costs; and start up and testing costs, as necessary, for the reference plant's independent spent fuel storage installation to accept the dry storage cask systems. In addition, the licensee would need to purchase and load dry storage casks on a periodic basis in compliance with regulatory requirements. As these actions are assumed not to affect normal power operations, there are no assumed replacement energy costs or construction delays. A more detailed analysis of the industry implementation and operation costs is provided in sections D.3.4.6 and D.3.4.7.

D.4.2.5 Potential Safety Impact of Changes in Plant or Operational Complexity, including the Relationship to Proposed and Existing Regulatory Requirements

If imposed on licensees, these requirements are not expected to have a significant effect on facility complexity. The scheduling and performance of loading spent fuel assemblies from the spent fuel pool into casks and transporting them to the ISFSI would add additional complexity to plant operations, especially during the initial 5-year loading phase. The added plant operations complexity is not significant and will not substantially affect the reference plant operational practices or result in substantial indirect costs. However, should a cask drop accident occur during plant operation, even though its likelihood is remote, the event could challenge plant safety systems in mitigating the consequences.

D.4.2.6 Estimated Resource Burden on the NRC Associated with the Proposed Backfit and the Availability of Such Resources.

The establishment of the requirements needed to require the reference plant to move expeditiously all spent fuel assemblies that have cooled for at least five years (older spent fuel assemblies) after discharge from the reactor core from spent fuel pool storage to dry cask storage would require rulemaking. The rulemaking would not result in a substantial increase in annual expenditures of agency resources.

D.4.2.7 Potential Impact of Differences in Facility Type, Design, or Age on the Relevancy and Practicality of the Proposed Action

There is no expected significant differentiation in how individual plants would implement the requirement to expeditiously move all spent fuel assemblies that have cooled for at least five years (older spent fuel assemblies) after discharge from the reactor core from spent fuel pool storage to dry cask storage. If imposed on licensees, these requirements do not directly relate to the facility type, design, or age.

D.4.2.8 Whether the Proposed Backfit is Interim or Final and, if Interim, the Justification for Imposing the Proposed Backfit on an Interim Basis

This consideration is not relevant to the analysis at this time because no requirements are being proposed.

D.4.2.9 Other Information Relevant and Material to the Proposed Backfit

Table 112 Summary of Backfitting Net Benefits for Low-density Spent Fuel Pool Storage for All Initiator Events (within 50 miles)

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$247,700	$179,500	$124,600	$119,700	$86,700	$60,200	$2,520,000	$1,825,500	$1,267,000
Occupational Health (Accident)	$1,300	$900	$700	$700	$500	$300	$21,300	$15,400	$10,700
Occupational Health (Routine)	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000
Total Benefits	***$240,000***	***$156,400***	***$98,300***	***$111,400***	***$63,200***	***$33,500***	***$2,532,300***	***$1,816,900***	***$1,250,700***
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***
Net Benefit	**-$16,150,000**	**-$41,916,000**	**-$46,736,000**	**-$16,279,000**	**-$42,009,000**	**-$46,801,000**	**-$13,858,000**	**-$40,255,000**	**-$45,583,000**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

Table 112 summarizes the described benefits and costs associated with the proposed backfit to require the reference plant to expeditiously move all older spent fuel assemblies after discharge from the reactor core from spent fuel pool storage to dry cask storage. The analyzed alternative would also incur onsite and offsite property cost offsets from an accident. These cost offsets are summarized in Table 113

Table 113 Summary of Cost Offsets for Onsite and Offsite Property

Attribute	Total Cost Offsets								
	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Offsite Property	$723,300	$524,000	$363,700	$1,073,300	$777,500	$539,700	$4,587,800	$3,323,400	$2,306,700
Onsite Property	$10,400	$6,900	$4,300	$4,480	$2,950	$1,830	$378,600	$249,600	$155,800
Total Benefits	***$733,700***	***$530,900***	***$368,000***	***$1,077,800***	***$780,500***	***$541,500***	***$4,966,400***	***$3,573,000***	***$2,462,500***

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

Table 114 Combined Sensitivity Analysis of the Backfitting Net Benefits for Low-density Spent Fuel Pool Storage for All Initiator Events (extending analysis beyond 50 miles and using a Revised Dollar per Person-Rem Conversion Factor)

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$3,566,900	$2,583,800	$1,793,400	$2,162,500	$1,566,500	$1,087,300	$31,471,600	$22,798,200	$15,823,400
Occupational Health (Accident)	$2,500	$1,900	$1,400	$1,300	$1,000	$700	$42,700	$30,900	$21,400
Occupational Health (Routine)	-$18,000	-$49,000	-$54,000	-$18,000	-$49,000	-$54,000	-$18,000	-$49,000	-$54,000
Total Benefits	***$3,551,400***	***$2,536,700***	***$1,740,800***	***$2,145,800***	***$1,518,500***	***$1,034,000***	***$31,496,300***	***$22,780,100***	***$15,790,800***
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***
Net Benefit	**-$12,838,600**	**-$39,535,300**	**-$45,093,200**	**-$14,244,200**	**-$40,553,500**	**-$45,800,000**	**$15,106,300**	**-$19,291,900**	**-$31,043,200**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

Table 114 summarizes the results of the combined sensitivity analyses that extended the backfitting net benefit analysis beyond 50 miles from the plant site and used a higher per person-rem conversion factor to monetize averted dose. The analyzed alternative would also incur onsite and offsite property cost offsets from an accident. These cost offsets for the combined sensitivity analysis are summarized in Table 115.

Table 115 Summary of Combined Sensitivity Analysis Cost Offsets for Onsite and Offsite Property

Attribute	Total Cost Offsets								
	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Offsite Property	$2,139,300	$1,549,700	$1,075,600	$4,968,300	$3,599,100	$2,498,000	$11,586,600	$8,393,400	$5,825,500
Onsite Property	$10,400	$6,900	$4,300	$4,680	$3,150	$2,030	$378,600	$249,600	$155,800
Total Benefits	***$2,149,700***	***$1,556,600***	***$1,079,900***	***$4,973,000***	***$3,602,300***	***$2,500,000***	***$11,965,200***	***$8,643,000***	***$5,981,300***

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

D.4.3 Disaggregation

In order to comply with the guidance provided in Section 4.3.2 (“Criteria for the Treatment of Individual Requirements”) of the Regulatory Analysis Guidelines, the NRC conducted a screening review to ensure that the aggregate analysis does not mask the inclusion of individual rule provisions that are not cost-beneficial when considered individually and not necessary to meet the goals of the rulemaking. Consistent with the Regulatory Guidelines, the NRC evaluated, on a disaggregated basis, each new regulatory provision expected to result in incremental costs. Based on this screening review, the NRC did not identify any requirements needing further consideration. The NRC believes that each of these provisions described in section D.4.2 is necessary in the aggregate for the expedited transfer of spent fuel to DSCs. However, as noted above, the Commission has not found that accelerated transfer to DSCs to provide a substantial safety benefit, nor to be cost justified.

D.4.4 Safety Goal Evaluation

Safety goal evaluations are applicable only to regulatory initiatives considered to be generic safety enhancement backfits subject to the substantial additional protection standard in 10 CFR 50.109(a)(3).

The frequency of damage to the spent fuel is estimated to be range from 7.11×10^{-7} to 5.39×10^{-6} per reactor-year when considering all initiators that could challenge spent fuel pool cooling or integrity. These values, when compared to a target value of 1×10^{-4}, which is the quantitative health objective for latent cancer fatalities derived using reactor accident characterizations, represents a 0.71% to 5.39% of the overall frequency of core damage.

The frequency of a release of radioactive material to the environment is assumed to be the same as the frequency of spent fuel damage. The reactor building, which houses the spent fuel pool, does not provide a containment barrier similar to the containment structure surrounding the reactor core, especially under the conditions postulated to dominate the release of radioactive materials from spent fuel.

It is difficult to compare the estimated 7.11×10^{-7} to 5.39×10^{-6} per reactor-year release frequencies for the postulated spent fuel pool accident when considering all initiators to a target value of 1×10^{-5} per reactor year for a large early release frequency (LERF). The spent fuel pool source term is not similar to the core damage (or melt) source term for which the consequences of a spent fuel pool accident are predicted to have no early fatalities and public health risk is dominated by latent cancer risks resulting from long-term exposures. Because the analyzed spent fuel accident is a slow progression with at least eight hours before an environmental release occurs, and the resultant release is not expected to result in any offsite early fatalities, the analysis suggests that the spent fuel pool release does not fall within the definition of a large early release. Although this analyzed accident is different from a reactor accident, the spent fuel pool estimated release frequencies of 7.11×10^{-7} to 5.39×10^{-6} per reactor-year meets the 1×10^{-5} LERF guidelines.

Societal risk is based on the statistically expected number of early and latent cancer fatalities. The Safety Goals for the Operation of Nuclear Power Plants: Policy Statement (51 FR 28044) defines the early fatality area calculation as that within one mile from the site boundary. As discussed above, the resultant release is not expected to result in any offsite early fatalities. A ten-mile radius is defined for calculating latent cancer fatalities. The second quantitative objective of the Policy Statement is for the risk to the population in the vicinity of a nuclear power plant from an accident at a nuclear power plant should not exceed 0.1 percent of the sum of cancer fatality risks resulting from all other causes. Based on recent data (http://www.cancer.org/research/cancerfactsfigures/index) the total fatality rate from cancer in the U.S. is 580,350 per 315,747,500 persons (http://www.census.gov/popclock/) or a risk of 1.84×10^{-3} per year, which results in a safety goal of 1.84×10^{-6} per year. Using the bounding frequency of damage to the spent fuel of 5.39×10^{-6} per reactor-year, which considers all initiators that could challenge spent fuel pool cooling or integrity, and the conditional individual latent cancer fatality risk within a ten-mile radius is 4.4×10^{-4} yields a bounding latent cancer fatality risk of 2.37×10^{-9} of cancer fatality per year. This calculated value of 2.37×10^{-9} latent cancer fatalities per reactor-year associated with a spent fuel pool accident is less than represents a 0.13% fraction of the 1.84×10^{-6} per year societal risk goal value based on the calculation area specified in the Safety Goal Policy Statement.

Therefore, the risk of a spent fuel pool accident at the reference plant appears to meet the Safety Goal Policy Statement public health objectives. They also meet the $1x10^{-5}$ per reactor-year LERF guideline. Therefore, the Regulatory Baseline is justified for the alternative described in section D.2.2 as evaluated for the reference plant.

D.4.5 CRGR Results

This section addresses regulatory analysis information requirements for rulemaking actions or staff positions subject to review by the Committee to Review Generic Requirements (CRGR). All information called for by the CRGR is presented in this regulatory analysis.

D.5 DECISION RATIONALE

This section presents the decision rationale, including the basis for selection, any decision criteria used, the regulatory instrument to be used (if applicable), and the statutory basis for the selected regulatory action. The decision rationale is presented in two different ways, in order to address the differing decision criteria between regulatory analyses and backfit analyses (10 CFR 50.109).

D.5.1 Regulatory Analysis

Table 103 shows that a requirement for low-density spent fuel storage alternative does not achieve a cost-beneficial increase in public health and safety for the reference plant using the current regulatory framework when all event initiators, which may challenge spent fuel cooling or pool integrity, are considered. Furthermore, the three sensitivity studies provided in section D.4.1.4 also showed that the low-density spent fuel storage alternative was not cost-justified for any of the discounted sensitivity cases.

The NRC believes that there are other considerations discussed in section D.3.4.10 that would further reduce the quantified benefits and make the proposed alternative less justifiable. Based on the NRC's assessment of the costs and benefits, the agency has concluded that the risk due to beyond design basis accidents in spent fuel pools, while not negligible, is sufficiently low that the added costs involved with expediting the movement of spent fuel from the pool to achieve the low-density fuel pool storage alternative evaluated for the reference plant is not warranted.

D.5.2 Backfit Analysis

The NRC conducted a backfit analysis for the reference plant relative to the backfit requirements in 10 CFR 50.109. The NRC does not believe that this alternative results in a cost-justified substantial safety enhancement for the reference plant. First, the risk of a spent fuel pool accident at the reference plant appears to meet the Safety Goal Policy Statement public health objectives. The estimated spent fuel pool accident release frequency is also less than the $1x10^{-5}$ per reactor-year LERF guideline. Second, the cost-justified criteria are not met when evaluating the averted accident consequences within 50 miles of the site consistent with the regulatory framework. Sensitivity analyses that extend the analyses beyond 50 miles also show that the low-density spent fuel storage alternative was not cost-justified for any of the discounted sensitivity cases. Therefore, the Regulatory Baseline is justified for the alternative described in section D.2.2 as evaluated for the reference plant.

In light of the findings above, the NRC concludes that the quantified safety benefits of the proposed rule provisions that qualify as backfits, considered in the aggregate, would not constitute a substantial increase in protection to public health or safety or the common defense and security, and the costs of this rule would not be justified in view of the increase in protection to safety and security provided by the backfits embodied in the proposed rule.

D.5.3 Conclusion

The regulatory screening analysis and the backfitting discussion in this appendix indicate that for the reference plant a requirement for low-density spent fuel pool storage, and an associated requirement for expedited transfer of spent fuel from the spent fuel pool to meet a low-density spent fuel pool storage requirement, are not justified.

The risk due to beyond design basis accidents in the spent fuel pool analyzed in this study, is sufficiently low that the added costs involved with expediting the movement of spent fuel from the pool to achieve the low-density fuel pool storage alternative are not warranted. While the expedited fuel movement alternative evaluated is not cost-beneficial, the report has discovered that an alternative 1x8 high-density fuel configuration may have significantly lower costs in implementation and potentially similar benefits to the low-density configuration. This alternative should be evaluated further, in addition to other possible spent fuel pool loading configurations, as part of the regulatory analysis for expedited fuel movement described in SECY-12-0095 to evaluate the transfer of spent fuel to dry cask storage for existing and new (future) nuclear power plants.

D.6 REFERENCES

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Attachment 4

Official Transcript of Proceedings (ML13277A215) ("9/18 Transcript")

Public Meeting

Japan Lessons Learned Project Directorate

September 18, 2013

(Excerpt)

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Japan Lessons Learned Project Directorate
Public Meeting

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Wednesday, September 18, 2013

Work Order No.: NRC-272 Pages 1-245

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NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

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JAPAN LESSONS LEARNED PROJECT DIRECTORATE

+ + + + +

PUBLIC MEETING

+ + + + +

WEDNESDAY, SEPTEMBER 18, 2013

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The meeting was convened in the Commissioners' Hearing Room, One White Flint North, 11545 Rockville Pike, Rockville, Maryland, at 10:00 a.m., Lance Rakovan, moderating.

PRESENT:

BRIAN SHERON, Director, Office of Nuclear Regulatory Research

JENNIFER UHLE, Deputy Director for Reactor Safety Programs, NRR

HOSSEIN ESMAILI, Senior Reactor Systems Engineer

STEVEN JONES, Senior Reactor Systems Engineer, DSS

JOSE PIRES, Senior Technical Advisor for Civil Engineering

KEVIN WITT, Project Manager, Japan Lessons Learned Project Directorate

NRC STAFF PRESENT:

LANCE RAKOVAN

SCOTT BURNELL

KEITH COMPTON

LYNNE FINCH

LAUREN GIBSON

A.J. NOSEK

FRED SCHOFER

RANDY SULLIVAN RALPH WAY

T-A-B-L-E O-F C-O-N-T-E-N-T-S

like to go to John Sipos, David Weisman and then Tom Cochran. We'll take an hour for lunch. So let's try to be back here at 1 o'clock so we can get started shortly after 1:00.

(Whereupon, the foregoing matter went off the record at 12:00 p.m. and went back on the record at 1:00 p.m.)

MR. RAKOVAN: Welcome back, everyone. I hope everyone had at least a fair lunch, if not a good one. I'm going to continue. This is Lance Rakovan again, facilitating the meeting.

I'm going to continue to go through those who pre-signed up to speak today. And we'll hopefully try to get through everyone in the next four hours.

As I said before lunch, we're going to go to John Sipos, then David Weisman, and then Tom Cochran. I'll try to give a three person, you know, queue if you will so people know when their time is coming up so they can prepare. So Mr. Sipos, if you would, please.

MR. SIPOS: Thank you very much. Good afternoon, everyone. My name is John Sipos, for those of you who I haven't met. On behalf of the State of New York for whom I work, I would like to express the thanks to NRC and to the distinguished group of people here today for holding this public meeting.

It's very important, I think, for the process. And it is an important issue, and it is an important issue to the State. So thank you very much. I appreciate that very much.

Just one question I had at the beginning. Is this hearing or meeting being transcribed? I think there was a question about that.

MR. RAKOVAN: Yes, the meeting is being both transcribed, and since we are webcasting it, we should have the archive of that, as well.

MR. SIPOS: Fantastic. Some preliminary questions, and I guess I'll direct them either to Dr. Sheron or Dr. Uhle or whoever else is on the panel. But as I understand the consequence study, it examined a type of severe accident at a spent fuel pool at the Peach Bottom Atomic Power Reactor Site, correct?

MR. SHERON: Yes.

MR. SIPOS: And so the consequence study was a site specific severe accident analysis of a spent fuel pool accident, is that correct?

MR. SHERON: Yes, it was for one reactor.

MR. SIPOS: Okay. And the consequence study used a computer code known as MACCS, M-A-C-C-S, numeral 2?

MR. SHERON: Yes, I think, yes that was the correct one.

MR. SIPOS: And from our experience in the Indian Point proceedings, we understand that and I think Dr. Ooly, is it Ooly or --

MS. UHLE: It's Uhle. But I answer to most everything.

MR. SIPOS: Uhle, excuse me. Uhle, I'll try to pronounce that correctly. Thank you. That's for the MELCOR Accident Consequence Code System, I guess that's the acronym, is that correct?

And from our experience in Indian Point, we understand that it's also used for site specific severe reactor accident analyses as well, correct?

MR. SHERON: Yes.

MR. SIPOS: Okay. And amongst the NRC staff, can you tell us who was the principal author of Chapter 7 of the consequence study? Understanding you all work as a team. Yes, sir and I haven't met you so I'm not sure who you are.

MR. NOSEK: Hi, my name is A.J. Nosek. I'm from the Office of Research.

MR. COMPTON: I'll introduce myself. I've worked also with A.J. on Chapter 7, and a few of the other consequence pieces. I'm Keith Compton.

MR. SIPOS: Thank you very much. And as

part of the MACCS2 analysis that was done, who performed that aspect of the consequence study?

MR. NOSEK: I did.

MR. SIPOS: And so you were responsible for the inputs that were made to the MACCS2 code analysis, is that correct?

MR. NOSEK: Yes.

MR. SIPOS: Okay. And did you also make the decisions as to what values should be used for the inputs?

MR. NOSEK: Yes.

MR. SIPOS: Okay. And what version of the MACCS2 code did you use? I read in the report I think it was revision 3.7.0?

MR. NOSEK: Correct.

MR. SIPOS: Okay. And was it MACCS2 or WinMACCS?

MR. NOSEK: So WinMACCS is the user interface that we now have a framework for MACCS2. So MACCS2 is one of the components within the WinMACCS interface. So you could say I use WinMACCS/MACCS2.

MR. SIPOS: Okay, thank you. And how many runs of the MACCS2 code were performed?

MR. NOSEK: That's a good question. It depends on what you consider a code calculation and for what purpose. For our base case scenarios of these spent fuel pool study they were looking at, we had seven major source terms we were looking at.

And of those, we had a number of different weather trials within there. And yes? We had upwards of 1,000 weather trials per scenario, and we also looked at a number of different sensitivities within those base cases for different dose truncations or NLNC calculations. So seven times three times upwards of 1,000.

MR. SIPOS: And we were using the term run. I've also seen the term case used with respect to MACCS. Are those interchangeable in your understanding or in your parlance?

MR. NOSEK: Yes. It depends on the context.

MR. SIPOS: And you also mentioned sensitivity studies, or sensitivity analyses. Those also factor into the number of runs that were performed, is that correct?

MR. NOSEK: There was additional runs done for sensitivities. Each in both the, I believe, Chapter 9? I don't know if it's still Chapter 9, but the sensitivities chapter as well as in support of the regulatory analysis as well.

MR. SIPOS: Okay. And did both of you also work on Chapter 9? I should have asked that earlier. Yes?

MR. NOSEK: Yes.

MR. SIPOS: Thanks. And do you know roughly when those MACCS runs were performed?

MR. NOSEK: The final calculations were in the span of around November and December of last year.

MR. SIPOS: Okay, 2012. And there were earlier runs done, as well, it sounds like?

MR. NOSEK: Yes. I mean, we will, as we refine our calculations will be doing a number of different runs.

MR. SIPOS: And were each of those runs documented in some manner?

MR. NOSEK: The ones that were documented were the final runs and the sensitivities.

MR. SIPOS: Okay. And were the runs that were done prior to the end of 2012, were they also documented?

MR. NOSEK: I do not believe they were documented in the final report.

MR. SIPOS: Would it be possible for the state to get copies of the input and output files for the runs for which there is documentation?

MS. UHLE: The question of what we have as far as distribution is if there was any proprietary information from the site. So can we get back to you on that question? And the only concern would be the proprietary nature of the data, recognizing your state.

I know there's different arrangements that can be made. So I think it's hard to answer just off the top of our head.

MR. SIPOS: Okay, well --

MS. UHLE: We can meet with you after the meeting and continue the discussion. That would be helpful to us.

MR. SIPOS: I appreciate that.

MR. NOSEK: We leveraged to allow the best practices from the SOARCA report. And we do have a report becoming available that much of those inputs will become available in that document.

MR. SIPOS: And just to close the circle on that, this state is interested in seeing the input and outputs and the results. What went into the runs, what the runs generated, so that we could look at it as well. Thank you.

And were there quality assurance or quality control aspects of the runs? Did either of you perform QA/QC on the runs?

MR. NOSEK: Yes. I mean, one of the number of reasons that we do a number of calculations up into our final runs is as a quality assurance measure.

In addition, we also had our subject matter expert from Sandia review all the inputs. And we also had the ACRS review our report.

MR. SIPOS: And the subject matter expert from Sandia, would that be Nate Bixler?

MR. NOSEK: Correct.

MR. SIPOS: And Joe Jones?

MR. NOSEK: Correct.

MR. SIPOS: And the rest of the Sandia people that are listed, I think, on the second or third page of the report?

MR. NOSEK: Not off the top of my head.

MR. SIPOS: I could read --

MR. NOSEK: I do not know who's on that paper.

MS. UHLE: I believe some of those people are the MELCOR support.

MR. SIPOS: Right, there is --

MS. UHLE: So we can't say off the top of our head whether or not they are all for MACCS.

MR. SIPOS: Okay. But Nate Bixler is, I guess, the custodian of the code for Sandia? So he was involved in it, correct?

MR. NOSEK: Correct.

MR. SIPOS: And Mr. Jones, as well?

MR. NOSEK: Yes.

MR. SIPOS: Okay. And I think this is set out in the reporter, the information digest. But our understanding is that the Peach Bottom site has two reactors, each with a spent fuel pool.

So there's two pools, two reactors at the Peach Bottom site, correct? And this study looked at an accident to one of those pools, correct?

MR. SHERON: Yes.

MR. SIPOS: Okay. And Peach Bottom is located central Pennsylvania roughly, I don't know, 18 miles from Lancaster, Pennsylvania, correct?

MR. SHERON: I believe so, yes.

MR. SIPOS: Okay. And I checked the 1996 generic environmental impact statement for license renewal. And I think as of 1990, which was the population data that was used in this study, there were roughly 4.7 million people that lived within a 50 mile radius of Peach Bottom. Is that square with your general knowledge? I got it from the GEIS at Table 2.1.

MR. NOSEK: I do not recall the population off the top of my head. But that seems feasible.

MR. SIPOS: And I have a couple of questions that I think Lance, I'm sorry excuse me, that Kevin was discussing this morning concerning the relationship with other activities that NRC is conducting right now.

And I think on the PowerPoint that you handed out this morning, Page 14, it looks like the consequence study is expected to be finalized very soon by NRC staff, correct?

MR. WITT: Yes, they are both expected to be provided at the commission on or before October 11th.

MR. SIPOS: Okay. And that would be before the public comment period ends on the waste confidence proceeding, is that correct?

MR. WITT: That is correct. I believe the waste confidence comment period ends late November.

MR. SIPOS: Right, around Thanksgiving, I think. And I think going back to Page 4 of the hand out from this morning, there was a statement that the schedules have been aligned to facilitate public involvement with the Tier 3 issue, the study, and ongoing waste confidence activities and relating policy issues. And it sounds like that is exactly what is going on, correct?

MR. WITT: That is correct.

MR. SIPOS: And just to be clear, when you talk about the Tier 3 issue, you're talking about the regulatory analysis or Appendix D that is attached to the consequence study?

MR. WITT: Those are two slightly different documents. The Tier 3 analysis is a generic regulatory analysis applicable to all plants. The Appendix D of the spent fuel pool study was done for that specific plant studied in the report.

MR. SIPOS: Okay, thank you. I appreciate that clarification. I guess I would like to come back to the MACCS2 issues that were part of the consequence study. Could you tell us what role Dr. Bixler played in the MACCS2 analyses that were done?

MR. NOSEK: Nate Bixler is a consultant, and he also is a lead developer for the MACCS2 code. And so we use him as consulting support. But we did the calculations and the model development in-house.

MR. SIPOS: And did he make any suggestions regarding inputs or assumptions to any of the inputs?

MR. NOSEK: Yes, where necessary.

MR. SIPOS: And could you summarize what those suggestions were by Dr. Bixler?

MR. NOSEK: On an overall scheme of things, the models that we started with were leveraged from SOARCA. So our initial starting point was harnessing the best practices from that report, which also is Peach Bottom, which has site specific meteorology and geography.

So it's also very applicable to our site. And starting from there, we took the source terms generated from the MELCOR code to make it specific to the spent fuel pool study, as well as updates regarding the emergency response aspects. And a few variety of small changes to inputs from different areas.

MR. SIPOS: And did Mr. Jones make any recommendations?

MR. NOSEK: Mr. Jones was assisting NSRG in recommendations for the emergency preparedness and the emergency response and all the protective actions in that part of the code.

MR. SIPOS: And when you refer to protective actions, are you referring to the protection action guidelines that EPA has developed?

MR. NOSEK: Partly. Bottling the emergency response and evacuation as a whole. So including emergency phase relocation, evacuation, shelter in place, and setting up an appropriate response based on the site's emergency action levels, and the specific evacuation time estimates.

MR. SIPOS: There's another individual at

Sandia, I may be mispronouncing his name, Randal is it Gauntt, and my understanding is he's done work with MELCOR as opposed to MACCS. Did he have any involvement with the MACCS analysis that was performed as part of the consequence study?

MR. NOSEK: Not directly.

MR. SIPOS: But he does have experience with MELCOR, correct?

MR. NOSEK: Correct.

MR. SIPOS: Okay. And did he work on any of the MELCOR aspects of the consequence study?

MR. ESMAILI: No, he did not.

MR. SIPOS: Thank you.

MR. RAKOVAN: Sir, just one or two more questions, if you wouldn't mind. Sorry, I'm sitting down right here. I was trying to stay out of the way of everybody. Just a couple more questions, and then we'll move on to the next speaker, please.

MR. SIPOS: It also appears that Oak Ridge National Laboratories had some role in the consequence study. Could one of the NRC staff members here summarize the role of Oak Ridge?

MR. ESMAILI: Oak Ridge did two things for us. First, provided the inventories, you know, radionuclide inventories. So they did a scale origin calculations for us.

And they also did, you know, doses in the refueling flow, once the spent fuel pool becomes uncovered to see what the radiation levels would be. These are all documented in the report. I believe it's in Chapter 5.

MR. RAKOVAN: Thank you, Hossein.

MR. ESMAILI: Yes.

MR. RAKOVAN: I was just trying to get your name into the statement, that's all.

MR. ESMAILI: Why he keeps telling it. Sorry.

MR. SIPOS: Thank you very much. And also, there was a company, DAKOTA, LLC. Could anyone describe what their role was? I may be mispronouncing it.

MR. ESMAILI: Correct. The individual is Casey Wagner. He was, at the time, he's right now at DAKOTA but he used to be at Sandia. So he was involved in, you know, the MELCOR code development, applications, et cetera. So we used him to some extent, you know, as a consultant.

MR. SIPOS: But it was on the MELCOR side of --

MR. ESMAILI: On the MELCOR side.

MR. SIPOS: Thank you. Just I notice there were three people involved in the study who's last name was Wagner. Any relationship amongst them?

MR. ESMAILI: No.

MR. SIPOS: Thank you. I do have further questions. Thank you. I do have further questions, but recognizing that there are a number of people, as I said, my flight is very late. I'm happy to have other people --

MR. RAKOVAN: If we have time, we'll loop around to you.

MR. SIPOS: Thank you very much.

MR. RAKOVAN: Okay, thank you. If we could go to David Weisman, who I believe is on the phone, followed by Tom Cochran and then Kyle Landis-Marinello. Operator, can we see if David Weisman is on the phone, please?

OPERATOR: Yes, please press Star 1 if you are connected.

MR. RAKOVAN: Mr. Weisman, are you there?

MR. WEISMAN: Hello. Are you there?

OPERATOR: Mr. Weisman, your line is now open.

MR. RAKOVAN: Please go ahead, David. We can hear you.

MR. WEISMAN: David Weisman, Alliance for

Attachment 5

Various NYS & NRC Communications Re Request for MACCS2 Input and Output Files
For Spent Fuel Pool Consequence Study

John J. Sipos

From: John J. Sipos
Sent: Thursday, August 22, 2013 12:07 PM
To: Witt, Kevin (Kevin.Witt@nrc.gov)
Subject: FW: NRC Tier 3 Spent Fuel Transfer Public Meeting

Importance: High

Kevin:

On behalf of the State of New York, I waited more than 1 hour 15 minutes+ to ask questions on behalf of an Agreement State that hosts various facilities. This is not acceptable. When can representatives of the State meet with the federal employees responsible for this study and discuss the State's concerns and questions?

John Sipos
Assistant Attorney General
518-402-2251

-----Original Message-----
From: John J. Sipos
Sent: Thursday, August 22, 2013 11:41 AM
To: 'Witt, Kevin'
Subject: RE: NRC Tier 3 Spent Fuel Transfer Public Meeting
Importance: High

Kevin:

I've been waiting for 55+ minutes on the phone after pressing *1 -- while others have had multiple occasions to speak.

I would appreciate the opportunity to ask questions on behalf of the State of New York, which is an agreement state.

John

-----Original Message-----
From: Witt, Kevin [mailto:Kevin.Witt@nrc.gov]
Sent: Thursday, August 22, 2013 6:53 AM
To: John J. Sipos
Subject: RE: NRC Tier 3 Spent Fuel Transfer Public Meeting

Hi John, the phone line is 888-831-8980 and passcode 8431922. Please call 301-415-3091 if you have any concerns connecting.

Thanks,
Kevin

____

From: John J. Sipos [John.Sipos@ag.ny.gov]
Sent: Wednesday, August 21, 2013 6:57 PM
To: Witt, Kevin
Subject: RE: NRC Tier 3 Spent Fuel Transfer Public Meeting

Hello Kevin:

Thank you for your email.

I understand that in recent days there have been several changes to the call in number and pass code for tomorrow's public meeting. Please let me know the number/code that will be operative tomorrow morning. Thank you.

John Sipos
Assistant Attorney General
tel. 518-402-2251

[cid:image001.gif@01CE9EA0.50923C40]

From: Witt, Kevin [mailto:Kevin.Witt@nrc.gov]
Sent: Monday, August 12, 2013 3:26 PM
To: John J. Sipos
Subject: NRC Tier 3 Spent Fuel Transfer Public Meeting

Mr. Sipos, I understand that you have submitted a comment on the NRC's Spent Fuel Pool Study and thought you may be interested in a related activity that we have underway. Please see the following website for the public meeting announcement on the Tier 3 issue for expedited transfer of spent fuel to dry cask storage http://www.nrc.gov/public-involve/public-meetings/index.cfm?action=search.detail&MeetingCode=20130735. We plan to have the documents available later this week for the public to review.

We hope that you are able to attend the meeting (we will also accommodate remote participation) to provide feedback on the staffs work. Please let me know if I can provide any additional information.

Best Regards,

Kevin

Kevin Witt
Project Manager
Japan Lessons Learned Project Directorate Office of Nuclear Reactor Regulation US Nuclear Regulatory Commission
Washington, DC 20555 Office (301) 415-2145

John J. Sipos

From: John J. Sipos
Sent: Sunday, September 29, 2013 2:22 PM
To: Witt, Kevin (Kevin.Witt@nrc.gov); William.Reckley@nrc.gov
Subject: follow up re consequence study analyses

Hello William and Kevin:

The State of New York is following up on its request for the documentation/files reflecting the MACCS runs performed in connection with the 2011-13 Spent Fuel Pool Consequence Study.

When do you anticipate that the State will receive them?

Thank you.

John Sipos
518-402-2251

John J. Sipos

From: Witt, Kevin <Kevin.Witt@nrc.gov>
Sent: Wednesday, October 23, 2013 5:22 PM
To: John J. Sipos
Subject: State of NY Request for MACCS2 Files

Dear Mr. Sipos,

During the September 18, 2013 public meeting and in your September 29, 2013 email, you requested that the State of New York be provided the documentation/files reflecting the MACCS runs performed in connection with the recently completed Spent Fuel Pool Study (SFPS). The final SFPS can be accessed at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2013/2013-0112scy.pdf. The NRC is making the files from the SFPS publicly available through the NRC's Agencywide Documents Access and Management System (ADAMS) under accession number ML13282A535. ADAMS can be accessed at http://adams.nrc.gov/wba/. These files contain the information from the MACCS2 input and output files for the SFPS. The output files represent a site-specific analysis of a highly unlikely spent fuel pool accident at a BWR plant with Mark I containment reference plant. As with other BWR plants with Mark I containments, the spent fuel pool is located outside the containment structure and generally contains several times the source term found in an operating reactor. The MACCS2 code is a widely accepted accident consequence analysis code used to model various types of radiological accidents. If you have any difficulty accessing the files or need any other assistance, please contact me at Kevin.Witt@nrc.gov.

Best regards,
Kevin

Kevin Witt
Project Manager
Japan Lessons Learned Project Directorate
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555
Office (301) 415-2145

John J. Sipos

From: John J. Sipos
Sent: Monday, October 28, 2013 1:59 PM
To: Witt, Kevin (Kevin.Witt@nrc.gov)
Subject: State of New York Request for MACCS2 Files

Good Afternoon Mr. Witt:

Thank you for your email of October 23, 2013. As you note, the State of New York has requested the input and output files for the MACCS runs that were performed by NRC and its contractors and consultants in connection with the Spent Fuel Consequence Study. Your email referenced documents posted to the public portion of the Agency-Wide Document Access Management System on the afternoon of October 23 within accession number ML13282A535. Those documents appear to be from MACCS runs performed a year ago back on November 13, 2012. Each of the files that were posted within ML13282A535 are in an Adobe Portable Document Format (PDF) format. It appears that these PDF files were converted to the PDF format from their native file format. The State wishes to review the input and output files in their native format. Native MACCS input and output files frequently have the following suffixes: ".inp" and ".out". The State requests that NRC make the MACCS input and output files available to the State in their native format. Thank you.

Best regards,

John

John Sipos
Assistant Attorney General
tel. 518-402-2251

John J. Sipos

From: Witt, Kevin <Kevin.Witt@nrc.gov>
Sent: Tuesday, November 12, 2013 5:09 PM
To: John J. Sipos
Subject: RE: State of New York Request for MACCS files

Hi John, I apologize for the delayed response, we are planning to send you the files on CD, as they are too large for sending over email. Can you please let me know the address to send the cds?

Thanks,
Kevin

From: John J. Sipos [mailto:John.Sipos@ag.ny.gov]
Sent: Thursday, November 07, 2013 8:10 AM
To: Witt, Kevin
Subject: State of New York Request for MACCS files

Good morning Mr. Witt:

Following up on previous emails, I am checking back with you concerning the State's request for the native file format of the input and output files for the MACCS runs undertaken in connection with the Spent Fuel Consequence Study. The State understands that some of the runs were conducted in 2012. Thank you.

Best regards,

John

John Sipos
Assistant Attorney General
State of New York
tel. 518-402-2251

John J. Sipos

From: John J. Sipos
Sent: Monday, November 25, 2013 4:51 PM
To: Witt, Kevin (Kevin.Witt@nrc.gov)
Subject: NYS Request for MACCS inputs - still not received

Hello Mr. Witt:

The afternoon mail run has been completed, and the State has still not received the MACCS inputs in native format.

Is there some way you can track the mail package that you sent out?

John

John Sipos
Assistant Attorney General
tel. 518-402-2251

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 18, 2013

NYS OFFICE OF THE ATTORNEY GENERAL
RECEIVED

NOV 26 2013

ENVIRONMENTAL PROTECTION BUREAU
ALBANY

John Sipos
Assistant Attorney General
Office of the Attorney General
The Capitol, State Street
Albany, NY 122224

Dear Mr. Sipos:

During the September 18, 2013 public meeting and in your September 29, 2013 email, you requested that the State of New York be provided the documentation/files reflecting the MACCS2 Accident Consequence Analysis Code runs performed in connection with the recently completed Spent Fuel Pool Study, and clarified on October 28th, 2013 that you wish to review the files in their native ".inp" and ".out" format. Attached to this letter is a CD containing the project folders of the base case MACCS2 calculations for the Spent Fuel Pool Study with the ".inp" and ".out" files These folders represent a site-specific analysis of a highly unlikely spent fuel pool accident at a Boiling Water Reactor (BWR) nuclear power plant with a Mark I containment. As with other BWR plants with Mark I containments, the spent fuel pool is located outside the containment structure and generally contains several times the source term found in an operating reactor. The MACCS2 Accident Consequence Analysis Code is a widely accepted accident consequence analysis code used to model various types of radiological accidents. If you have any difficulty accessing the files or need any other assistance, please feel free to contact me at (301) 415-2145 or by email at kevin.witt@nrc.gov.

Sincerely,

Kevin Witt

Kevin M. Witt, Project Manager
Policy and Support Branch
Japan Lessons-Learned Project Directorate
Office of Nuclear Reactor Regulation

John J. Sipos

From: John J. Sipos
Sent: Wednesday, November 27, 2013 7:59 AM
To: Witt, Kevin (Kevin.Witt@nrc.gov)
Subject: NYS request for MACCS files

Good morning Mr. Witt:

A disc arrived here in yesterday afternoon's mail run (Nov. 27). Thank you.

John Sipos
Assistant Attorney General
tel. 518-402-2251

Attachment 6

Package Description Accompanying Compact Disc of Native Format MACCS2 files;

Various MACCS2 Input and Output Files for Spent Fuel Pool Consequence Study
(dated November 13, 2012)

(with red highlights added as indicated)

In this zip file, there are the 5 project folders from the base case MACCS2 (rev 3.7.0) calculations for the Spent Fuel Pool Study, and an additional 2 project folders for a sensitivity for alternative source terms based on a uniform loading distribution for an applicable amount of time after reactor shutdown. These 7 project folders represent 7 source terms, each containing 3 different applications of dose response for a total of 21 calculations. The project folders are labeled based on their source term ID (e.g. 2.6u), then loading configuration. Please note that the "Data" folder, which is the same for all the calculations, has been extracted from the project folders and given once to save disc space.

MACCS2 11/13/2012 17:00:14 Version 3.7.0.0 : 11/9/12 170014.859
P1: ATMOS USER INPUT (UNIT 24) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Input\Atmos1.inp
P2: EARLY USER INPUT (UNIT 25) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Input\Early1.inp
P3: CHRONC USER INPUT (UNIT 26) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Input\Chronc1.inp
P4: METEOROLOGY DATA (UNIT 28) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Data\PB MACCS2 2006 Met Data 64WD.inp
P5: SITE DATA INPUT (UNIT 29) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Data\PBsite2011_64cp_26r rev1_64.inp
P6: LIST OUTPUT (UNIT 06) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Output\Model1.out

```
USER INPUT IS READ FROM UNIT 24
RECORD IDENTIFIER FIELDS  11 CHARACTERS LONG ARE EXPECTED.
THE FIRST 499 COLUMNS OF EACH INPUT RECORD ARE PROCESSED.

RECORD
NUMBER                RECORD

     * File created using WinMACCS version 3.7.0 11/13/2012 4:58:00 PM
     *
     * MACCS2 Cyclical File: C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Data\LNT.txt
     *
     * Peach Bottom Revision 7 for Spent Fuel Pool Scoping Study
     *
     * The initial WinMACCS file for the seismic runs was created May 12, 2009 using the Jan 21 2009 file for the PB STSBO.
     *
     * Identifies this MACCS calculation
   1  RIATNAM1001     'OCP3 low density no spray'
     *
     * NUMRAD, Number of Radial Spatial Elements
   2  GENUMRAD001     26
     *
     * SPAEND, Spatial Endpoint Distances (km)
   3  GESPAEND001     0.16
   4  GESPAEND002     0.52
   5  GESPAEND003     1.21
   6  GESPAEND004     1.61
   7  GESPAEND005     2.13
   8  GESPAEND006     3.22
   9  GESPAEND007     4.02
  10  GESPAEND008     4.83
  11  GESPAEND009     5.63
  12  GESPAEND010     8.05
  13  GESPAEND011     11.27
  14  GESPAEND012     16.09
  15  GESPAEND013     20.92
  16  GESPAEND014     25.75
  17  GESPAEND015     32.19
  18  GESPAEND016     40.23
  19  GESPAEND017     48.28
  20  GESPAEND018     64.37
  21  GESPAEND019     80.47
  22  GESPAEND020     112.65
  23  GESPAEND021     160.93
  24  GESPAEND022     241.14
  25  GESPAEND023     321.87
  26  GESPAEND024     563.27
  27  GESPAEND025     804.67
  28  GESPAEND026     1609.34
     *
     * Form 'Site File' Comment:
     * Updated to 2011 using Census Bureau data and CPI data.
     *
     * NUMCOR, Number of angular compass directions
  29  GENUMCOR001     64
     *
     * Form 'Radionuclides' Comment:
     * From ORIGEN, and updated to correct for isotope-by-isotope release fractions (which cannot be done in the chemical group release fractions).
     *
     * NUMISO, Number of Nuclides
  30  ISNUMISO001     69
     *
     * Form 'Chemical Names' Comment:
     * Group names are imported from MELMACCS.
     *
     * MAXGRP, Number of Element Groups
  31  ISMAXGRP001     9
     *
     * Form 'Wet/Dry Depos Flags' Comment:
     * No change
     *
     * WETDEP, DRYDEP, Wet and Dry Deposition Flags for Each Nuclide Group
  32  ISDEPFLA001     .FALSE. .FALSE.
  33  ISDEPFLA002     .TRUE.  .TRUE.
  34  ISDEPFLA003     .TRUE.  .TRUE.
  35  ISDEPFLA004     .TRUE.  .TRUE.
  36  ISDEPFLA005     .TRUE.  .TRUE.
  37  ISDEPFLA006     .TRUE.  .TRUE.
  38  ISDEPFLA007     .TRUE.  .TRUE.
  39  ISDEPFLA008     .TRUE.  .TRUE.
  40  ISDEPFLA009     .TRUE.  .TRUE.
     *
     * NUMSTB_ZERO = 0
  41  ISNUMSTB001     0
     *
     * Form 'Pseudostable Radionuclides' Comment:
     * Come in thru MELMACCS.
     *
     * NUMSTB, Number of Pseudostable Radionuclides
  42  ISNUMSTB001     16
********   RECORD NUMBER    42 REPLACES RECORD NUMBER    41   ********
     *
     * NAMSTB, List of Pseudostable Radionuclides
  43  ISNAMSTB001     I-129
  44  ISNAMSTB002     Xe-131m
  45  ISNAMSTB003     Xe-133m
  46  ISNAMSTB004     Cs-135
  47  ISNAMSTB005     Sm-147
  48  ISNAMSTB006     U-234
  49  ISNAMSTB007     U-235
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```
50  ISNAMSTB008      U-236
51  ISNAMSTB009      U-237
52  ISNAMSTB010      Np-237
53  ISNAMSTB011      Rb-87
54  ISNAMSTB012      Zr-93
55  ISNAMSTB013      Nb-93m
56  ISNAMSTB014      Nb-95m
57  ISNAMSTB015      Tc-99
58  ISNAMSTB016      Pm-147
    *
    * NUCNAM, IGROUP, Chemical group associated with each nuclide
59  ISOTPGRP001      Kr-85   1
60  ISOTPGRP002      Kr-85m  1
61  ISOTPGRP003      Kr-87   1
62  ISOTPGRP004      Kr-88   1
63  ISOTPGRP005      Xe-133  1
64  ISOTPGRP006      Xe-135  1
65  ISOTPGRP007      Xe-135m 1
66  ISOTPGRP008      Cs-134  2
67  ISOTPGRP009      Cs-136  2
68  ISOTPGRP010      Cs-137  2
69  ISOTPGRP011      Rb-86   2
70  ISOTPGRP012      Rb-88   2
71  ISOTPGRP013      Ba-139  3
72  ISOTPGRP014      Ba-140  3
73  ISOTPGRP015      Sr-89   3
74  ISOTPGRP016      Sr-90   3
75  ISOTPGRP017      Sr-91   3
76  ISOTPGRP018      Sr-92   3
77  ISOTPGRP019      Ba-137m 3
78  ISOTPGRP020      I-131   4
79  ISOTPGRP021      I-132   4
80  ISOTPGRP022      I-133   4
81  ISOTPGRP023      I-134   4
82  ISOTPGRP024      I-135   4
83  ISOTPGRP025      Te-127  5
84  ISOTPGRP026      Te-127m 5
85  ISOTPGRP027      Te-129  5
86  ISOTPGRP028      Te-129m 5
87  ISOTPGRP029      Te-131m 5
88  ISOTPGRP030      Te-132  5
89  ISOTPGRP031      Te-131  5
90  ISOTPGRP032      Rh-105  6
91  ISOTPGRP033 Ru-103  6
92  ISOTPGRP034      Ru-105  6
93  ISOTPGRP035      Ru-106  6
94  ISOTPGRP036      Rh-103m 6
95  ISOTPGRP037     Rh-106  6
96  ISOTPGRP038      Nb-95   7
97  ISOTPGRP039      Co-58   7
98  ISOTPGRP040      Co-60   7
99  ISOTPGRP041      Mo-99   7
100  ISOTPGRP042      Tc-99m  7
101  ISOTPGRP043      Nb-97   7
102  ISOTPGRP044      Nb-97m  7
103  ISOTPGRP045      Ce-141  8
104  ISOTPGRP046      Ce-143  8
105  ISOTPGRP047      Ce-144  8
106  ISOTPGRP048      Np-239  8
107  ISOTPGRP049      Pu-238  8
108  ISOTPGRP050      Pu-239  8
109  ISOTPGRP051      Pu-240  8
110  ISOTPGRP052      Pu-241  8
111  ISOTPGRP053      Zr-95   8
112  ISOTPGRP054      Zr-97   8
113  ISOTPGRP055      Am-241  9
114  ISOTPGRP056      Cm-242  9
115  ISOTPGRP057      Cm-244  9
116  ISOTPGRP058      La-140  9
117  ISOTPGRP059      La-141  9
118  ISOTPGRP060      La-142  9
119  ISOTPGRP061      Nd-147  9
120  ISOTPGRP062      Pr-143  9
121  ISOTPGRP063      Y-90    9
122  ISOTPGRP064      Y-91    9
123  ISOTPGRP065      Y-92    9
124  ISOTPGRP066      Y-93    9
125  ISOTPGRP067      Y-91m   9
126  ISOTPGRP068      Pr-144  9
127  ISOTPGRP069      Pr-144m 9
    *
    * Form 'Wet Deposition' Comment:
    * Values from Nate et al's report, table 7, page 64 (April 2007).  Derived  assuming 1 micrometer particles.  Do not change.
    *
    * CWASH1, Washout Coefficient Number One, Linear Factor
128  WDCWASH1001      1.89E-05
    *
    * CWASH2, Washout Coefficient Number Two, Exponential Factor
129  WDCWASH2001      .664
    *
    * Form 'Dry Deposition' Comment:
    * Value Given by Nate.  MELMACCS cannot currently calculate a DDV based on a surface roughness greater than 20 cm
    *
    * NPSGRP, Number of Particle Size Groups
130  DDNPSGRP001      10
    *
    * VDEPOS, Dry Deposition Velocities for Each Particle Size Group (m/sec)
131  DDVDEPOS001      0.0011
132  DDVDEPOS002      0.001
133  DDVDEPOS003      0.0014
134  DDVDEPOS004      0.0023
135  DDVDEPOS005      0.0045
136  DDVDEPOS006      0.0092
137  DDVDEPOS007      0.0177
138  DDVDEPOS008      0.0291
139  DDVDEPOS009      0.0367
140  DDVDEPOS010      0.0367
    *
    * Form 'Dispersion Function' Comment:
```

```
    * From Nate's draft report (April 2007).
    *
    * CYSIGA, Dispersion function parameter
141  DPCYSIGA001      .7507
142  DPCYSIGA002      .7507
143  DPCYSIGA003      .4063
144  DPCYSIGA004      .2779
145  DPCYSIGA005      .2158
146  DPCYSIGA006      .2158
    *
    * CYSIGB, Dispersion function parameter
147  DPCYSIGB001      .866
148  DPCYSIGB002      .866
149  DPCYSIGB003      .865
150  DPCYSIGB004      .881
151  DPCYSIGB005      .866
152  DPCYSIGB006      .866
    *
    * CZSIGA, Dispersion function parameter
153  DPCZSIGA001      .0361
154  DPCZSIGA002      .0361
155  DPCZSIGA003      .2036
156  DPCZSIGA004      .2636
157  DPCZSIGA005      .2463
158  DPCZSIGA006      .2463
    *
    * CZSIGB, Dispersion function parameter
159  DPCZSIGB001      1.277
160  DPCZSIGB002      1.277
161  DPCZSIGB003      .859
162  DPCZSIGB004      .751
163  DPCZSIGB005      .619
164  DPCZSIGB006      .619
    *
    * Form 'Scaling Factors' Comment:
    * ZSCALE correspond to a surface roughness of 60 cm.  The formula for calculating it is in the NUREG/CR-4691.
    *
    * YSCALE, linear scaling factor for sigma-y
165  DPYSCALE001      1.
    *
    * ZSCALE, linear scaling factor for sigma-z
166  DPZSCALE001      1.82
    *
    * DISPMD - dispersion long-range model
167  DPDISPMD001      LRDIST
    *
    * MNDMOD, plume meander model
168  PMMNDMOD001      NEW
    *
    * WINSP1, wind speed where the meander factor changes from constant to decreasing
169  PMWINSP1001      2.
    *
    * WINSP2, wind speed where the meander factor reaches one
170  PMWINSP2001      6.
    *
    * MNDIST, distance, for use in 1.145
171  PMMNDIST001      800.
    *
    * MNDFAC, plume meander stability class factors, for use in 1.145
172  PMMNDFAC001      1.
173  PMMNDFAC002      1.
174  PMMNDFAC003      1.
175  PMMNDFAC004      2.
176  PMMNDFAC005      3.
177  PMMNDFAC006      4.
    *
    * Form 'Plume Rise Scale Factor' Comment:
    * Using standard modeling options.
    *
    * SCLCRW, scaling factor for entrainment of buoyant plume
178  PRSCLCRW001      1.
    *
    * SCLADP, scaling factor for the a–d stability plume rise formula
179  PRSCLADP001      1.
    *
    * SCLEFP, scaling factor for the e–f stability plume rise formula
180  PRSCLEFP001      1.
    *
    * Form 'Wake Effect Data' Comment:
    * Data for Peach Bottom from NUREG-1150.
    *
    * BUILDH, building height (meters)
181  WEBUILDH001      50.
182  WEBUILDH002      50.
183  WEBUILDH003      50.
184  WEBUILDH004      50.
185  WEBUILDH005      50.
186  WEBUILDH006      50.
187  WEBUILDH007      50.
188  WEBUILDH008      50.
189  WEBUILDH009      50.
190  WEBUILDH010      50.
191  WEBUILDH011      50.
192  WEBUILDH012      50.
193  WEBUILDH013      50.
194  WEBUILDH014      50.
195  WEBUILDH015      50.
196  WEBUILDH016      50.
197  WEBUILDH017      50.
198  WEBUILDH018      50.
199  WEBUILDH019      50.
200  WEBUILDH020      50.
201  WEBUILDH021      50.
202  WEBUILDH022      50.
203  WEBUILDH023      50.
204  WEBUILDH024      50.
205  WEBUILDH025      50.
206  WEBUILDH026       50.
207  WEBUILDH027      50.
```

208 WEBUILDH028 50.
209 WEBUILDH029 50.
210 WEBUILDH030 50.
211 WEBUILDH031 50.
*
* SIGYINIT, initial value of sigma-y for each of the plumes (meters)
212 SIGYINIT001 11.6
213 SIGYINIT002 11.6
214 SIGYINIT003 11.6
215 SIGYINIT004 11.6
216 SIGYINIT005 11.6
217 SIGYINIT006 11.6
218 SIGYINIT007 11.6
219 SIGYINIT008 11.6
220 SIGYINIT009 11.6
221 SIGYINIT010 11.6
222 SIGYINIT011 11.6
223 SIGYINIT012 11.6
224 SIGYINIT013 11.6
225 SIGYINIT014 11.6
226 SIGYINIT015 11.6
227 SIGYINIT016 11.6
228 SIGYINIT017 11.6
229 SIGYINIT018 11.6
230 SIGYINIT019 11.6
231 SIGYINIT020 11.6
232 SIGYINIT021 11.6
233 SIGYINIT022 11.6
234 SIGYINIT023 11.6
235 SIGYINIT024 11.6
236 SIGYINIT025 11.6
237 SIGYINIT026 11.6
238 SIGYINIT027 11.6
239 SIGYINIT028 11.6
240 SIGYINIT029 11.6
241 SIGYINIT030 11.6
242 SIGYINIT031 11.6
*
* SIGZINIT, initial value of sigma-z for each of the plumes (meters)
243 SIGZINIT001 23.3
244 SIGZINIT002 23.3
245 SIGZINIT003 23.3
246 SIGZINIT004 23.3
247 SIGZINIT005 23.3
248 SIGZINIT006 23.3
249 SIGZINIT007 23.3
250 SIGZINIT008 23.3
251 SIGZINIT009 23.3
252 SIGZINIT010 23.3
253 SIGZINIT011 23.3
254 SIGZINIT012 23.3
255 SIGZINIT013 23.3
256 SIGZINIT014 23.3
257 SIGZINIT015 23.3
258 SIGZINIT016 23.3
259 SIGZINIT017 23.3
260 SIGZINIT018 23.3
261 SIGZINIT019 23.3
262 SIGZINIT020 23.3
263 SIGZINIT021 23.3
264 SIGZINIT022 23.3
265 SIGZINIT023 23.3
266 SIGZINIT024 23.3
267 SIGZINIT025 23.3
268 SIGZINIT026 23.3
269 SIGZINIT027 23.3
270 SIGZINIT028 23.3
271 SIGZINIT029 23.3
272 SIGZINIT030 23.3
273 SIGZINIT031 23.3
*
* ATNAM2, specific descriptive text describing this particular source term
274 RDATNAM2001 'OCP3 low density no spray'
*
* OALARM, time after accident initiation that off-site alarm is initiated (sec)
275 RDOALARM001 3600.
*
* Form 'Plume Parameters' Comment:
* These values come from MELCOR PTF file. Plume discretization is done by user.
*
* NUMREL, number of plumes
276 RDNUMREL001 31
*
* MAXRIS, selection of risk-dominant plume segment
277 RDMAXRIS001 2
*
* REFTIM, representative time point for dispersion and radioactive decay
278 RDREFTIM001 0.
279 RDREFTIM002 0.5
280 RDREFTIM003 0.5
281 RDREFTIM004 0.5
282 RDREFTIM005 0.5
283 RDREFTIM006 0.5
284 RDREFTIM007 0.5
285 RDREFTIM008 0.5
286 RDREFTIM009 0.5
287 RDREFTIM010 0.5
288 RDREFTIM011 0.5
289 RDREFTIM012 0.5
290 RDREFTIM013 0.5
291 RDREFTIM014 0.5
292 RDREFTIM015 0.5
293 RDREFTIM016 0.5
294 RDREFTIM017 0.5
295 RDREFTIM018 0.5
296 RDREFTIM019 0.5
297 RDREFTIM020 0.5
298 RDREFTIM021 0.5
299 RDREFTIM022 0.5

300 RDREFTIM023 0.5
301 RDREFTIM024 0.5
302 RDREFTIM025 0.5
303 RDREFTIM026 0.5
304 RDREFTIM027 0.5
305 RDREFTIM028 0.5
306 RDREFTIM029 0.5
307 RDREFTIM030 0.5
308 RDREFTIM031 0.5
*
* PLM_DEN, plume rise model density
309 RDPLMMOD001 DENSITY
*
* Form 'Density and Flow' Comment:
* Come in thru MELMACCS.
*
* PLMFLO, Heat by Density
310 RDPLMFLA001 0.13828
311 RDPLMFLA002 1.0663
312 RDPLMFLA003 1.7425
313 RDPLMFLA004 1.4175
314 RDPLMFLA005 1.3887
315 RDPLMFLA006 1.3389
316 RDPLMFLA007 1.332
317 RDPLMFLA008 1.3021
318 RDPLMFLA009 1.2124
319 RDPLMFLA010 1.2111
320 RDPLMFLA011 1.2104
321 RDPLMFLA012 1.2025
322 RDPLMFLA013 1.2016
323 RDPLMFLA014 1.1983
324 RDPLMFLA015 1.1943
325 RDPLMFLA016 1.1948
326 RDPLMFLA017 1.1969
327 RDPLMFLA018 1.1968
328 RDPLMFLA019 1.197
329 RDPLMFLA020 1.1974
330 RDPLMFLA021 1.2029
331 RDPLMFLA022 1.2041
332 RDPLMFLA023 1.2017
333 RDPLMFLA024 1.2527
334 RDPLMFLA025 1.2524
335 RDPLMFLA026 1.2042
336 RDPLMFLA027 1.2056
337 RDPLMFLA028 1.2055
338 RDPLMFLA029 1.2063
339 RDPLMFLA030 1.2074
340 RDPLMFLA031 1.2085
*
* PLMDEN, Heat by Density
341 RDPLMDEN001 1.0601
342 RDPLMDEN002 0.89765
343 RDPLMDEN003 0.72937
344 RDPLMDEN004 0.72574
345 RDPLMDEN005 0.73513
346 RDPLMDEN006 0.74423
347 RDPLMDEN007 0.75136
348 RDPLMDEN008 0.75816
349 RDPLMDEN009 0.76735
350 RDPLMDEN010 0.77797
351 RDPLMDEN011 0.78769
352 RDPLMDEN012 0.7957
353 RDPLMDEN013 0.80169
354 RDPLMDEN014 0.80682
355 RDPLMDEN015 0.81097
356 RDPLMDEN016 0.81426
357 RDPLMDEN017 0.81682
358 RDPLMDEN018 0.81878
359 RDPLMDEN019 0.82025
360 RDPLMDEN020 0.82132
361 RDPLMDEN021 0.82206
362 RDPLMDEN022 0.82254
363 RDPLMDEN023 0.82282
364 RDPLMDEN024 0.82293
365 RDPLMDEN025 0.82291
366 RDPLMDEN026 0.82278
367 RDPLMDEN027 0.82255
368 RDPLMDEN028 0.82225
369 RDPLMDEN029 0.8219
370 RDPLMDEN030 0.82151
371 RDPLMDEN031 0.82109
*
* BRGSMD, Briggs plume rise model
372 RDBRGSMD001 IMPROVED
*
* PLHITE, height of each plume segment at release (meters)
373 RDPLHITE001 50.
374 RDPLHITE002 50.
375 RDPLHITE003 50.
376 RDPLHITE004 50.
377 RDPLHITE005 50.
378 RDPLHITE006 50.
379 RDPLHITE007 50.
380 RDPLHITE008 50.
381 RDPLHITE009 50.
382 RDPLHITE010 50.
383 RDPLHITE011 50.
384 RDPLHITE012 50.
385 RDPLHITE013 50.
386 RDPLHITE014 50.
387 RDPLHITE015 50.
388 RDPLHITE016 50.
389 RDPLHITE017 50.
390 RDPLHITE018 50.
391 RDPLHITE019 50.
392 RDPLHITE020 50.
393 RDPLHITE021 50.
394 RDPLHITE022 50.
395 RDPLHITE023 50.

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396  RDPLHITE024      50.
397  RDPLHITE025      50.
398  RDPLHITE026      50.
399  RDPLHITE027      50.
400  RDPLHITE028      50.
401  RDPLHITE029      50.
402  RDPLHITE030      50.
403  RDPLHITE031      50.
     *
     * PLUDUR, duration of each plume segment (sec)
404  RDPLUDUR001      1439.
405  RDPLUDUR002      3600.
406  RDPLUDUR003      3600.
407  RDPLUDUR004      3600.
408  RDPLUDUR005      3600.
409  RDPLUDUR006      3600.
410  RDPLUDUR007      3600.
411  RDPLUDUR008      3600.
412  RDPLUDUR009      3600.
413  RDPLUDUR010      3600.
414  RDPLUDUR011      3600.
415  RDPLUDUR012      3600.
416  RDPLUDUR013      3600.
417  RDPLUDUR014      3600.
418  RDPLUDUR015      3600.
419  RDPLUDUR016      3600.
420  RDPLUDUR017      3600.
421  RDPLUDUR018      3600.
422  RDPLUDUR019      3600.
423  RDPLUDUR020      3600.
424  RDPLUDUR021      3600.
425  RDPLUDUR022      3600.
426  RDPLUDUR023      3600.
427  RDPLUDUR024      3600.
428  RDPLUDUR025      3600.
429  RDPLUDUR026      3600.
430  RDPLUDUR027      3600.
431  RDPLUDUR028      3600.
432  RDPLUDUR029      3600.
433  RDPLUDUR030      3600.
434  RDPLUDUR031      3600.
     *
     * PDELAY, time of release for each plume from xxxx (sec)
435  RDPDELAY001      1.49761E+05
436  RDPDELAY002      1.51200E+05
437  RDPDELAY003      1.54800E+05
438  RDPDELAY004      1.58400E+05
439  RDPDELAY005      1.62000E+05
440  RDPDELAY006      1.65600E+05
441  RDPDELAY007      1.69200E+05
442  RDPDELAY008      1.72800E+05
443  RDPDELAY009      1.76400E+05
444  RDPDELAY010      1.80000E+05
445  RDPDELAY011      1.83600E+05
446  RDPDELAY012      1.87200E+05
447  RDPDELAY013      1.90800E+05
448  RDPDELAY014      1.94400E+05
449  RDPDELAY015      1.98000E+05
450  RDPDELAY016      2.01600E+05
451  RDPDELAY017      2.05200E+05
452  RDPDELAY018      2.08800E+05
453  RDPDELAY019      2.12400E+05
454  RDPDELAY020      2.16000E+05
455  RDPDELAY021      2.19600E+05
456  RDPDELAY022      2.23200E+05
457  RDPDELAY023      2.26800E+05
458  RDPDELAY024      2.30400E+05
459  RDPDELAY025      2.34000E+05
460  RDPDELAY026      2.37600E+05
461  RDPDELAY027      2.41200E+05
462  RDPDELAY028      2.44800E+05
463  RDPDELAY029      2.48400E+05
464  RDPDELAY030      2.52000E+05
465  RDPDELAY031      2.55600E+05
     *
     * Form 'Particle Size Distribution' Comment:
     * Particle size distribution from MELMACCS.
     *
     * PSDIST, particle size distribution of each element group
466  RDPSDIST001    1.E-01  1.E-01  1.E-01  1.E-01  1.E-01  1.E-01  1.E-01  1.E-01  1.E-01  1.E-01
467  RDPSDIST002    4.322E-02     1.2312E-01    1.6798E-01    7.6416E-02    1.5644E-01    2.6678E-01    1.2244E-01    1.4178E-02    6.9871E-04    2.8727E-02
468  RDPSDIST003    4.5577E-02    9.8415E-02    1.1364E-01    6.6482E-02    1.7592E-01    3.0033E-01    1.4245E-01    1.7462E-02    8.9204E-04    3.8834E-02
469  RDPSDIST004    4.3872E-02    1.0599E-01    1.2062E-01    6.5441E-02    1.7898E-01    3.0619E-01    1.4506E-01    1.7497E-02    7.9859E-04    1.5549E-02
470  RDPSDIST005    4.3866E-02    1.1383E-01    1.4488E-01    7.1586E-02    1.663E-01     2.8387E-01    1.3234E-01    1.5666E-02    7.5146E-04    2.691E-02
471  RDPSDIST006    3.5221E-02    2.1349E-01    3.7254E-01    1.1522E-01    7.9686E-02    1.3419E-01    4.3894E-02    1.5061E-03    7.2501E-06    4.2505E-03
472  RDPSDIST007    3.5226E-02    2.1345E-01    3.7237E-01    1.1517E-01    7.9751E-02    1.343E-01     4.3958E-02    1.5144E-03    7.4842E-06    4.2565E-03
473  RDPSDIST008    3.5221E-02    2.1349E-01    3.7254E-01    1.1522E-01    7.9686E-02    1.3419E-01    4.3894E-02    1.5061E-03    7.2501E-06    4.2505E-03
474  RDPSDIST009    3.5221E-02    2.1349E-01    3.7254E-01    1.1522E-01    7.9686E-02    1.3419E-01    4.3894E-02    1.5061E-03    7.2501E-06    4.2505E-03
     *
     * CORINV, inventory of each radionuclide present in the facility at the time of accident initiation (becquerels)
475  RDCORINV001      Kr-85   4.58E+16
476  RDCORINV002      Kr-85m  0.
477  RDCORINV003      Kr-87   0.
478  RDCORINV004      Kr-88   0.
479  RDCORINV005      Xe-133  4.2E+16
480  RDCORINV006      Xe-135  7.02E-11
481  RDCORINV007      Xe-135m 0.
482  RDCORINV008      Cs-134  8.52E+17
483  RDCORINV009      Cs-136  3.55E+16
484  RDCORINV010      Cs-137  6.33E+17
485  RDCORINV011      Rb-86   1.98E+15
486  RDCORINV012      Rb-88   0.
487  RDCORINV013      Ba-139  0.
488  RDCORINV014      Ba-140  4.56E+17
489  RDCORINV015      Sr-89   9.48E+17
490  RDCORINV016      Sr-90   3.38E+17
491  RDCORINV017      Sr-91   1.72E-10
492  RDCORINV018      Sr-92   0.
493  RDCORINV019      Ba-137m 4.54E+17
```

```
494 RDCORINV020 I-131 5.28E+16
495 RDCORINV021 I-132 6.81E+14
496 RDCORINV022 I-133 3.54000E+05
497 RDCORINV023 I-134 0.
498 RDCORINV024 I-135 0.
499 RDCORINV025 Te-127 1.67E+16
500 RDCORINV026 Te-127m 1.69E+16
501 RDCORINV027 Te-129 2.02E+16
502 RDCORINV028 Te-129m 3.15E+16
503 RDCORINV029 Te-131m 3.07E+08
504 RDCORINV030 Te-132 6.62E+14
505 RDCORINV031 Te-131 6.91E+07
506 RDCORINV032 Rh-105 5.29E+10
507 RDCORINV033 Ru-103 1.26E+18
508 RDCORINV034 Ru-105 0.
509 RDCORINV035 Ru-106 1.05E+18
510 RDCORINV036 Rh-103m 1.25E+18
511 RDCORINV037 Rh-106 1.05E+18
512 RDCORINV038 Nb-95 1.72E+18
513 RDCORINV039 Co-58 2.05E+14
514 RDCORINV040 Co-60 8.82E+14
515 RDCORINV041 Mo-99 1.97E+14
516 RDCORINV042 Tc-99m 1.91E+14
517 RDCORINV043 Nb-97 328.
518 RDCORINV044 Nb-97m 288.
519 RDCORINV045 Ce-141 1.06E+18
520 RDCORINV046 Ce-143 1.67E+10
521 RDCORINV047 Ce-144 1.77E+18
522 RDCORINV048 Np-239 6.88E+14
523 RDCORINV049 Pu-238 1.11E+16
524 RDCORINV050 Pu-239 6.88E+14
525 RDCORINV051 Pu-240 1.44E+15
526 RDCORINV052 Pu-241 3.1E+17
527 RDCORINV053 Zr-95 1.52E+18
528 RDCORINV054 Zr-97 362.
529 RDCORINV055 Am-241 5.69E+14
530 RDCORINV056 Cm-242 1.37E+17
531 RDCORINV057 Cm-244 1.56E+16
532 RDCORINV058 La-140 3.83E+17
533 RDCORINV059 La-141 0.
534 RDCORINV060 La-142 0.
535 RDCORINV061 Nd-147 9.22E+16
536 RDCORINV062 Pr-143 3.53E+17
537 RDCORINV063 Y-90 2.46E+17
538 RDCORINV064 Y-91 1.01E+18
539 RDCORINV065 Y-92 0.
540 RDCORINV066 Y-93 6.7E-09
541 RDCORINV067 Y-91m 7.93E-11
542 RDCORINV068 Pr-144 1.79E+18
543 RDCORINV069 Pr-144m 2.53E+16
    *
    * Form 'Inventory Scale Factor' Comment:
    * Set by MELMACCS.
    *
    * CORSCA, scaling factor to adjust the core inventory
544 RDCORSCA001 1.0
    *
    * APLFRC, Specifies how release fractions are applied to daughter ingrowth products
545 RDAPLFRC001 PARENT
    *
    * GRPNAM, user assigned name of each chemical group.  May have been imported from MelMACCS
    *ISGRPNAM001 Xe
    *ISGRPNAM002 Cs
    *ISGRPNAM003 Ba
    *ISGRPNAM004 I
    *ISGRPNAM005 Te
    *ISGRPNAM006 Ru
    *ISGRPNAM007 Mo
    *ISGRPNAM008 Ce
    *ISGRPNAM009 La
    *
    * Form 'Release Fractions' Comment:
    * These values come from MELCOR PTF file.  Plume discretization is done by user.  MACCS2 Radionuclide Inventory will account for the correct release magnitude on a isotope-by-isotope basis .
    *
    * RELFRC, release fractions for each of the plume segments for each chemical group
546 RDRELFRC001 7.1633E-05 3.5753E-04 9.4851E-05 6.9529E-04 7.6566E-04 1.8367E-09 2.004E-07 8.323E-14 8.2182E-14
547 RDRELFRC002 0.0033996 0.0022147 5.7966E-04 0.0050827 0.0048564 1.6672E-08 1.8434E-06 3.045E-13 3.0067E-13
548 RDRELFRC003 0.0011582 0.0013222 3.496E-04 0.0024428 0.002813 5.2816E-08 5.8063E-06 6.7283E-12 6.6436E-12
549 RDRELFRC004 7.6393E-04 0.0010165 2.6679E-04 0.0018128 0.0021472 9.0953E-08 9.992E-06 3.5351E-12 3.4906E-12
550 RDRELFRC005 5.951E-04 6.8541E-04 1.791E-04 0.001212 0.0014444 7.8752E-08 8.6489E-06 2.6606E-12 2.6271E-12
551 RDRELFRC006 4.9234E-04 2.6977E-04 6.697E-05 4.844E-04 5.618E-04 9.791E-08 1.0765E-05 4.5906E-12 4.5329E-12
552 RDRELFRC007 4.0424E-04 2.643E-05 1.92E-06 5.85E-05 4.65E-05 9.7396E-08 1.0706E-05 2.6187E-12 2.5857E-12
553 RDRELFRC008 3.3065E-04 2.109E-05 1.24E-06 4.27E-05 3.58E-05 8.4039E-08 9.2318E-06 1.0348E-12 1.0218E-12
554 RDRELFRC009 2.5794E-04 1.38E-05 8.E-07 2.71E-05 2.33E-05 5.5634E-08 6.1108E-06 3.E-15 3.E-15
555 RDRELFRC010 2.0927E-04 9.27E-06 5.3E-07 1.8E-05 1.57E-05 3.7424E-08 4.1106E-06 1.1E-15 1.1E-15
556 RDRELFRC011 1.6702E-04 1.744E-05 3.5E-06 3.07E-05 3.45E-05 2.2393E-08 2.4596E-06 5.E-16 4.E-16
557 RDRELFRC012 1.3323E-04 7.68E-06 2.01E-06 1.31E-05 1.61E-05 9.97E-10 1.095E-07 3.E-16 3.E-16
558 RDRELFRC013 1.1116E-04 3.62E-06 9.4E-07 6.E-06 7.6E-06 4.72E-10 5.19E-08 3.E-16 3.E-16
559 RDRELFRC014 9.328E-05 2.55E-06 6.7E-07 4.2E-06 5.4E-06 3.38E-10 3.7E-08 3.E-16 3.E-16
560 RDRELFRC015 7.852E-05 1.82E-06 4.8E-07 2.9E-06 3.8E-06 2.42E-10 2.66E-08 3.E-16 3.E-16
561 RDRELFRC016 6.623E-05 1.29E-06 3.4E-07 2.1E-06 2.7E-06 1.75E-10 1.92E-08 3.E-16 3.E-16
562 RDRELFRC017 5.592E-05 9.2E-07 2.4E-07 1.4E-06 1.9E-06 1.27E-10 1.4E-08 2.E-16 2.E-16
563 RDRELFRC018 4.727E-05 6.5E-07 1.7E-07 1.E-06 1.4E-06 9.3E-11 1.02E-08 3.E-16 3.E-16
564 RDRELFRC019 3.998E-05 4.7E-07 1.2E-07 7.E-07 1.E-06 6.9E-11 7.5E-09 3.E-16 3.E-16
565 RDRELFRC020 3.383E-05 3.3E-07 9.E-08 5.E-07 7.E-07 5.1E-11 5.7E-09 3.E-16 3.E-16
566 RDRELFRC021 2.865E-05 2.3E-07 6.E-08 3.E-07 5.E-07 4.E-11 4.3E-09 3.E-16 3.E-16
567 RDRELFRC022 2.425E-05 1.7E-07 4.E-08 3.E-07 3.E-07 3.E-11 3.4E-09 3.E-16 3.E-16
568 RDRELFRC023 2.055E-05 1.2E-07 3.E-08 2.E-07 3.E-07 2.4E-11 2.7E-09 3.E-16 3.E-16
569 RDRELFRC024 1.739E-05 8.E-08 2.E-08 1.E-07 1.E-07 2.E-11 2.1E-09 3.E-16 2.E-16
570 RDRELFRC025 1.474E-05 6.E-08 2.E-08 1.E-07 2.E-07 1.7E-11 1.9E-09 3.E-16 3.E-16
571 RDRELFRC026 1.248E-05 5.E-08 1.E-08 0. 0. 1.5E-11 1.6E-09 3.E-16 3.E-16
572 RDRELFRC027 1.057E-05 3.E-08 1.E-08 1.E-07 1.E-07 1.3E-11 1.4E-09 3.E-16 3.E-16
573 RDRELFRC028 8.95E-06 2.E-08 0. 0. 1.E-07 1.2E-11 1.3E-09 3.E-16 3.E-16
574 RDRELFRC029 7.58E-06 2.E-08 1.E-08 0. 0. 1.1E-11 1.3E-09 3.E-16 3.E-16
575 RDRELFRC030 6.41E-06 1.E-08 0. 0. 0. 1.E-11 1.1E-09 3.E-16 4.E-16
576 RDRELFRC031 5.43E-06 1.E-08 0. 0. 0. 1.1E-11 1.2E-09 3.E-16 3.E-16
    *
    * ENDAT1, flag indicating whether only atmos is run
577 OCENDAT1001 .FALSE.
```

```
  *
  * IDEBUG, specifies set of debug results to report
578  OCIDEBUG001     0
  *
  * NUCOUT, name of the nuclide to be listed on the dispersion listings
579  OCNUCOUT001     Cs-137
  *
  * METCOD, meteorological sampling option code
580  M1METCOD001     2
  *
  * Form 'Boundary Limit' Comment:
  * From NUREG-1150.
  *
  * LIMSPA, last spatial interval for measured weather
581  M2LIMSPA001     25
  *
  * Form 'Constant or Boundary Conditions' Comment:
  * Stability class 5 is the most prevalent in the PB data.  2.2 is average speed data, and other values are from NUREG-1150 data.
  *
  * BNDMXH, boundary weather mixing layer height (meters)
582  M2BNDMXH001     1000.
  *
  * IBDSTB, boundary weather stability class index
583  M2IBDSTB001     4
  *
  * BNDRAN, boundary weather rain rate (mm/hr)
584  M2BNDRAN001     5.
  *
  * BNDWND, boundary weather wind speed (m/sec)
585  M2BNDWND001     2.2
  *
  * MAXHGT, if equal DAY_AND_NIGHT, then time of sunrise/sunset is used to calculate max mixing height. DAY_ONLY uses MACCS2 1.12 model
586  M1MAXHGT001     DAY_AND_NIGHT
  *
  * Form 'Site Location' Comment:
  * Consistent with PB site file.
  *
  * LATITUDE_DEG, LATITUDE_MIN, LATITUDE_SEC, indicates latitude of site, used with longitude
587  M1LATITU001     39.
  *
  * LATITU_MIN minutes portion of latitude
588  M1LATITU002     45.
  *
  * LATITU_SEC, seconds portion of latitude
589  M1LATITU003     32.
  *
  * LONGIT_DEG, LONGIT_MIN, LONGIT_SEC, indicates longitude of site, used with latitude
590  M1LONGIT001     76.
  *
  * LONGIT_MIN, minutes portion of longitude
591  M1LONGIT002     16.
  *
  * LONGIT_SEC, seconds portion of longitude
592  M1LONGIT003     9.
  *
  * Form 'Rain Distances' Comment:
  * From NUREG-1150.
  *
  * NRNINT, number of rain distance intervals for binning
593  M4NRNINT001     5
  *
  * RNDSTS, endpoints of the rain distance intervals (km)
594  M4RNDSTS001     3.22
595  M4RNDSTS002     5.63
596  M4RNDSTS003     11.27
597  M4RNDSTS004     20.92
598  M4RNDSTS005     32.19
  *
  * Form 'Rain Intensities' Comment:
  * From NUREG-1150.
  *
  * NRINTN, number of rain intensity breakpoints
599  M4NRINTN001     3
  *
  * RNRATE, rain intensity breakpoints for weather binning (mm/hr)
600  M4RNRATE001     2.
601  M4RNRATE002     4.
602  M4RNRATE003     6.
  *
  * IRSEED, initial seed for random number generator
603  M4IRSEED001     79
  *
  * Form 'Bins' Comment:
  * Minimum of 12 or 10% of samples in bin.
  *
  * NSBINS, number of bins to be sampled when NSMPLS = 0
604  M4NSBINS001     36
  *
  * INDXBN, INWGHT, number of weather sequences to be selected from specific weather bins
605  M4SMPLDF001     1     71
606  M4SMPLDF002     2     42
607  M4SMPLDF003     3     12
608  M4SMPLDF004     4     52
609  M4SMPLDF005     5     57
610  M4SMPLDF006     6     74
611  M4SMPLDF007     7     21
612  M4SMPLDF008     8     12
613  M4SMPLDF009     9     49
614  M4SMPLDF010     10     103
615  M4SMPLDF011     11     77
616  M4SMPLDF012     12     35
617  M4SMPLDF013     13     51
618  M4SMPLDF014     14     75
619  M4SMPLDF015     15     14
620  M4SMPLDF016     16     4
621  M4SMPLDF017     17     44
622  M4SMPLDF018     18     12
623  M4SMPLDF019     19     17
624  M4SMPLDF020     20     24
```

625 M4SMPLDF021 21 24
626 M4SMPLDF022 22 12
627 M4SMPLDF023 23 4
628 M4SMPLDF024 24 8
629 M4SMPLDF025 25 12
630 M4SMPLDF026 26 12
631 M4SMPLDF027 27 12
632 M4SMPLDF028 28 1
633 M4SMPLDF029 29 3
634 M4SMPLDF030 30 5
635 M4SMPLDF031 31 4
636 M4SMPLDF032 32 12
637 M4SMPLDF033 33 1
638 M4SMPLDF034 34 7
639 M4SMPLDF035 35 9
640 M4SMPLDF036 36 12
*
* ATMOS_ZERO = 0
641 TYPE0NUMBER 0
*
* NUM0, number of results
642 TYPE0NUMBER 14
******** RECORD NUMBER 642 REPLACES RECORD NUMBER 641 ********
*
* INDREL, INDRAD, CCDF, ATMOS release and spatial interval
643 TYPE0OUT001 1 1 NONE
644 TYPE0OUT002 1 2 NONE
645 TYPE0OUT003 1 3 NONE
646 TYPE0OUT004 1 4 NONE
647 TYPE0OUT005 1 5 NONE
648 TYPE0OUT006 1 6 NONE
649 TYPE0OUT007 1 7 NONE
650 TYPE0OUT008 1 8 NONE
651 TYPE0OUT009 1 9 NONE
652 TYPE0OUT010 1 10 NONE
653 TYPE0OUT011 1 11 NONE
654 TYPE0OUT012 1 12 NONE
655 TYPE0OUT013 1 19 NONE
656 TYPE0OUT014 1 21 NONE
*
* NUM_DIST2, used for Dispersion Power Law (always 0)
657 NUM_DIST001 0
*
* NSMPLS2, used for non-uniform Bin Sampling (always 0)
658 M4NSMPLS001 0
.
******** TERMINATOR RECORD ENCOUNTERED -- END OF BASE CASE USER INPUT ********

USER INPUT PROCESSING SUMMARY - BASE CASE

NUMBER OF RECORDS READ = 899
NUMBER OF BLANK OR COMMENT RECORDS READ = 240
NUMBER OF TERMINATOR RECORDS = 1
NUMBER OF RECORDS PROCESSED = 658
NUMBER OF PROCESSED RECORDS DUPLICATED = 2
NUMBER OF PROCESSED RECORDS SORTED = 656
****

Decay Chain # Ba-139

Decay Chain # Ba-140 La-140
Fraction of Ba-140 going to La-140 in this chain = 1.000000

Decay Chain # Ce-143 Pr-143
Fraction of Ce-143 going to Pr-143 in this chain = 1.000000

Decay Chain # Ce-144 Pr-144
Fraction of Ce-144 going to Pr-144 in this chain = 0.982200

Decay Chain # Ce-144 Pr-144m Pr-144
Fraction of Ce-144 going to Pr-144m in this chain = 0.017800
Fraction of Ce-144 going to Pr-144 in this chain = 0.017782
Fraction of Pr-144m going to Pr-144 in this chain = 0.999000

Decay Chain # Cm-242 Pu-238
Fraction of Cm-242 going to Pu-238 in this chain = 1.000000

Decay Chain # Cm-244 Pu-240
Fraction of Cm-244 going to Pu-240 in this chain = 1.000000

Decay Chain # Co-58

Decay Chain # Co-60

Decay Chain # Cs-134

Decay Chain # Cs-136

Decay Chain # Cs-137 Ba-137m
Fraction of Cs-137 going to Ba-137m in this chain = 0.946000

Decay Chain # I-133 Xe-133
Fraction of I-133 going to Xe-133 in this chain = 0.971000

Decay Chain # I-134

Decay Chain # I-135 Xe-135
Fraction of I-135 going to Xe-135 in this chain = 0.846000

Decay Chain # I-135 Xe-135m Xe-135
Fraction of I-135 going to Xe-135m in this chain = 0.154000
Fraction of I-135 going to Xe-135 in this chain = 0.153985
Fraction of Xe-135m going to Xe-135 in this chain = 0.999900

Decay Chain # Kr-85m Kr-85
Fraction of Kr-85m going to Kr-85 in this chain = 0.211000

Decay Chain # Kr-87

Decay Chain # Kr-88 Rb-88
Fraction of Kr-88 going to Rb-88 in this chain = 1.000000

Decay Chain # La-141 Ce-141
Fraction of La-141 going to Ce-141 in this chain = 1.000000

Decay Chain # La-142

Decay Chain # Mo-99 Tc-99m
Fraction of Mo-99 going to Tc-99m in this chain = 0.876000

Decay Chain # Nd-147

Decay Chain # Np-239 Pu-239
Fraction of Np-239 going to Pu-239 in this chain = 1.000000

Decay Chain # Pu-241 Am-241
Fraction of Pu-241 going to Am-241 in this chain = 1.000000

Decay Chain # Rb-86

Decay Chain # Ru-103 Rh-103m
Fraction of Ru-103 going to Rh-103m in this chain = 0.997000

Decay Chain # Ru-105 Rh-105
Fraction of Ru-105 going to Rh-105 in this chain = 1.000000

Decay Chain # Ru-106 Rh-106
Fraction of Ru-106 going to Rh-106 in this chain = 1.000000

Decay Chain # Sr-89

Decay Chain # Sr-90 Y-90
Fraction of Sr-90 going to Y-90 in this chain = 1.000000

Decay Chain # Sr-91 Y-91
Fraction of Sr-91 going to Y-91 in this chain = 0.422000

Decay Chain # Sr-91 Y-91m Y-91
Fraction of Sr-91 going to Y-91m in this chain = 0.578000
Fraction of Sr-91 going to Y-91 in this chain = 0.578000
Fraction of Y-91m going to Y-91 in this chain = 1.000000

Decay Chain # Sr-92 Y-92
Fraction of Sr-92 going to Y-92 in this chain = 1.000000

Decay Chain # Te-127m Te-127
Fraction of Te-127m going to Te-127 in this chain = 0.976000

Decay Chain # Te-129m Te-129
Fraction of Te-129m going to Te-129 in this chain = 0.650000

Decay Chain # Te-131m I-131
Fraction of Te-131m going to I-131 in this chain = 0.778000

Decay Chain # Te-131m Te-131 I-131
Fraction of Te-131m going to Te-131 in this chain = 0.222000
Fraction of Te-131m going to I-131 in this chain = 0.222000
Fraction of Te-131 going to I-131 in this chain = 1.000000

Decay Chain # Te-132 I-132
Fraction of Te-132 going to I-132 in this chain = 1.000000

Decay Chain # Y-93

Decay Chain # Zr-95 Nb-95
Fraction of Zr-95 going to Nb-95 in this chain = 0.993000

Decay Chain # Zr-97 Nb-97
Fraction of Zr-97 going to Nb-97 in this chain = 0.053000

Decay Chain # Zr-97 Nb-97m Nb-97
Fraction of Zr-97 going to Nb-97m in this chain = 0.947000
Fraction of Zr-97 going to Nb-97 in this chain = 0.947000
Fraction of Nb-97m going to Nb-97 in this chain = 1.000000

Using distance dispersion model for sigma-y/sigma-z

Using NEW Plume Meander model for sigma-y

THE DENSITY PLUME BUOYANCY MODEL IS IN EFFECT

RELEASED INVENTORY OF ALL PLUMES

Rel #	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25
Kr-85	3.28E+12	1.56E+14	5.30E+13	3.50E+13	2.72E+13	2.25E+13	1.85E+13	1.51E+13	1.18E+13	9.58E+12	7.65E+12	6.10E+12	5.09E+12	4.27E+12	3.59E+12	3.03E+12	2.56E+12	2.16E+12	1.83E+12	1.55E+12	1.31E+12	1.11E+12	9.41E+11	7.96E+11	6.75E+11
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	2.39E+12	1.13E+14	3.83E+13	2.51E+13	1.95E+13	1.60E+13	1.31E+13	1.06E+13	8.25E+12	6.66E+12	5.28E+12	4.19E+12	3.48E+12	2.90E+12	2.43E+12	2.04E+12	1.71E+12	1.44E+12	1.21E+12	1.02E+12	8.58E+11	7.22E+11	6.08E+11	5.12E+11	4.32E+11
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	3.04E+14	1.88E+15	1.12E+15	8.65E+14	5.83E+14	2.29E+14	2.25E+13	1.79E+13	1.17E+13	7.88E+12	1.48E+13	6.53E+12	3.08E+12	2.17E+12	1.55E+12	1.10E+12	7.82E+11	5.53E+11	4.00E+11	2.81E+11	1.95E+11	1.44E+11	1.02E+11	6.80E+10	5.10E+10
Cs-136	1.16E+13	7.16E+13	4.26E+13	3.27E+13	2.20E+13	8.64E+12	8.45E+11	6.73E+11	4.39E+11	2.94E+11	5.53E+11	2.43E+11	1.14E+11	8.03E+10	5.72E+10	4.04E+10	2.88E+10	2.03E+10	1.46E+10	1.03E+10	7.13E+09	5.26E+09	3.70E+09	2.46E+09	1.84E+09
Cs-137	2.26E+14	1.40E+15	8.37E+14	6.43E+14	4.34E+14	1.71E+14	1.67E+13	1.33E+13	8.73E+12	5.87E+12	1.10E+13	4.86E+12	2.29E+12	1.61E+12	1.15E+12	8.16E+11	5.82E+11	4.11E+11	2.97E+11	2.09E+11	1.46E+11	1.08E+11	7.59E+10	5.06E+10	3.80E+10
Rb-86	6.64E+11	4.11E+12	2.45E+12	1.88E+12	1.26E+12	4.97E+11	4.86E+10	3.87E+10	2.53E+10	1.70E+10	3.19E+10	1.40E+10	6.60E+09	4.64E+09	3.31E+09	2.34E+09	1.67E+09	1.18E+09	8.49E+08	5.95E+08	4.14E+08	3.06E+08	2.15E+08	1.43E+08	1.07E+08
Rb-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	3.94E+13	2.40E+14	1.44E+14	1.10E+14	7.37E+13	2.75E+13	7.86E+11	5.07E+11	3.26E+11	2.16E+11	1.42E+12	8.14E+11	3.80E+11	2.70E+11	1.93E+11	1.36E+11	9.61E+10	6.79E+10	4.78E+10	3.58E+10	2.38E+10	1.58E+10	1.18E+10	7.88E+09	7.86E+09
Sr-89	8.78E+13	5.36E+14	3.23E+14	2.47E+14	1.65E+14	6.18E+13	1.77E+12	1.14E+12	7.37E+11	4.88E+11	3.22E+12	1.85E+12	8.64E+11	6.16E+11	4.41E+11	3.12E+11	2.20E+11	1.56E+11	1.10E+11	8.24E+10	5.49E+10	3.66E+10	2.74E+10	1.83E+10	1.83E+10
Sr-90	3.21E+13	1.96E+14	1.18E+14	9.02E+13	6.05E+13	2.26E+13	6.49E+11	4.19E+11	2.70E+11	1.79E+11	1.18E+12	6.79E+11	3.18E+11	2.26E+11	1.62E+11	1.15E+11	8.11E+10	5.75E+10	4.06E+10	3.04E+10	2.03E+10	1.35E+10	1.01E+10	6.76E+09	6.76E+09
Sr-91	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-137m	2.14E+14	1.33E+15	7.92E+14	6.09E+14	4.10E+14	1.62E+14	1.58E+13	1.26E+13	8.26E+12	5.55E+12	1.04E+13	4.60E+12	2.17E+12	1.53E+12	1.09E+12	7.72E+11	5.51E+11	3.89E+11	2.81E+11	1.98E+11	1.38E+11	1.02E+11	7.18E+10	4.79E+10	3.59E+10
I-131	3.16E+13	2.30E+14	1.10E+14	8.16E+13	5.43E+13	2.16E+13	2.60E+12	1.89E+12	1.20E+12	7.93E+11	1.35E+12	5.73E+11	2.61E+11	1.82E+11	1.25E+11	9.05E+10	6.01E+10	4.28E+10	2.98E+10	2.12E+10	1.27E+10	1.27E+10	8.41E+09	4.19E+09	4.17E+09

I-132	3.61E+11	2.27E+12	1.30E+12	9.87E+11	6.58E+11	2.54E+11	2.08E+10	1.59E+10	1.02E+10	6.84E+09	1.49E+10	6.90E+09	3.23E+09	2.27E+09	1.58E+09	1.12E+09	7.78E+08	5.69E+08	4.03E+08	2.79E+08	1.98E+08	1.18E+08	1.17E+08	3.85E+07	7.63E+07
I-133	6.15E+01	4.37E+02	2.03E+02	1.46E+02	9.42E+01	3.64E+01	4.25E+00	3.00E+00	1.84E+00	1.18E+00	1.95E+00	8.06E-01	3.57E-01	2.42E-01	1.61E-01	1.13E-01	7.29E-02	5.04E-02	3.41E-02	2.36E-02	1.37E-02	1.32E-02	8.53E-03	4.13E-03	3.99E-03
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-127	1.25E+13	7.95E+13	4.60E+13	3.51E+13	2.36E+13	9.19E+12	7.60E+11	5.85E+11	3.81E+11	2.56E+11	5.63E+11	2.63E+11	1.24E+11	8.81E+10	6.20E+10	4.40E+10	3.10E+10	2.28E+10	1.63E+10	1.14E+10	8.14E+09	4.88E+09	4.88E+09	1.63E+09	3.25E+09
Te-127m	1.28E+13	8.12E+13	4.70E+13	3.59E+13	2.41E+13	9.38E+12	7.76E+11	5.97E+11	3.89E+11	2.62E+11	5.75E+11	2.68E+11	1.27E+11	9.00E+10	6.33E+10	4.50E+10	3.16E+10	2.33E+10	1.66E+10	1.16E+10	8.31E+09	4.99E+09	4.99E+09	1.66E+09	3.32E+09
Te-129	1.51E+13	9.60E+13	5.56E+13	4.24E+13	2.85E+13	1.11E+13	9.15E+11	7.04E+11	4.58E+11	3.08E+11	6.77E+11	3.16E+11	1.49E+11	1.06E+11	7.43E+10	5.27E+10	3.71E+10	2.73E+10	1.95E+10	1.36E+10	9.72E+09	5.83E+09	5.82E+09	1.94E+09	3.88E+09
Te-129m	2.33E+13	1.47E+14	8.54E+13	6.51E+13	4.38E+13	1.70E+13	1.41E+12	1.08E+12	7.03E+11	4.74E+11	1.04E+12	4.85E+11	2.29E+11	1.62E+11	1.14E+11	8.10E+10	5.70E+10	4.19E+10	2.99E+10	2.09E+10	1.49E+10	8.96E+09	8.95E+09	2.98E+09	5.96E+09
Te-131m	8.99E+04	5.58E+05	3.16E+05	2.36E+05	1.55E+05	5.89E+04	4.76E+03	3.58E+03	2.28E+03	1.50E+03	3.22E+03	1.47E+03	6.78E+02	4.71E+02	3.24E+02	2.25E+02	1.54E+02	1.11E+02	7.76E+01	5.31E+01	3.71E+01	2.17E+01	2.12E+01	6.92E+00	1.35E+01
Te-132	3.51E+11	2.21E+12	1.27E+12	9.58E+11	6.39E+11	2.46E+11	2.02E+10	1.54E+10	9.95E+09	6.64E+09	1.45E+10	6.69E+09	3.13E+09	2.21E+09	1.54E+09	1.08E+09	7.56E+08	5.52E+08	3.91E+08	2.71E+08	1.92E+08	1.14E+08	1.13E+08	3.74E+07	7.41E+07
Te-131	2.02E+04	1.26E+05	7.12E+04	5.31E+04	3.49E+04	1.33E+04	1.07E+03	8.07E+02	5.13E+02	3.38E+02	7.25E+02	3.31E+02	1.53E+02	1.06E+02	7.29E+01	5.06E+01	3.48E+01	2.50E+01	1.75E+01	1.20E+01	8.34E+00	4.89E+00	4.78E+00	1.56E+00	3.04E+00
Rh-105	4.30E+01	3.83E+02	1.19E+03	2.01E+03	1.71E+03	2.08E+03	2.03E+03	1.72E+03	1.12E+03	7.36E+02	4.32E+02	1.88E+01	8.75E+00	6.14E+00	4.31E+00	3.06E+00	2.18E+00	1.56E+00	1.14E+00	8.24E-01	6.34E-01	4.66E-01	3.66E-01	2.99E-01	2.49E-01
Ru-103	2.24E+09	2.04E+10	6.45E+10	1.11E+11	9.60E+10	1.19E+11	1.19E+11	1.02E+11	6.76E+10	4.54E+10	2.72E+10	1.21E+09	5.72E+08	4.09E+08	2.93E+08	2.12E+08	1.53E+08	1.12E+08	8.32E+07	6.15E+07	4.82E+07	3.61E+07	2.89E+07	2.40E+07	2.04E+07
Ru-105	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.92E+09	1.74E+10	5.53E+10	9.52E+10	8.24E+10	1.02E+11	1.02E+11	8.79E+10	5.82E+10	3.91E+10	2.34E+10	1.04E+09	4.94E+08	3.53E+08	2.53E+08	1.83E+08	1.33E+08	9.72E+07	7.21E+07	5.33E+07	4.18E+07	3.13E+07	2.51E+07	2.09E+07	1.78E+07
Rh-103m	2.24E+09	2.03E+10	6.43E+10	1.11E+11	9.58E+10	1.19E+11	1.18E+11	1.02E+11	6.75E+10	4.53E+10	2.71E+10	1.21E+09	5.71E+08	4.08E+08	2.92E+08	2.11E+08	1.53E+08	1.12E+08	8.31E+07	6.13E+07	4.81E+07	3.60E+07	2.88E+07	2.40E+07	2.04E+07
Rh-106	1.92E+09	1.74E+10	5.53E+10	9.52E+10	8.24E+10	1.02E+11	1.02E+11	8.79E+10	5.82E+10	3.91E+10	2.34E+10	1.04E+09	4.94E+08	3.53E+08	2.53E+08	1.83E+08	1.33E+08	9.72E+07	7.21E+07	5.33E+07	4.18E+07	3.13E+07	2.51E+07	2.09E+07	1.78E+07
Nb-95	3.33E+11	3.06E+12	9.64E+12	1.66E+13	1.43E+13	1.78E+13	1.77E+13	1.53E+13	1.01E+13	6.78E+12	4.06E+12	1.80E+11	8.54E+10	6.09E+10	4.37E+10	3.15E+10	2.30E+10	1.67E+10	1.23E+10	9.33E+09	7.03E+09	5.56E+09	4.41E+09	3.43E+09	3.10E+09
Co-58	4.04E+07	3.71E+08	1.17E+09	2.01E+09	1.74E+09	2.17E+09	2.15E+09	1.86E+09	1.23E+09	8.25E+08	4.94E+08	2.20E+07	1.04E+07	7.42E+06	5.33E+06	3.85E+06	2.80E+06	2.04E+06	1.50E+06	1.14E+06	8.60E+05	6.79E+05	5.39E+05	4.19E+05	3.79E+05
Co-60	1.77E+08	1.62E+09	5.12E+09	8.81E+09	7.62E+09	9.49E+09	9.44E+09	8.14E+09	5.39E+09	3.62E+09	2.17E+09	9.65E+07	4.57E+07	3.26E+07	2.34E+07	1.69E+07	1.23E+07	8.99E+06	6.61E+06	5.02E+06	3.79E+06	3.00E+06	2.38E+06	1.85E+06	1.67E+06
Mo-99	2.55E+07	2.32E+08	7.24E+08	1.23E+09	1.06E+09	1.30E+09	1.28E+09	1.09E+09	7.16E+08	4.76E+08	2.82E+08	1.24E+07	5.83E+06	4.11E+06	2.93E+06	2.09E+06	1.51E+06	1.09E+06	7.91E+05	5.95E+05	4.44E+05	3.47E+05	2.73E+05	2.10E+05	1.88E+05
Tc-99m	2.46E+07	2.24E+08	6.98E+08	1.19E+09	1.02E+09	1.25E+09	1.23E+09	1.05E+09	6.90E+08	4.59E+08	2.72E+08	1.20E+07	5.62E+06	3.96E+06	2.82E+06	2.01E+06	1.45E+06	1.05E+06	7.62E+05	5.73E+05	4.28E+05	3.35E+05	2.63E+05	2.03E+05	1.81E+05
Nb-97	0.00E+00	1.36E-14	2.41E-14	2.33E-14	1.13E-14	7.93E-15	4.43E-15	2.14E-15	7.97E-16	3.01E-16	1.01E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-97m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	8.50E+04	3.11E+05	6.86E+06	3.60E+06	2.71E+06	4.67E+06	2.66E+06	1.05E+06	3.04E+03	1.11E+03	5.06E+02	3.04E+02	3.03E+02	3.03E+02	3.03E+02	3.02E+02	2.01E+02	3.02E+02	3.02E+02	3.01E+02	3.01E+02	3.01E+02	3.01E+02	3.00E+02	3.00E+02
Ce-143	5.80E-04	2.08E-03	4.51E-02	2.32E-02	1.71E-02	2.89E-02	1.61E-02	6.24E-03	1.77E-05	6.36E-06	2.83E-06	1.66E-06	1.63E-06	1.59E-06	1.56E-06	1.53E-06	9.98E-07	1.47E-06	1.44E-06	1.41E-06	1.38E-06	1.35E-06	1.32E-06	1.29E-06	1.27E-06
Ce-144	1.47E+05	5.37E+05	1.19E+07	6.23E+06	4.69E+06	8.09E+06	4.61E+06	1.82E+06	5.28E+03	1.94E+03	8.80E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	3.52E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.27E+02
Np-239	3.44E+01	1.24E+02	2.72E+03	1.41E+03	1.05E+03	1.79E+03	1.01E+03	3.93E+02	1.12E+00	4.07E-01	1.83E-01	1.08E-01	1.07E-01	1.06E-01	1.04E-01	1.03E-01	6.80E-02	1.01E-01	9.95E-02	9.83E-02	9.71E-02	9.59E-02	9.47E-02	9.36E-02	9.24E-02
Pu-238	9.24E+02	3.38E+03	7.47E+04	3.93E+04	2.95E+04	5.10E+04	2.91E+04	1.15E+04	3.33E+01	1.22E+01	5.55E+00	3.33E+00	3.33E+00	3.33E+00	3.33E+00	3.33E+00	2.22E+00	3.33E+00	3.33E+00	3.33E+00	3.33E+00	3.33E+00	3.33E+00	3.33E+00	3.33E+00
Pu-239	5.73E+01	2.09E+02	4.63E+03	2.43E+03	1.83E+03	3.16E+03	1.80E+03	7.12E+02	2.06E+00	7.57E-01	3.44E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	1.38E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01
Pu-240	1.20E+02	4.38E+02	9.69E+03	5.09E+03	3.83E+03	6.61E+03	3.77E+03	1.49E+03	4.32E+00	1.58E+00	7.20E-01	4.32E-01	4.32E-01	4.32E-01	4.32E-01	4.32E-01	2.88E-01	4.32E-01	4.32E-01	4.32E-01	4.32E-01	4.32E-01	4.32E-01	4.32E-01	4.32E-01
Pu-241	2.58E+04	9.44E+04	2.09E+06	1.10E+06	8.25E+05	1.42E+06	8.12E+05	3.21E+05	9.30E+02	3.41E+02	1.55E+02	9.30E+01	9.30E+01	9.30E+01	9.30E+01	9.30E+01	6.20E+01	9.30E+01	9.30E+01	9.30E+01	9.30E+01	9.30E+01	9.30E+01	9.30E+01	9.30E+01
Zr-95	1.24E+05	4.54E+05	1.00E+07	5.27E+06	3.96E+06	6.83E+06	3.90E+06	1.54E+06	4.46E+03	1.63E+03	7.43E+02	4.45E+02	4.45E+02	4.45E+02	4.45E+02	4.45E+02	2.96E+02	4.44E+02	4.44E+02	4.44E+02	4.44E+02	4.43E+02	4.43E+02	4.43E+02	4.43E+02
Zr-97	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am-241	4.70E+01	1.72E+02	3.80E+03	2.00E+03	1.50E+03	2.59E+03	1.48E+03	5.84E+02	1.72E+00	6.29E-01	2.29E-01	1.72E-01	1.72E-01	1.72E-01	1.72E-01	1.72E-01	1.14E-01	1.72E-01	1.72E-01	1.72E-01	1.72E-01	1.72E-01	1.72E-01	1.15E-01	1.72E-01
Cm-242	1.12E+04	4.09E+04	9.03E+05	4.74E+05	3.57E+05	6.16E+05	3.51E+05	1.39E+05	4.07E+02	1.49E+02	5.43E+01	4.07E+01	4.07E+01	4.07E+01	4.07E+01	4.07E+01	2.71E+01	4.07E+01	4.07E+01	4.07E+01	4.07E+01	4.06E+01	4.06E+01	2.71E+01	4.06E+01
Cm-244	1.28E+03	4.69E+03	1.04E+05	5.44E+04	4.10E+04	7.07E+04	4.03E+04	1.59E+04	4.68E+01	1.72E+01	6.24E+00	4.68E+00	4.68E+00	4.68E+00	4.68E+00	4.68E+00	3.12E+00	4.68E+00	4.68E+00	4.68E+00	4.68E+00	4.68E+00	4.68E+00	3.12E+00	4.68E+00
La-140	2.10E+13	1.30E+14	7.95E+13	6.15E+13	4.19E+13	1.59E+13	4.60E+11	3.01E+11	1.96E+11	1.32E+11	8.78E+11	5.10E+11	2.41E+11	1.73E+11	1.25E+11	8.96E+10	6.38E+10	4.56E+10	3.24E+10	2.45E+10	1.65E+10	1.11E+10	8.36E+09	5.61E+09	5.65E+09
La-141	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nd-147	6.79E+03	2.48E+04	5.46E+05	2.86E+05	2.15E+05	3.70E+05	2.10E+05	8.29E+04	2.43E+02	8.88E+01	3.22E+01	2.41E+01	2.40E+01	2.40E+01	2.39E+01	2.38E+01	1.59E+01	2.37E+01	2.37E+01	2.36E+01	2.35E+01	2.35E+01	2.34E+01	1.56E+01	2.33E+01
Pr-143	2.66E+04	9.70E+04	2.14E+06	1.12E+06	8.42E+05	1.45E+06	8.25E+05	3.25E+05	9.53E+02	3.49E+02	1.27E+02	9.47E+01	9.45E+01	9.43E+01	9.41E+01	9.39E+01	6.25E+01	9.35E+01	9.33E+01	9.31E+01	9.29E+01	9.27E+01	9.25E+01	6.15E+01	9.21E+01
Y-90	1.16E+13	7.23E+13	4.44E+13	3.45E+13	2.36E+13	8.96E+12	2.61E+11	1.71E+11	1.12E+11	7.55E+10	5.06E+11	2.95E+11	1.40E+11	1.01E+11	7.33E+10	5.26E+10	3.76E+10	2.70E+10	1.93E+10	1.46E+10	9.86E+09	6.65E+09	5.04E+09	3.40E+09	3.43E+09
Y-91	8.13E+04	2.97E+05	6.57E+06	3.45E+06	2.59E+06	4.47E+06	2.55E+06	1.01E+06	2.96E+03	1.08E+03	3.94E+02	2.95E+02	2.95E+02	2.95E+02	2.95E+02	2.95E+02	1.96E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	2.94E+02	1.96E+02	2.93E+02
Y-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	3.17E-23	1.09E-22	2.25E-21	1.10E-21	7.75E-22	1.25E-21	6.65E-22	2.45E-22	6.73E-25	2.30E-25	7.82E-26	5.48E-26	5.11E-26	4.77E-26	4.46E-26	4.16E-26	2.59E-26	3.63E-26	3.39E-26	3.16E-26	2.95E-26	2.76E-26	2.57E-26	1.60E-26	2.24E-26
Y-91m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	1.47E+05	5.37E+05	1.19E+07	6.23E+06	4.69E+06	8.09E+06	4.61E+06	1.82E+06	5.28E+03	1.94E+03	8.80E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	3.52E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.28E+02	5.27E+02
Pr-144m	2.61E+03	9.55E+03	2.11E+05	1.11E+05	8.34E+04	1.44E+05	8.21E+04	3.24E+04	9.40E+01	3.45E+01	1.57E+01	9.40E+00	9.40E+00	9.40E+00	9.40E+00	9.40E+00	6.26E+00	9.40E+00	9.40E+00	9.39E+00	9.39E+00	9.39E+00	9.39E+00	9.39E+00	9.39E+00

Rel #	26	27	28	29	30	31
Kr-85	5.71E+11	4.84E+11	4.10E+11	3.47E+11	2.93E+11	2.49E+11
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	3.63E+11	3.06E+11	2.58E+11	2.17E+11	1.83E+11	1.54E+11
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	4.25E+10	2.55E+10	1.70E+10	1.70E+10	8.50E+09	8.50E+09
Cs-136	1.53E+09	9.18E+08	6.10E+08	6.09E+08	3.04E+08	3.03E+08
Cs-137	3.16E+10	1.90E+10	1.27E+10	1.27E+10	6.33E+09	6.33E+09
Rb-86	8.93E+07	5.35E+07	3.56E+07	3.56E+07	1.78E+07	1.77E+07
Rb-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	3.92E+09	3.91E+09	0.00E+00	3.90E+09	0.00E+00	0.00E+00
Sr-89	9.13E+09	9.12E+09	0.00E+00	9.11E+09	0.00E+00	0.00E+00
Sr-90	3.38E+09	3.38E+09	0.00E+00	3.38E+09	0.00E+00	0.00E+00
Sr-91	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-137m	2.99E+10	1.80E+10	1.20E+10	1.20E+10	5.99E+09	5.99E+09
I-131	0.00E+00	4.14E+09	3.27E+00	0.00E+00	0.00E+00	0.00E+00
I-132	0.00E+00	3.75E+07	3.72E+07	0.00E+00	0.00E+00	0.00E+00
I-133	0.00E+00	3.73E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-127	0.00E+00	1.63E+09	1.63E+09	0.00E+00	0.00E+00	0.00E+00
Te-127m	0.00E+00	1.66E+09	1.66E+09	0.00E+00	0.00E+00	0.00E+00
Te-129	0.00E+00	1.93E+09	1.93E+09	0.00E+00	0.00E+00	0.00E+00
Te-129m	0.00E+00	2.97E+09	2.97E+09	0.00E+00	0.00E+00	0.00E+00
Te-131m	0.00E+00	6.45E+00	6.31E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	0.00E+00	3.64E+07	3.61E+07	0.00E+00	0.00E+00	0.00E+00
Te-131	0.00E+00	1.45E+00	1.42E+00	0.00E+00	0.00E+00	0.00E+00
Rh-105	2.15E-01	1.83E-01	1.66E-01	1.49E-01	1.33E-01	1.43E-01
Ru-103	1.80E+07	1.56E+07	1.44E+07	1.32E+07	1.20E+07	1.32E+07
Ru-105	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.57E+07	1.36E+07	1.25E+07	1.15E+07	1.04E+07	1.15E+07
Rh-103m	1.80E+07	1.56E+07	1.43E+07	1.31E+07	1.19E+07	1.31E+07
Rh-106	1.57E+07	1.36E+07	1.25E+07	1.15E+07	1.04E+07	1.15E+07
Nb-95	2.61E+09	2.28E+09	2.11E+09	2.11E+09	1.79E+09	1.95E+09
Co-58	3.19E+05	2.79E+05	2.59E+05	2.59E+05	2.19E+05	2.39E+05
Co-60	1.41E+06	1.23E+06	1.15E+06	1.15E+06	9.69E+05	1.06E+06
Mo-99	1.57E+05	1.36E+05	1.25E+05	1.23E+05	1.03E+05	1.12E+05
Tc-99m	1.51E+05	1.31E+05	1.20E+05	1.19E+05	9.96E+04	1.08E+05
Nb-97	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-97m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	3.00E+02	2.99E+02	2.99E+02	2.99E+02	2.99E+02	2.98E+02
Ce-143	1.24E-06	1.21E-06	1.19E-06	1.16E-06	1.14E-06	1.12E-06
Ce-144	5.27E+02	5.27E+02	5.27E+02	5.27E+02	5.27E+02	5.27E+02
Np-239	9.13E-02	9.02E-02	8.91E-02	8.80E-02	8.69E-02	8.59E-02
Pu-238	3.33E+00	3.33E+00	3.33E+00	3.33E+00	3.33E+00	3.33E+00
Pu-239	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01
Pu-240	4.32E-01	4.32E-01	4.32E-01	4.32E-01	4.32E-01	4.32E-01
Pu-241	9.30E+01	9.30E+01	9.30E+01	9.30E+01	9.30E+01	9.30E+01
Zr-95	4.43E+02	4.42E+02	4.42E+02	4.42E+02	4.42E+02	4.42E+02
Zr-97	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am-241	1.72E-01	1.72E-01	1.72E-01	1.72E-01	2.29E-01	1.72E-01
Cm-242	4.06E+01	4.06E+01	4.06E+01	4.06E+01	5.41E+01	4.06E+01
Cm-244	4.68E+00	4.68E+00	4.68E+00	4.68E+00	6.24E+00	4.68E+00
La-140	2.84E+09	2.86E+09	3.53E+01	2.90E+09	4.55E+01	3.36E+01

La-141	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nd-147	2.32E+01	2.32E+01	2.31E+01	2.30E+01	3.06E+01	2.29E+01
Pr-143	9.19E+01	9.17E+01	9.15E+01	9.13E+01	1.22E+02	9.09E+01
Y-90	1.73E+09	1.75E+09	3.51E+01	1.79E+09	4.59E+01	3.40E+01
Y-91	2.93E+02	2.93E+02	2.93E+02	2.93E+02	3.90E+02	2.92E+02
Y-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	2.09E-26	1.96E-26	1.83E-26	1.71E-26	2.12E-26	1.49E-26
Y-91m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	5.27E+02	5.27E+02	5.27E+02	5.27E+02	5.27E+02	5.27E+02
Pr-144m	9.39E+00	9.39E+00	9.39E+00	9.39E+00	9.38E+00	9.38E+00

MAXIMUM HEIGHT PLUME RISE FLAG = DAY_AND_NIGHT

READING FROM A WEATHER FILE WITH THE FOLLOWING HEADER:
Peach Bottom MACCS2 2006 Data
64 WD for SOAR CA Trials
Weather file uses 60 minute intervals
Weather file uses 64 wind directions
METEOROLOGICAL DATA FILE CONTAINS 602 PERIODS OF OBSERVED RAIN DATA.
ACCUMULATED RAIN MEASUREMENTS TOTALED 44.42 INCHES FOR THE YEAR.
MORNING LID HEIGHTS (M) FOR 4 SEASONS = 760 650 500 570
AFTERNOON LID HEIGHTS (M) FOR 4 SEASONS = 770 1450 1620 1140
NON-ZERO WINDSPEEDS LESS THAN 0.5 M/S ARE SET TO 0.5 M/S
NUMTRI= 984

* * * * METEOROLOGICAL BIN SUMMARY * * * *

BIN PRIORITIES
RI XX - RAIN INTENSITY I WITHIN THE INTERVAL ENDING AT XX
INTERVAL ENDPOINTS ARE IN KILOMETERS FROM THE ACCIDENT SITE, THE 5 INTERVAL ENDPOINTS ARE 3 6 11 21 32
RAIN INTENSITIES ARE IN MILLIMETERS OF RAIN PER HOUR, THE 3 INTENSITY BREAKPOINTS ARE 2.0 4.0 6.0
S V - INITIAL WEATHER CONDITIONS WITH STABILITY CLASS S AND WIND SPEED INTERVAL V
STABILITY CLASSES ARE B = A/B, D = C/D, E = E, AND F = F
WIND SPEED INTERVALS ARE IN METERS PER SECOND, 1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (5-7), 6 (GT 7)

WIND DIRECTION

METBIN			1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	B	3	0.017	0.010	0.004	0.011	0.004	0.000	0.003	0.008	0.004	0.010	0.010	0.007	0.004	0.007	0.011	0.016
2	B	4	0.017	0.014	0.012	0.017	0.005	0.007	0.014	0.017	0.010	0.012	0.005	0.017	0.010	0.014	0.026	0.029
3	D	1	0.000	0.000	0.000	0.000	0.000	0.011	0.011	0.011	0.000	0.022	0.011	0.022	0.022	0.000	0.000	0.033
4	D	2	0.017	0.021	0.013	0.010	0.011	0.010	0.010	0.008	0.006	0.008	0.008	0.011	0.011	0.021	0.019	0.013
5	D	3	0.012	0.021	0.011	0.012	0.011	0.007	0.009	0.005	0.011	0.005	0.009	0.005	0.012	0.007	0.012	0.027
6	D	4	0.004	0.005	0.003	0.004	0.008	0.003	0.003	0.005	0.011	0.004	0.008	0.011	0.013	0.011	0.013	0.015
7	D	5	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.005	0.000	0.010	0.005	0.038	0.072
8	D	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
9	E	1	0.023	0.029	0.031	0.004	0.008	0.037	0.012	0.006	0.016	0.010	0.019	0.014	0.004	0.008	0.010	0.010
10	E	2	0.028	0.030	0.021	0.011	0.014	0.011	0.012	0.014	0.010	0.014	0.017	0.025	0.022	0.015	0.022	0.030
11	E	3	0.010	0.013	0.008	0.009	0.005	0.004	0.004	0.005	0.009	0.022	0.027	0.030	0.031	0.023	0.034	0.040
12	E	4	0.008	0.011	0.003	0.006	0.006	0.000	0.008	0.006	0.011	0.008	0.011	0.008	0.028	0.017	0.034	0.056
13	F	1	0.018	0.012	0.041	0.022	0.014	0.024	0.031	0.010	0.027	0.031	0.025	0.024	0.020	0.022	0.027	0.025
14	F	2	0.005	0.004	0.016	0.005	0.009	0.019	0.012	0.027	0.029	0.057	0.060	0.077	0.096	0.055	0.067	0.056
15	F	3	0.000	0.007	0.000	0.000	0.007	0.000	0.007	0.000	0.022	0.058	0.080	0.066	0.117	0.139	0.095	0.102
16	F	4	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.250	0.000	0.250	0.000	0.250	0.000
17	R1	3	0.018	0.018	0.009	0.007	0.007	0.005	0.009	0.002	0.009	0.018	0.014	0.011	0.005	0.002	0.014	0.009
18	R1	6	0.000	0.000	0.016	0.016	0.032	0.032	0.016	0.016	0.000	0.032	0.032	0.000	0.048	0.016	0.000	0.016
19	R1	11	0.018	0.042	0.024	0.012	0.000	0.024	0.024	0.018	0.006	0.018	0.006	0.012	0.024	0.036	0.012	0.024
20	R1	21	0.021	0.034	0.004	0.004	0.004	0.008	0.004	0.025	0.000	0.021	0.021	0.008	0.000	0.013	0.000	0.021
21	R1	32	0.008	0.021	0.000	0.004	0.030	0.017	0.034	0.008	0.004	0.004	0.030	0.004	0.008	0.004	0.021	0.030
22	R2	3	0.026	0.000	0.017	0.035	0.000	0.000	0.009	0.009	0.009	0.000	0.009	0.009	0.009	0.000	0.009	0.000
23	R2	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
24	R2	11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000	0.000	0.000
25	R2	21	0.000	0.000	0.000	0.062	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
26	R2	32	0.080	0.040	0.040	0.000	0.000	0.040	0.000	0.000	0.000	0.200	0.000	0.000	0.000	0.000	0.000	0.000
27	R3	3	0.026	0.000	0.051	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.051	0.051	0.000	0.000	0.000
28	R3	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
29	R3	11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
30	R3	21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.400	0.000	0.000	0.000	0.000
31	R3	32	0.000	0.250	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.250
32	R4	3	0.000	0.000	0.000	0.000	0.000	0.000	0.029	0.029	0.029	0.000	0.000	0.000	0.000	0.000	0.000	0.000
33	R4	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
34	R4	11	0.143	0.143	0.000	0.000	0.143	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.143	0.000
35	R4	21	0.000	0.222	0.111	0.222	0.000	0.000	0.000	0.000	0.000	0.000	0.111	0.000	0.000	0.000	0.000	0.000
36	R4	32	0.067	0.000	0.133	0.000	0.067	0.067	0.067	0.000	0.067	0.000	0.000	0.000	0.000	0.067	0.000	0.000
37	ALL		0.015	0.016	0.013	0.009	0.009	0.010	0.011	0.010	0.012	0.018	0.020	0.021	0.024	0.019	0.024	0.029

METBIN			17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	B	3	0.003	0.013	0.007	0.006	0.010	0.010	0.007	0.006	0.007	0.016	0.010	0.027	0.040	0.035	0.023	0.038
2	B	4	0.005	0.021	0.024	0.038	0.055	0.043	0.041	0.060	0.036	0.050	0.074	0.088	0.064	0.019	0.014	0.005
3	D	1	0.033	0.022	0.000	0.022	0.000	0.000	0.022	0.022	0.011	0.011	0.000	0.022	0.011	0.011	0.000	0.033
4	D	2	0.015	0.011	0.019	0.011	0.025	0.015	0.017	0.021	0.025	0.010	0.023	0.019	0.025	0.029	0.017	0.029
5	D	3	0.011	0.016	0.018	0.021	0.039	0.034	0.032	0.018	0.019	0.048	0.060	0.055	0.046	0.032	0.028	0.048
6	D	4	0.017	0.023	0.057	0.035	0.058	0.048	0.085	0.071	0.065	0.081	0.062	0.039	0.057	0.030	0.009	0.011
7	D	5	0.019	0.024	0.034	0.053	0.072	0.077	0.058	0.115	0.125	0.072	0.096	0.034	0.034	0.014	0.000	0.000
8	D	6	0.031	0.000	0.031	0.031	0.031	0.000	0.000	0.000	0.000	0.469	0.375	0.000	0.000	0.000	0.000	0.000
9	E	1	0.006	0.006	0.014	0.012	0.008	0.004	0.012	0.019	0.014	0.016	0.008	0.004	0.004	0.014	0.004	0.035
10	E	2	0.023	0.019	0.026	0.021	0.020	0.013	0.022	0.020	0.012	0.021	0.026	0.010	0.017	0.009	0.013	0.023
11	E	3	0.045	0.050	0.058	0.044	0.039	0.044	0.031	0.025	0.021	0.040	0.028	0.031	0.021	0.014	0.025	0.016
12	E	4	0.031	0.023	0.045	0.025	0.068	0.071	0.065	0.054	0.054	0.045	0.028	0.011	0.011	0.020	0.014	0.025
13	F	1	0.016	0.016	0.014	0.022	0.010	0.016	0.022	0.014	0.020	0.016	0.033	0.002	0.000	0.002	0.002	0.031
14	F	2	0.031	0.031	0.028	0.017	0.029	0.011	0.020	0.015	0.023	0.013	0.013	0.012	0.013	0.001	0.004	0.025
15	F	3	0.044	0.044	0.015	0.015	0.022	0.007	0.007	0.029	0.000	0.015	0.000	0.007	0.007	0.000	0.007	0.007
16	F	4	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
17	R1	3	0.005	0.007	0.007	0.014	0.009	0.009	0.009	0.011	0.016	0.018	0.025	0.027	0.020	0.018	0.007	0.039
18	R1	6	0.000	0.000	0.016	0.000	0.016	0.016	0.000	0.000	0.000	0.000	0.016	0.000	0.032	0.000	0.016	0.032
19	R1	11	0.000	0.018	0.006	0.006	0.006	0.000	0.018	0.012	0.024	0.006	0.000	0.012	0.012	0.012	0.006	0.012
20	R1	21	0.004	0.008	0.000	0.017	0.013	0.004	0.021	0.013	0.021	0.017	0.034	0.013	0.021	0.017	0.008	0.025
21	R1	32	0.008	0.017	0.008	0.000	0.013	0.008	0.021	0.000	0.013	0.021	0.004	0.004	0.017	0.021	0.034	0.021
22	R2	3	0.000	0.000	0.000	0.000	0.009	0.000	0.000	0.009	0.000	0.009	0.035	0.009	0.009	0.009	0.000	0.017
23	R2	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
24	R2	11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000
25	R2	21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.063	0.000	0.000	0.063	0.000	0.000	0.000
26	R2	32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
27	R3	3	0.000	0.000	0.000	0.000	0.000	0.000	0.026	0.000	0.000	0.103	0.026	0.026	0.026	0.051	0.000	0.103
28	R3	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
29	R3	11	0.000	0.000	0.000	0.000	0.333	0.000	0.000	0.000	0.000	0.333	0.000	0.000	0.000	0.000	0.000	0.000
30	R3	21	0.000	0.000	0.000	0.000	0.000	0.200	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
31	R3	32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
32	R4	3	0.029	0.000	0.000	0.000	0.000	0.029	0.000	0.000	0.000	0.059	0.088	0.029	0.000	0.029	0.059	0.000

33 R4 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
34 R4 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
35 R4 21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
36 R4 32	0.000	0.000	0.000	0.133	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
37 ALL	0.018	0.020	0.025	0.021	0.028	0.023	0.028	0.026	0.025	0.032	0.032	0.024	0.025	0.017	0.013	0.025

METBIN	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1 B 3	0.024	0.048	0.045	0.049	0.021	0.018	0.017	0.008	0.007	0.020	0.024	0.018	0.024	0.020	0.023	0.028
2 B 4	0.000	0.019	0.010	0.010	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
3 D 1	0.000	0.022	0.000	0.044	0.033	0.022	0.033	0.011	0.000	0.022	0.033	0.022	0.044	0.022	0.044	0.056
4 D 2	0.019	0.044	0.040	0.031	0.029	0.013	0.015	0.006	0.006	0.015	0.010	0.008	0.013	0.004	0.013	0.013
5 D 3	0.027	0.016	0.032	0.007	0.007	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
6 D 4	0.011	0.012	0.024	0.001	0.001	0.000	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
7 D 5	0.000	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
8 D 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
9 E 1	0.012	0.008	0.010	0.023	0.006	0.004	0.006	0.006	0.008	0.012	0.008	0.004	0.027	0.019	0.029	0.025
10 E 2	0.011	0.011	0.017	0.006	0.009	0.008	0.003	0.002	0.000	0.005	0.002	0.002	0.002	0.001	0.002	0.008
11 E 3	0.009	0.004	0.008	0.006	0.004	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
12 E 4	0.017	0.011	0.006	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
13 F 1	0.000	0.006	0.002	0.018	0.002	0.000	0.002	0.000	0.000	0.002	0.008	0.002	0.014	0.004	0.016	0.022
14 F 2	0.001	0.000	0.000	0.003	0.001	0.001	0.000	0.001	0.000	0.000	0.001	0.004	0.000	0.000	0.001	0.004
15 F 3	0.007	0.000	0.000	0.007	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
16 F 4	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
17 R1 3	0.025	0.075	0.018	0.011	0.020	0.011	0.016	0.007	0.005	0.020	0.020	0.011	0.011	0.011	0.005	0.016
18 R1 6	0.000	0.016	0.032	0.000	0.000	0.016	0.000	0.032	0.032	0.000	0.048	0.032	0.016	0.000	0.032	0.016
19 R1 11	0.036	0.018	0.018	0.012	0.012	0.006	0.006	0.006	0.012	0.006	0.012	0.006	0.006	0.006	0.012	0.018
20 R1 21	0.017	0.034	0.013	0.017	0.004	0.008	0.004	0.008	0.004	0.004	0.013	0.004	0.013	0.008	0.025	0.030
21 R1 32	0.008	0.025	0.021	0.013	0.008	0.000	0.008	0.013	0.004	0.004	0.000	0.008	0.004	0.000	0.021	0.030
22 R2 3	0.000	0.017	0.035	0.026	0.026	0.017	0.017	0.026	0.000	0.017	0.009	0.009	0.009	0.017	0.043	0.052
23 R2 6	0.250	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.000	0.000	0.000	0.000
24 R2 11	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
25 R2 21	0.000	0.063	0.000	0.000	0.063	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
26 R2 32	0.000	0.040	0.000	0.040	0.040	0.040	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
27 R3 3	0.026	0.000	0.000	0.000	0.026	0.000	0.026	0.000	0.000	0.026	0.000	0.026	0.000	0.000	0.000	0.000
28 R3 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
29 R3 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
30 R3 21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
31 R3 32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
32 R4 3	0.000	0.029	0.029	0.000	0.029	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
33 R4 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	1.000	0.000	0.000	0.000	0.000	0.000
34 R4 11	0.000	0.000	0.143	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.143	0.000	0.143
35 R4 21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.111	0.111	0.000	0.000	0.000
36 R4 32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
37 ALL	0.012	0.019	0.017	0.013	0.009	0.006	0.005	0.003	0.002	0.006	0.006	0.005	0.007	0.005	0.008	0.011

METBIN	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	TOTAL	PER CENT
1 B 3	0.011	0.031	0.021	0.010	0.020	0.007	0.017	0.010	0.013	0.010	0.010	0.014	0.008	0.016	0.008	0.016	708	8.0822
2 B 4	0.000	0.000	0.000	0.000	0.000	0.002	0.002	0.000	0.000	0.002	0.000	0.005	0.005	0.019	0.036	0.026	419	4.7831
3 D 1	0.000	0.044	0.022	0.011	0.011	0.011	0.033	0.011	0.000	0.011	0.022	0.011	0.022	0.000	0.011	0.000	90	1.0274
4 D 2	0.017	0.006	0.002	0.010	0.010	0.011	0.017	0.015	0.011	0.011	0.017	0.019	0.019	0.015	0.013	0.019	524	5.9817
5 D 3	0.000	0.000	0.000	0.000	0.000	0.005	0.009	0.000	0.016	0.027	0.021	0.023	0.028	0.018	0.028	0.030	565	6.4498
6 D 4	0.000	0.000	0.000	0.000	0.000	0.001	0.001	0.003	0.000	0.008	0.001	0.001	0.009	0.020	0.015	0.020	743	8.4817
7 D 5	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.010	0.005	0.014	0.010	208	2.3744
8 D 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.031	0.000	0.000	32	0.3653
9 E 1	0.031	0.027	0.021	0.019	0.014	0.016	0.031	0.023	0.010	0.016	0.041	0.023	0.027	0.027	0.031	0.010	486	5.5479
10 E 2	0.005	0.017	0.011	0.011	0.010	0.012	0.016	0.021	0.020	0.020	0.029	0.025	0.031	0.028	0.031	0.033	1029	11.7466
11 E 3	0.000	0.003	0.000	0.000	0.003	0.003	0.001	0.001	0.010	0.018	0.019	0.019	0.019	0.025	0.025	0.016	773	8.8242
12 E 4	0.000	0.000	0.000	0.000	0.000	0.003	0.003	0.000	0.006	0.003	0.003	0.008	0.014	0.014	0.034	0.065	354	4.0411
13 F 1	0.045	0.035	0.043	0.006	0.061	0.012	0.025	0.004	0.000	0.008	0.012	0.024	0.006	0.006	0.012	0.000	510	5.8219
14 F 2	0.004	0.013	0.009	0.007	0.011	0.005	0.009	0.004	0.001	0.009	0.000	0.005	0.007	0.005	0.007	0.003	750	8.5616
15 F 3	0.000	0.000	0.000	0.000	0.007	0.000	0.000	0.000	0.000	0.015	0.000	0.000	0.015	0.007	0.007	0.007	137	1.5639
16 F 4	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	4	0.0457
17 R1 3	0.007	0.005	0.005	0.023	0.020	0.011	0.011	0.023	0.018	0.023	0.029	0.027	0.034	0.025	0.039	0.020	441	5.0342
18 R1 6	0.063	0.000	0.016	0.016	0.000	0.000	0.000	0.063	0.000	0.000	0.048	0.032	0.016	0.016	0.032	0.016	63	0.7192
19 R1 11	0.030	0.006	0.018	0.018	0.018	0.018	0.012	0.030	0.030	0.012	0.042	0.030	0.024	0.018	0.018	0.024	165	1.8836
20 R1 21	0.025	0.013	0.013	0.017	0.021	0.004	0.025	0.013	0.038	0.059	0.017	0.013	0.017	0.017	0.030	0.042	236	2.6941
21 R1 32	0.025	0.017	0.008	0.013	0.038	0.021	0.021	0.017	0.038	0.051	0.038	0.017	0.008	0.021	0.034	0.025	237	2.7055
22 R2 3	0.000	0.000	0.009	0.017	0.035	0.000	0.017	0.052	0.043	0.017	0.009	0.052	0.070	0.026	0.043	0.043	115	1.3128
23 R2 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.000	0.250	4	0.0457
24 R2 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.250	8	0.0913
25 R2 21	0.000	0.000	0.000	0.000	0.063	0.063	0.000	0.000	0.000	0.000	0.000	0.188	0.125	0.063	0.063	0.000	16	0.1826
26 R2 32	0.000	0.000	0.040	0.000	0.000	0.000	0.000	0.080	0.000	0.000	0.000	0.080	0.120	0.040	0.000	0.080	25	0.2854
27 R3 3	0.000	0.000	0.051	0.000	0.000	0.026	0.026	0.051	0.000	0.077	0.000	0.000	0.000	0.051	0.026	0.026	39	0.4452
28 R3 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	1.000	1	0.0114
29 R3 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.333	0.000	0.000	3	0.0342
30 R3 21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.200	0.000	0.000	0.200	0.000	5	0.0571
31 R3 32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.000	4	0.0457
32 R4 3	0.059	0.000	0.000	0.000	0.000	0.000	0.029	0.029	0.029	0.000	0.029	0.088	0.000	0.029	0.147	0.059	34	0.3881
33 R4 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	1	0.0114
34 R4 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	7	0.0799
35 R4 21	0.000	0.111	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	9	0.1027
36 R4 32	0.133	0.067	0.000	0.000	0.000	0.000	0.000	0.067	0.000	0.000	0.000	0.000	0.000	0.067	0.000	0.000	15	0.1712
37 ALL	0.010	0.012	0.009	0.007	0.013	0.008	0.012	0.011	0.011	0.016	0.016	0.017	0.018	0.019	0.023	0.021	8760	100.0000

* * * * METEOROLOGICAL BIN SUMMARY * * * *

BIN PRIORITIES

RI XX - RAIN INTENSITY I WITHIN THE INTERVAL ENDING AT XX

INTERVAL ENDPOINTS ARE IN KILOMETERS FROM THE ACCIDENT SITE, THE 5 INTERVAL ENDPOINTS ARE 3 6 11 21 32

RAIN INTENSITIES ARE IN MILLIMETERS OF RAIN PER HOUR, THE 3 INTENSITY BREAKPOINTS ARE 2.0 4.0 6.0

S V - INITIAL WEATHER CONDITIONS WITH STABILITY CLASS S AND WIND SPEED INTERVAL V

STABILITY CLASSES ARE B = A/B, D = C/D, E = E, AND F = F

WIND SPEED INTERVALS ARE IN METERS PER SECOND (M/S), 1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (5-7), 6 (GT 7)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1 B 3	12	7	3	8	3	0	2	6	3	7	7	5	3	5	8	11
2 B 4	7	6	5	7	2	3	6	7	4	5	2	7	4	6	11	12
3 D 1	0	0	0	0	0	1	1	1	0	2	1	2	2	0	0	3
4 D 2	9	11	7	5	6	5	5	4	3	4	4	6	6	11	10	7
5 D 3	7	12	6	7	6	4	5	3	6	3	5	3	7	4	7	15
6 D 4	3	4	2	3	6	2	2	4	8	3	6	8	10	8	10	11
7 D 5	0	0	0	0	0	0	0	0	0	0	1	0	2	1	8	15
8 D 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
9 E 1	11	14	15	2	4	18	6	3	8	5	9	7	2	4	5	5
10 E 2	29	31	22	11	14	11	12	14	10	14	18	26	23	15	23	31
11 E 3	8	10	6	7	4	3	3	4	7	17	21	23	24	18	26	31

12 E 4	3	4	1	2	2	0	3	2	4	3	4	3	10	6	12	20
13 F 1	9	6	21	11	7	12	16	5	14	16	13	12	10	11	14	13
14 F 2	4	3	12	4	7	14	9	20	22	43	45	58	72	41	50	42
15 F 3	0	1	0	0	1	0	1	0	3	8	11	9	16	19	13	14
16 F 4	0	0	0	0	0	0	0	0	0	1	1	0	1	0	1	0
17 R1 3	8	8	4	3	3	2	4	1	4	8	6	5	2	1	6	4
18 R1 6	0	0	1	1	2	2	1	1	0	2	2	0	3	1	0	1
19 R1 11	3	7	4	2	0	4	4	3	1	3	1	2	4	6	2	4
20 R1 21	5	8	1	1	1	2	1	6	0	5	5	2	0	3	0	5
21 R1 32	2	5	0	1	7	4	8	2	1	1	7	1	2	1	5	7
22 R2 3	3	0	2	4	0	0	1	1	1	0	1	1	1	0	1	0
23 R2 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
24 R2 11	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0
25 R2 21	0	0	0	1	0	0	2	0	0	0	0	0	0	0	0	0
26 R2 32	2	1	1	0	0	1	0	0	0	5	0	0	0	0	0	0
27 R3 3	1	0	2	0	0	0	0	0	0	0	0	2	2	0	0	0
28 R3 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
29 R3 11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
30 R3 21	0	0	0	0	0	0	0	0	0	0	0	2	0	0	0	0
31 R3 32	0	1	0	0	0	0	0	0	0	0	0	0	0	1	0	1
32 R4 3	0	0	0	0	0	0	1	1	1	0	0	0	0	0	0	0
33 R4 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
34 R4 11	1	1	0	0	1	0	0	0	0	0	0	0	0	0	1	0
35 R4 21	0	2	1	2	0	0	0	0	0	0	1	0	0	0	0	0
36 R4 32	1	0	2	0	1	1	1	0	1	0	0	0	0	1	0	0

METBIN	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1 B 3	2	9	5	4	7	7	5	4	5	11	7	19	28	25	16	27
2 B 4	2	9	10	16	23	18	17	25	15	21	31	37	27	8	6	2
3 D 1	3	2	0	2	0	0	2	2	1	1	0	2	1	1	0	3
4 D 2	8	6	10	6	13	8	9	11	13	5	12	10	13	15	9	15
5 D 3	6	9	10	12	22	19	18	10	11	27	34	31	26	18	16	27
6 D 4	13	17	42	26	43	36	63	53	48	60	46	29	42	22	7	8
7 D 5	4	5	7	11	15	16	12	24	26	15	20	7	7	3	0	0
8 D 6	1	0	1	1	1	0	0	0	0	15	12	0	0	0	0	0
9 E 1	3	3	7	6	4	2	6	9	7	8	4	2	2	7	2	17
10 E 2	24	20	27	22	21	13	23	21	12	22	27	10	18	9	13	24
11 E 3	35	39	45	34	30	34	24	19	16	31	22	24	16	11	19	12
12 E 4	11	8	16	9	24	25	23	19	19	16	10	4	4	7	5	9
13 F 1	8	8	7	11	5	8	11	7	10	8	17	1	0	1	1	16
14 F 2	23	23	21	13	22	8	15	11	17	10	10	9	10	1	3	19
15 F 3	6	6	2	2	3	1	1	4	0	2	0	1	1	0	1	1
16 F 4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
17 R1 3	2	3	3	6	4	4	4	5	7	8	11	12	9	8	3	17
18 R1 6	0	0	1	0	1	1	0	0	0	0	1	0	2	0	1	2
19 R1 11	0	3	1	1	1	0	3	2	4	1	0	2	2	2	1	2
20 R1 21	1	2	0	4	3	1	5	3	5	4	8	3	5	4	2	6
21 R1 32	2	4	2	0	3	2	5	0	3	5	1	1	4	5	8	5
22 R2 3	0	0	0	0	1	0	0	1	0	1	4	1	1	1	0	2
23 R2 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
24 R2 11	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0
25 R2 21	0	0	0	0	0	0	0	0	0	1	0	0	1	0	0	0
26 R2 32	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
27 R3 3	0	0	0	0	0	0	1	0	0	4	1	1	1	2	0	4
28 R3 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
29 R3 11	0	0	0	0	1	0	0	0	0	1	0	0	0	0	0	0
30 R3 21	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0
31 R3 32	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
32 R4 3	1	0	0	0	0	1	0	0	0	2	3	1	0	1	2	0
33 R4 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
34 R4 11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
35 R4 21	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
36 R4 32	0	0	0	2	0	0	0	0	0	0	0	0	0	0	0	0

METBIN	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1 B 3	17	34	32	35	15	13	12	6	5	14	17	13	17	14	16	20
2 B 4	0	8	4	4	1	0	0	0	0	0	0	0	0	0	0	0
3 D 1	0	2	0	4	3	2	3	1	0	2	3	2	4	2	4	5
4 D 2	10	23	21	16	15	7	8	3	3	8	5	4	7	2	7	7
5 D 3	15	9	18	4	4	3	0	0	0	0	0	0	0	0	0	0
6 D 4	8	9	18	1	1	0	1	0	0	0	0	0	0	0	0	0
7 D 5	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0
8 D 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
9 E 1	6	4	5	11	3	2	3	3	4	6	4	2	13	9	14	12
10 E 2	11	11	17	6	9	8	3	2	0	5	2	2	2	1	2	8
11 E 3	7	3	6	5	3	1	0	0	0	0	0	0	0	0	0	0
12 E 4	6	4	2	0	0	0	0	0	0	0	0	0	0	0	0	0
13 F 1	0	3	1	9	1	0	1	0	0	1	4	1	7	2	8	11
14 F 2	1	0	0	2	1	1	0	1	0	0	1	3	0	0	1	3
15 F 3	1	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0
16 F 4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
17 R1 3	11	33	8	5	9	5	7	3	2	9	9	5	5	5	2	7
18 R1 6	0	1	2	0	0	1	0	2	2	0	3	2	1	0	2	1
19 R1 11	6	3	3	2	2	1	1	1	2	1	2	1	1	1	2	3
20 R1 21	4	8	3	4	1	2	1	2	1	1	3	1	3	2	6	7
21 R1 32	2	6	5	3	2	0	2	3	1	1	0	2	1	0	5	7
22 R2 3	0	2	4	3	3	2	2	3	0	2	1	1	1	2	5	6
23 R2 6	1	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0
24 R2 11	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0
25 R2 21	0	1	0	0	1	0	0	0	0	0	0	0	0	0	0	0
26 R2 32	0	1	0	1	1	1	0	0	0	0	0	0	0	0	0	0
27 R3 3	1	0	0	0	1	0	1	0	0	1	0	1	0	0	0	0
28 R3 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
29 R3 11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
30 R3 21	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
31 R3 32	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
32 R4 3	0	1	1	0	1	0	0	0	0	0	0	0	0	0	0	0
33 R4 6	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0
34 R4 11	0	0	1	0	0	0	0	0	0	0	0	0	0	1	0	1
35 R4 21	0	0	0	0	0	0	0	0	0	0	0	1	1	0	0	0
36 R4 32	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

METBIN	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	TOTAL	PER CENT
1 B 3	8	22	15	7	14	5	12	7	9	7	7	10	6	11	6	11	708	8.0822
2 B 4	0	0	0	0	0	1	1	0	0	1	0	2	2	8	15	11	419	4.7831
3 D 1	0	4	2	1	1	1	3	1	0	1	2	1	2	0	1	0	90	1.0274
4 D 2	9	3	1	5	5	6	9	8	6	6	9	10	10	8	7	10	524	5.9817
5 D 3	0	0	0	0	0	3	5	0	9	15	12	13	16	10	16	17	565	6.4498
6 D 4	0	0	0	0	0	1	1	2	0	6	1	1	7	15	11	15	743	8.4817

```
 7 D  5     0   0   0   0   0   0   0   0   0   0   0   0   2   1   3   2  208     2.3744
 8 D  6     0   0   0   0   0   0   0   0   0   0   0   0   0   1   0   0   32     0.3653
 9 E  1    15  13  10   9   7   8  15  11   5   8  20  11  13  13  15   5  486     5.5479
10 E  2     5  18  11  11  10  12  16  22  21  21  30  26  32  29  32  34 1029    11.7466
11 E  3     0   2   0   0   2   2   1   1   8  14  15  15  15  19  19  12  773     8.8242
12 E  4     0   0   0   0   0   1   1   0   2   1   1   3   5   5  12  23  354     4.0411
13 F  1    23  18  22   3  31   6  13   2   0   4   6  12   3   3   6   0  510     5.8219
14 F  2     3  10   7   5   8   4   7   3   1   7   0   4   5   4   5   2  750     8.5616
15 F  3     0   0   0   0   1   0   0   0   0   2   0   0   2   1   1   1  137     1.5639
16 F  4     0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0    4     0.0457
17 R1  3    3   2   2  10   9   5   5  10   8  10  13  12  15  11  17   9  441     5.0342
18 R1  6    4   0   1   1   0   0   0   4   0   0   3   2   1   1   2   1   63     0.7192
19 R1 11    5   1   3   3   3   3   2   5   5   2   7   5   4   3   3   4  165     1.8836
20 R1 21    6   3   3   4   5   1   6   3   9  14   4   3   4   4   7  10  236     2.6941
21 R1 32    6   4   2   3   9   5   5   4   9  12   9   4   2   5   8   6  237     2.7055
22 R2  3    0   0   1   2   4   0   2   6   5   2   1   6   8   3   5   5  115     1.3128
23 R2  6    0   0   0   0   0   0   0   0   0   0   0   0   1   0   0   1    4     0.0457
24 R2 11    0   0   0   0   0   0   0   1   0   0   0   0   0   2   0   2    8     0.0913
25 R2 21    0   0   0   0   1   1   0   0   0   0   0   3   2   1   1   0   16     0.1826
26 R2 32    0   0   1   0   0   0   0   2   0   0   0   2   3   1   0   2   25     0.2854
27 R3  3    0   0   2   0   0   1   1   2   0   3   0   0   0   2   1   1   39     0.4452
28 R3  6    0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   1    1     0.0114
29 R3 11    0   0   0   0   0   0   0   0   0   0   0   0   0   1   0   0    3     0.0342
30 R3 21    0   0   0   0   0   0   0   0   0   0   0   1   0   0   1   0    5     0.0571
31 R3 32    0   0   0   0   0   0   0   0   0   0   0   0   0   1   0   0    4     0.0457
32 R4  3    2   0   0   0   0   0   1   1   1   0   1   3   0   1   5   2   34     0.3881
33 R4  6    0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0    1     0.0114
34 R4 11    0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0    7     0.0799
35 R4 21    0   1   0   0   0   0   0   0   0   0   0   0   0   0   0   0    9     0.1027
36 R4 32    2   1   0   0   0   0   0   1   0   0   0   0   0   1   0   0   15     0.1712


          * * * * SUMMARIES * * * *

  R      26  33  18  15  15  16  23  15  10  24  23  15  14  14  15  22
  B      19  13   8  15   5   3   8  13   7  12   9  12   7  11  19  23
  D      19  27  15  15  18  12  13  12  17  12  17  19  27  24  35  51
  E      51  59  44  22  24  32  24  23  29  39  52  59  59  43  66  87
  F      13  10  33  15  15  26  26  25  39  68  70  79  99  71  78  69
    1    20  20  37  14  11  31  23   9  22  23  23  21  14  15  19  24
    2    45  46  42  20  27  30  27  38  37  67  73  92 103  67  86  85
    3    24  29  13  21  14   7  10  13  17  29  38  38  48  46  51  63
    4    11  14   8  10   9   4  11  11  16   9  13  17  22  15  27  39
    5     2   0   0   2   1   1   0   1   0   2   1   1   5   6  14  19
    6     0   0   0   0   0   0   0   1   0   1   0   0   0   0   1   0

  R       6  12   7  13  14  10  18  11  19  27  30  21  25  23  17  38
  B       4  18  15  20  30  25  22  29  20  32  38  56  55  33  22  29
  D      35  39  70  58  94  79 104 100  99 123 124  79  89  59  32  53
  E      73  70  95  71  79  74  76  68  54  77  63  40  40  34  39  62
  F      37  37  30  26  30  17  27  22  27  20  27  11  11   2   5  36
    1    14  14  15  19  10  10  20  18  19  17  21   5   3  10   3  36
    2    56  51  59  43  57  32  48  45  43  40  51  31  48  32  30  64
    3    48  60  60  50  60  58  46  35  30  68  61  73  64  46  47  61
    4    25  31  67  47  79  74 100  90  75  95  79  67  72  37  18  19
    5     5   8   8  15  24  21  15  31  32  17  27  10   8   3   0   0
    6     1   0   1   1   3   0   0   0   1  15  13   0   0   0   0   0

  R      25  56  28  18  21  12  14  14   8  15  20  14  13  11  22  32
  B      17  42  36  39  16  13  12   6   5  14  17  13  17  14  16  20
  D      33  44  57  25  23  12  12   4   3  10   8   6  11   4  11  12
  E      30  22  30  22  15  11   6   5   4  11   6   4  15  10  16  20
  F       2   3   1  12   2   1   1   1   0   1   5   4   7   2   9  14
    1     6   9   6  29   7   4   8   4   4  10  11   8  25  14  26  28
    2    34  41  50  39  37  27  21  12   8  26  25  19  25  16  26  38
    3    28  39  44  25  10   6   1   0   0   0   0   0   0   0   0   0
    4    14  21  24   5   2   0   1   0   0   0   0   0   0   0   0   0
    5     0   1   0   0   0   0   0   0   0   0   0   0   0   0   0   0
    6     0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0

  R      28  12  15  23  31  16  22  39  37  43  38  41  40  37  50  44 1428    16.3014
  B       8  22  15   7  14   6  13   7   9   8   7  12   8  19  21  22 1127    12.8653
  D       9   7   3   6   6  11  18  11  15  28  24  25  37  35  38  44 2162    24.6804
  E      20  33  21  20  19  23  33  34  36  44  66  55  65  66  78  74 2642    30.1598
  F      26  28  29   8  40  10  20   5   1  13   6  16  10   8  12   3 1401    15.9932
    1    38  38  35  13  39  15  33  15   5  13  31  24  18  16  22   5 1119    12.7740
    2    24  50  33  27  32  23  40  38  32  37  40  42  50  43  47  47 2664    30.4110
    3     1   2   0   1   8   9   8   2  22  35  30  36  36  39  39  40 1789    20.4224
    4     0   0   0   0   0   3   3   2   2   8   2   6  13  27  38  46 1428    16.3014
    5     0   0   0   0   0   0   0   0   0   0   0   0   3   2   3   5  293     3.3447
    6     0   0   0   0   0   0   0   0   0   0   0   0   0   1   0   0   39     0.4452


          * * * * * BIN WINDROSE SUMMARY * * * * *

BIN                    DIRECTION
      1     2     3     4     5     6     7     8     9    10    11    12    13    14    15    16
 1 0.017 0.010 0.004 0.011 0.004 0.000 0.003 0.008 0.004 0.010 0.010 0.007 0.004 0.007 0.011 0.016
 2 0.017 0.014 0.012 0.017 0.005 0.007 0.014 0.017 0.010 0.012 0.005 0.017 0.010 0.014 0.026 0.029
 3 0.000 0.000 0.000 0.000 0.000 0.011 0.011 0.011 0.000 0.022 0.011 0.022 0.022 0.000 0.000 0.033
 4 0.017 0.021 0.013 0.010 0.011 0.010 0.010 0.008 0.006 0.008 0.008 0.011 0.011 0.021 0.019 0.013
 5 0.012 0.021 0.011 0.012 0.011 0.007 0.009 0.005 0.011 0.005 0.009 0.005 0.012 0.007 0.012 0.027
 6 0.004 0.005 0.003 0.004 0.008 0.003 0.003 0.005 0.011 0.004 0.008 0.011 0.013 0.011 0.013 0.015
 7 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.005 0.000 0.010 0.005 0.038 0.072
 8 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000
 9 0.023 0.029 0.031 0.004 0.008 0.037 0.012 0.006 0.016 0.010 0.019 0.014 0.004 0.008 0.010 0.010
10 0.028 0.030 0.021 0.011 0.014 0.011 0.012 0.014 0.010 0.014 0.017 0.025 0.022 0.015 0.022 0.030
11 0.010 0.013 0.008 0.009 0.005 0.004 0.004 0.005 0.009 0.022 0.027 0.030 0.031 0.023 0.034 0.040
12 0.008 0.011 0.003 0.006 0.006 0.000 0.008 0.006 0.011 0.008 0.011 0.008 0.028 0.017 0.034 0.056
13 0.018 0.012 0.041 0.022 0.014 0.024 0.031 0.010 0.027 0.031 0.025 0.024 0.020 0.022 0.027 0.025
14 0.005 0.004 0.016 0.005 0.009 0.019 0.012 0.027 0.029 0.057 0.060 0.077 0.096 0.055 0.067 0.056
15 0.000 0.007 0.000 0.000 0.007 0.000 0.007 0.000 0.022 0.058 0.080 0.066 0.117 0.139 0.095 0.102
16 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.000 0.250 0.250 0.000 0.250 0.000 0.250 0.000
17 0.018 0.023 0.013 0.011 0.011 0.011 0.016 0.011 0.007 0.017 0.016 0.011 0.010 0.010 0.011 0.015
18 0.018 0.023 0.013 0.011 0.011 0.011 0.016 0.011 0.007 0.017 0.016 0.011 0.010 0.010 0.011 0.015
19 0.018 0.023 0.013 0.011 0.011 0.011 0.016 0.011 0.007 0.017 0.016 0.011 0.010 0.010 0.011 0.015
20 0.018 0.023 0.013 0.011 0.011 0.011 0.016 0.011 0.007 0.017 0.016 0.011 0.010 0.010 0.011 0.015
21 0.018 0.023 0.013 0.011 0.011 0.011 0.016 0.011 0.007 0.017 0.016 0.011 0.010 0.010 0.011 0.015
22 0.018 0.023 0.013 0.011 0.011 0.011 0.016 0.011 0.007 0.017 0.016 0.011 0.010 0.010 0.011 0.015
```

23	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
24	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
25	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
26	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
27	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
28	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
29	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
30	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
31	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
32	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
33	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
34	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
35	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
36	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
37	0.015	0.016	0.013	0.009	0.009	0.010	0.011	0.010	0.012	0.018	0.020	0.021	0.024	0.019	0.024	0.029

	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	0.003	0.013	0.007	0.006	0.010	0.010	0.007	0.006	0.007	0.016	0.010	0.027	0.040	0.035	0.023	0.038
2	0.005	0.021	0.024	0.038	0.055	0.043	0.041	0.060	0.036	0.050	0.074	0.088	0.064	0.019	0.014	0.005
3	0.033	0.022	0.000	0.022	0.000	0.000	0.022	0.022	0.011	0.011	0.000	0.022	0.011	0.011	0.000	0.033
4	0.015	0.011	0.019	0.011	0.025	0.015	0.017	0.021	0.025	0.010	0.023	0.019	0.025	0.029	0.017	0.029
5	0.011	0.016	0.018	0.021	0.039	0.034	0.032	0.018	0.019	0.048	0.060	0.055	0.046	0.032	0.028	0.048
6	0.017	0.023	0.057	0.035	0.058	0.048	0.085	0.071	0.065	0.081	0.062	0.039	0.057	0.030	0.009	0.011
7	0.019	0.024	0.034	0.053	0.072	0.077	0.058	0.115	0.125	0.072	0.096	0.034	0.034	0.014	0.000	0.000
8	0.031	0.000	0.031	0.031	0.031	0.000	0.000	0.000	0.000	0.469	0.375	0.000	0.000	0.000	0.000	0.000
9	0.006	0.006	0.014	0.012	0.008	0.004	0.012	0.019	0.014	0.016	0.008	0.004	0.004	0.014	0.004	0.035
10	0.023	0.019	0.026	0.021	0.020	0.013	0.022	0.020	0.012	0.021	0.026	0.010	0.017	0.009	0.013	0.023
11	0.045	0.050	0.058	0.044	0.039	0.044	0.031	0.025	0.021	0.040	0.028	0.031	0.021	0.014	0.025	0.016
12	0.031	0.023	0.045	0.025	0.068	0.071	0.065	0.054	0.054	0.045	0.028	0.011	0.011	0.020	0.014	0.025
13	0.016	0.016	0.014	0.022	0.010	0.016	0.022	0.014	0.020	0.016	0.033	0.002	0.000	0.002	0.002	0.031
14	0.031	0.031	0.028	0.017	0.029	0.011	0.020	0.015	0.023	0.013	0.013	0.012	0.013	0.001	0.004	0.025
15	0.044	0.044	0.015	0.015	0.022	0.007	0.007	0.029	0.000	0.015	0.000	0.007	0.007	0.000	0.007	0.007
16	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
17	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
18	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
19	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
20	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
21	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
22	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
23	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
24	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
25	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
26	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
27	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
28	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
29	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
30	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
31	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
32	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
33	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
34	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
35	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
36	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
37	0.018	0.020	0.025	0.021	0.028	0.023	0.028	0.026	0.025	0.032	0.032	0.024	0.025	0.017	0.013	0.025

	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	0.024	0.048	0.045	0.049	0.021	0.018	0.017	0.008	0.007	0.020	0.024	0.018	0.024	0.020	0.023	0.028
2	0.000	0.019	0.010	0.010	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
3	0.000	0.022	0.000	0.044	0.033	0.022	0.033	0.011	0.000	0.022	0.033	0.022	0.044	0.022	0.044	0.056
4	0.019	0.044	0.040	0.031	0.029	0.013	0.015	0.006	0.006	0.015	0.010	0.008	0.013	0.004	0.013	0.013
5	0.027	0.016	0.032	0.007	0.007	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
6	0.011	0.012	0.024	0.001	0.001	0.000	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
7	0.000	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
8	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
9	0.012	0.008	0.010	0.023	0.006	0.004	0.006	0.006	0.008	0.012	0.008	0.004	0.027	0.019	0.029	0.025
10	0.011	0.011	0.017	0.006	0.009	0.008	0.003	0.002	0.000	0.005	0.002	0.002	0.002	0.001	0.002	0.008
11	0.009	0.004	0.008	0.006	0.004	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
12	0.017	0.011	0.006	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
13	0.000	0.006	0.002	0.018	0.002	0.000	0.002	0.000	0.000	0.002	0.008	0.002	0.014	0.004	0.016	0.022
14	0.001	0.000	0.000	0.003	0.001	0.001	0.000	0.001	0.000	0.000	0.001	0.004	0.000	0.000	0.001	0.004
15	0.007	0.000	0.000	0.007	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
16	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
17	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
18	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
19	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
20	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
21	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
22	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
23	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
24	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
25	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
26	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
27	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
28	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
29	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
30	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
31	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
32	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
33	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
34	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
35	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
36	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
37	0.012	0.019	0.017	0.013	0.009	0.006	0.005	0.003	0.002	0.006	0.006	0.005	0.007	0.005	0.008	0.011

	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	TOTAL
1	0.011	0.031	0.021	0.010	0.020	0.007	0.017	0.010	0.013	0.010	0.010	0.014	0.008	0.016	0.008	0.016	1.000000
2	0.000	0.000	0.000	0.000	0.000	0.002	0.002	0.000	0.000	0.002	0.000	0.005	0.005	0.019	0.036	0.026	1.000000
3	0.000	0.044	0.022	0.011	0.011	0.011	0.033	0.011	0.000	0.011	0.022	0.011	0.022	0.000	0.011	0.000	1.000000
4	0.017	0.006	0.002	0.010	0.010	0.011	0.017	0.015	0.011	0.011	0.017	0.019	0.019	0.015	0.013	0.019	1.000000
5	0.000	0.000	0.000	0.000	0.000	0.005	0.009	0.000	0.016	0.027	0.021	0.023	0.028	0.018	0.028	0.030	1.000000
6	0.000	0.000	0.000	0.000	0.000	0.001	0.001	0.003	0.000	0.008	0.001	0.001	0.009	0.020	0.015	0.020	1.000000
7	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.010	0.005	0.014	0.010	1.000000
8	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.031	0.000	0.000	1.000000
9	0.031	0.027	0.021	0.019	0.014	0.016	0.031	0.023	0.010	0.016	0.041	0.023	0.027	0.027	0.031	0.010	1.000000
10	0.005	0.017	0.011	0.011	0.010	0.012	0.016	0.021	0.020	0.020	0.029	0.025	0.031	0.028	0.031	0.033	1.000000
11	0.000	0.003	0.000	0.000	0.003	0.003	0.001	0.001	0.010	0.018	0.019	0.019	0.019	0.025	0.025	0.016	1.000000
12	0.000	0.000	0.000	0.000	0.000	0.003	0.003	0.000	0.006	0.003	0.003	0.008	0.014	0.014	0.034	0.065	1.000000
13	0.045	0.035	0.043	0.006	0.061	0.012	0.025	0.004	0.000	0.008	0.012	0.024	0.006	0.006	0.012	0.000	1.000001
14	0.004	0.013	0.009	0.007	0.011	0.005	0.009	0.004	0.001	0.009	0.000	0.005	0.007	0.005	0.007	0.003	1.000000
15	0.000	0.000	0.000	0.000	0.007	0.000	0.000	0.000	0.000	0.015	0.000	0.000	0.015	0.007	0.007	0.007	1.000000
16	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	1.000000
17	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000

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 18 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 19 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 20 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 21 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 22 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 23 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 24 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 25 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 26 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 27 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 28 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 29 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 30 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 31 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 32 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 33 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 34 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 35 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 36 0.020 0.008 0.011 0.016 0.022 0.011 0.015 0.027 0.026 0.030 0.027 0.029 0.028 0.026 0.035 0.031 1.000000
 37 0.010 0.012 0.009 0.007 0.013 0.008 0.012 0.011 0.011 0.016 0.016 0.017 0.018 0.019 0.023 0.021 1.000000


USER INPUT IS READ FROM UNIT 25
RECORD IDENTIFIER FIELDS  11 CHARACTERS LONG ARE EXPECTED.
THE FIRST 499 COLUMNS OF EACH INPUT RECORD ARE PROCESSED.

RECORD
NUMBER                    RECORD

      * File created using WinMACCS version 3.7.0 11/13/2012 4:58:02 PM
      *
      * DCF_FILE_TH - Identifies the DCF file to be used for the MACCS calculation
   1  DCF_FILE001     'C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Data\FGR13GyEquivDCF.INP'
      *
      * EANAM1 - Identifies the EARLY calculation
   2  MIEANAM1001     'OCP3 low density no spray, EARLY input'
      *
      * ENDAT2 - control flag allowing execution of ATMOS and EARLY without CHRONC
   3  MIENDAT2001     .FALSE.
      *
      * IPLUME - dispersion code option
   4  MIIPLUME001     3
      *
      * Form 'Grid Subdivisions' Comment:
      * Value used in NUREG-1150.
      *
      * NUMFIN - number of fine-grid subdivisions used by model
   5  MINUMFIN001     7
      *
      * IPRINT - amount of output desired
   6  MIIPRINT001     0
      *
      * POPFLG - is population uniform or defined by Site Data File.
   7  PDPOPFLG001     FILE
      *
      * ORGNAM_FGR13, ORGFLG_FGR13 - list of organs to be included in the calculations using FGR13 DCF file
   8  MIORGDEF001     A-SKIN  .TRUE.
   9  MIORGDEF002     'A-RED MARR'    .TRUE.
  10  MIORGDEF003     A-LUNGS .TRUE.
  11  MIORGDEF004     A-THYROID       .TRUE.
  12  MIORGDEF005     A-STOMACH       .TRUE.
  13  MIORGDEF006     'A-LOWER LI'    .TRUE.
  14  MIORGDEF007     L-ICRP60ED      .TRUE.
  15  MIORGDEF008     'L-RED MARR'    .TRUE.
  16  MIORGDEF009     'L-BONE SUR'    .TRUE.
  17  MIORGDEF010     L-BREAST        .TRUE.
  18  MIORGDEF011     L-LUNGS .TRUE.
  19  MIORGDEF012     L-THYROID       .TRUE.
  20  MIORGDEF013     'L-LOWER LI'    .TRUE.
  21  MIORGDEF014     'L-BLAD WAL'    .TRUE.
  22  MIORGDEF015     L-LIVER .TRUE.
      *
      * RISCAT - Output relative contribution of each weather category bins
  23  MIRISCAT001     .FALSE.
      *
      * OVRRID - Flag indicating if Wind Rose defaults from ATMOS are to be overridden
  24  MIOVRRID001     .FALSE.
      *
      * Form 'Shielding and Exposure' Comment:
      * Data taken directly from NUREG-1150 for Peach Bottom.
      *
      * CSFACT - Cloudshine shielding factor
  25  SECSFACT001     1.
  26  SECSFACT002     0.6
  27  SECSFACT003     0.5
      *
      * PROTIN - Inhalation protection factor
  28  SEPROTIN001     0.98
  29  SEPROTIN002     0.46
  30  SEPROTIN003     0.33
      *
      * BRRATE - Breathing rates
  31  SEBRRATE001     2.66E-04
  32  SEBRRATE002     2.66E-04
  33  SEBRRATE003     2.66E-04
      *
      * SKPFAC - skin protection factors
  34  SESKPFAC001     0.98
  35  SESKPFAC002     0.46
  36  SESKPFAC003     0.33
      *
      * GSHFAC - groundshine shielding factors
  37  SEGSHFAC001     0.5
  38  SEGSHFAC002     0.18
  39  SEGSHFAC003     0.1
      *
      * Form 'Emergency Phase Resuspension' Comment:
      * Values from NUREG-1150.
      *
```

* RESCON - Initial value for emergency-phase resuspension concentration factor.
40 SERESCON001 1.E-04
*
* RESHAF - Emergency-phase resuspension concentration coefficient weathering half-life.
41 SERESHAF001 1.82000E+05
*
* Form 'Basic Parameters' Comment:
* Shadow (0-20)
*
* EANAM2 - Name of emergency response cohort
42 EZEANAM2001 'Group 1'
*
* WTNAME - type of weighting factor to be used in generating weighted sum of results
43 EZWTNAME001 PEOPLE
*
* WTFRAC - weighting fraction applied to results of emergency response cohort
44 EZWTFRAC001 0.3
*
* EVATYP - decides on radial or network evacuation option.
45 EZEVATYP001 NETWORK
*
* TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
46 TRAVELPOINT CENTERPOINT
*
* ESPEED - evacuee travel speed during the three phases of evacuation
47 EZESPEED001 8.941
48 EZESPEED002 6.706
49 EZESPEED003 2.235
*
* ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
50 EZESPMUL001 0.7
51 EZESPMUL002 0.7
52 EZESPMUL003 0.7
*
* REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
53 EZREFPNT001 ALARM
*
* DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
54 EZDURBEG001 3600.
*
* DURMID - duration of middle phase of evacuation, in seconds.
55 EZDURMID001 1800.
*
* NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
56 EZNUMEVA001 15
*
* DLTSHL - delay from reference time point to when individual takes shelter. DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
57 EZDLTSHL001 0.
58 EZDLTSHL002 0.
59 EZDLTSHL003 0.
60 EZDLTSHL004 0.
61 EZDLTSHL005 0.
62 EZDLTSHL006 0.
63 EZDLTSHL007 0.
64 EZDLTSHL008 0.
65 EZDLTSHL009 0.
66 EZDLTSHL010 0.
67 EZDLTSHL011 0.
68 EZDLTSHL012 0.
69 EZDLTSHL013 7200.
70 EZDLTSHL014 7200.
71 EZDLTSHL015 7200.
*
* DLTEVA -Delay time to begin evacuation
72 EZDLTEVA001 0.
73 EZDLTEVA002 0.
74 EZDLTEVA003 0.
75 EZDLTEVA004 0.
76 EZDLTEVA005 0.
77 EZDLTEVA006 0.
78 EZDLTEVA007 0.
79 EZDLTEVA008 0.
80 EZDLTEVA009 0.
81 EZDLTEVA010 0.
82 EZDLTEVA011 0.
83 EZDLTEVA012 0.
84 EZDLTEVA013 3600.
85 EZDLTEVA014 3600.
86 EZDLTEVA015 3600.
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
87 EZESPGRD001 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
88 EZESPGRD002 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
89 EZESPGRD003 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
90 EZESPGRD004 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
91 EZESPGRD005 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
92 EZESPGRD006 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
93 EZESPGRD007 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
94 EZESPGRD008 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
95 EZESPGRD009 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
96 EZESPGRD010 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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97 EZESPGRD011 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
98 EZESPGRD012 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
99 EZESPGRD013 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75 1.
1. 1. 1. 1. 1. .75 .75 .75
100 EZESPGRD014 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75

101 EZESPGRD015 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75
102 EZESPGRD016 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
103 EZESPGRD017 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
104 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
105 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
106 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 4 1 1
107 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2
1 4 4 4 4 4 1 1
108 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2
1 1 1 1 1 1 1 1
109 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4
4 4 4 1 1 1 1 1
110 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 4 4 1 4 4 4 4
4 4 1 1 1 1 1 1
111 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 4 1 4 4 4 2
2 1 1 1 1 1 2 2
112 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 4 4 1 4 2 1 4
1 4 1 1 1 2 2 2
113 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 4 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1
4 4 4 2 2 1 1 1
114 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 4 4 2 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4
4 4 2 2 1 4 2 2
115 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 2 1 2 1 2 2 1 4 4 4 4 2 2 2 1 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4
2 1 1 2 2 1 4 2
116 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 4 4 4 1 2 1 4 4 4 4 2 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4
2 1 4 2 1 4 1 1
117 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4
1 4 1 1 1 1 1 1
118 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2
1 1 2 2 2 1 2 1
119 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
120 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
*
* LASMOV - The outermost spatial interval of the evacuation movement zone.
121 EZLASMOV001 17
*
* Form 'Duration of Early Phase' Comment:
* 1 week.
*
* ENDEMP - duration of the emergency-phase period, seconds
122 SRENDEMP001 6.04800E+05
*
* CRIORG - critical organ for relocation decisions during emergency-phase period
123 SRCRIORG001 L-ICRP60ED
*
* Form 'Hot Spot Relocation' Comment:
* Randy Sullivan recommended these values. (4 hours for 10 mile evac)
*
* TIMHOT - hot-spot relocation action time, seconds after plume arrival
124 SRTIMHOT001 14400.
*
* Form 'Normal Relocation' Comment:
* Randy Sullivan recommended these values. (8 hours for 10 mile evac)
*
*
* TIMNRM - Normal Relocation Time (Seconds from Plume Arrival)
125 SRTIMNRM001 28800.
*
* DOSHOT - Hot-Spot Relocation Dose Threshold (Sieverts)
126 SRDOSHOT001 0.05
*
* DOSNRM - Normal Relocation Dose Threshold (Sieverts)
127 SRDOSNRM001 0.01
*
* NUMEFA - Number of Early Fatality Effects
128 EFNUMEFA001 3
*
* ORGNAM2, EFFACA, EFFACB, EFFTHR Early Fatality Effects - target organ, alpha factor and beta factor for hazard function, and threshold dose (Sieverts)
129 EFATAGRP001 'A-RED MARR' 5.6 6.1 2.32
130 EFATAGRP002 A-LUNGS 23.5 9.6 13.6
131 EFATAGRP003 A-STOMACH 12.1 9.3 6.5
*
* NUMEIN - Number of Early Injury Effects
132 EINUMEIN001 7
*
* ORGNAM3, EINAME, EISUSC, EITHRE, EIFACA, EIFACB Early Injury Effects - name, target organ, affected population fract, threshold dose, alpha factor, beta factor.
133 EINJUGRP001 'PRODROMAL VOMIT' A-STOMACH 1. 0.5 2. 3.
134 EINJUGRP002 DIARRHEA A-STOMACH 1. 1. 3. 2.5
135 EINJUGRP003 PNEUMONITIS A-LUNGS 1. 9.2 16.6 7.3
136 EINJUGRP004 'SKIN ERYTHEMA' A-SKIN 1. 3. 6. 5.
137 EINJUGRP005 TRANSEPIDERMAL A-SKIN 1. 10. 20. 5.
138 EINJUGRP006 THYROIDITIS A-THYROID 1. 40. 240. 2.
139 EINJUGRP007 HYPOTHYROIDISM A-THYROID 1. 2. 60. 1.3
*
* Form 'Latent Cancer Parameters' Comment:
* Risk factors are those recommended by Keith Eckerman to use with a FGR-13 DCF file set modified as follows:
* Red marrow DCFs have been modified to use a RBE of 1 for alpha radiation; breast DCFs have been modified to use an RBE of 10 for alpha radiation.
* As a kluge, the organ named bladder wall contains data for the pancreas, which is used as a surrogate for the soft tissue for the purpose of evaluating residual cancers.
*
* NUMACA - number of latent cancer effects
140 LCNUMACA001 8
*
* ACTHRE - dose threshold for linear dose response, Sieverts
141 LCACTHRE001 0.E+00
*
* DDTHRE - dose threshold for applying dose-dependent reduction factor, DDREFA
142 LCDDTHRE001 .2
*
* ACNAME, ORGNAM4, ACSUSC, DOSEFA, DOSEFB, CFRISK, CIRISK, DDREFA - Latent Cancer Effects Parameters

```
 143 LCANCERS001    LEUKEMIA     'L-RED MARR'   1.    1.    0.    0.0111  0.0113  2.
 144 LCANCERS002    BONE   'L-BONE SUR'   1.    1.    0.    1.9E-04 2.71E-04      2.
 145 LCANCERS003    BREAST  L-BREAST      1.    1.    0.    0.00506 0.0101  1.
 146 LCANCERS004    LUNG   L-LUNGS 1.    1.    0.    0.0198  0.0208  2.
 147 LCANCERS005    THYROID L-THYROID     1.    1.    0.    6.48E-04      0.00648 2.
 148 LCANCERS006    LIVER   L-LIVER 1.    1.    0.    0.003   0.00316 2.
 149 LCANCERS007    COLON   'L-LOWER LI'   1.    1.    0.    0.0208  0.0378  2.
 150 LCANCERS008    RESIDUAL      'L-BLAD WAL'   1.    1.    0.    0.0493  0.169   2.
    *
    * NUM1=0
 151 TYPE1NUMBER     0
    *
    * NUM1 - Number of results of type 1
 152 TYPE1NUMBER     38
********   RECORD NUMBER   152 REPLACES RECORD NUMBER   151   ********
    *
    * NAME1, I1DIS1, I2DIS1, CCDF1 - Health-Effect Cases
 153 TYPE1OUT001     'ERL FAT/TOTAL' 1      12      REPORT
 154 TYPE1OUT002     'ERL FAT/TOTAL' 1      19      REPORT
 155 TYPE1OUT003     'ERL FAT/TOTAL' 1      26      REPORT
 156 TYPE1OUT004     'CAN INJ/TOTAL' 1      12      REPORT
 157 TYPE1OUT005     'CAN INJ/TOTAL' 1      15      REPORT
 158 TYPE1OUT006     'CAN INJ/TOTAL' 1      17      REPORT
 159 TYPE1OUT007     'CAN INJ/TOTAL' 1      18      REPORT
 160 TYPE1OUT008     'CAN INJ/TOTAL' 1      19      REPORT
 161 TYPE1OUT009     'CAN INJ/TOTAL' 1      21      REPORT
 162 TYPE1OUT010     'CAN INJ/TOTAL' 1      23      REPORT
 163 TYPE1OUT011     'CAN INJ/TOTAL' 1      25      REPORT
 164 TYPE1OUT012     'CAN INJ/TOTAL' 1      26      REPORT
 165 TYPE1OUT013     'CAN FAT/TOTAL' 1      12      REPORT
 166 TYPE1OUT014     'CAN FAT/TOTAL' 1      15      REPORT
 167 TYPE1OUT015     'CAN FAT/TOTAL' 1      17      REPORT
 168 TYPE1OUT016     'CAN FAT/TOTAL' 1      18      REPORT
 169 TYPE1OUT017     'CAN FAT/TOTAL' 1      19      REPORT
 170 TYPE1OUT018     'CAN FAT/TOTAL' 1      21      REPORT
 171 TYPE1OUT019     'CAN FAT/TOTAL' 1      23      REPORT
 172 TYPE1OUT020     'CAN FAT/TOTAL' 1      25      REPORT
 173 TYPE1OUT021     'CAN FAT/TOTAL' 1      26      REPORT
 174 TYPE1OUT022     'CAN FAT/THYROID'     1      12      REPORT
 175 TYPE1OUT023     'CAN FAT/THYROID'     1      19      REPORT
 176 TYPE1OUT024     'CAN FAT/THYROID'     1      21      REPORT
 177 TYPE1OUT025     'CAN FAT/THYROID'     1      26      REPORT
 178 TYPE1OUT026     'CAN FAT/BREAST'      1      12      REPORT
 179 TYPE1OUT027     'CAN FAT/BREAST'      1      19      REPORT
 180 TYPE1OUT028     'CAN FAT/BREAST'      1      21      REPORT
 181 TYPE1OUT029     'CAN FAT/BREAST'      1      26      REPORT
 182 TYPE1OUT030     'CAN FAT/LUNG'  1      12      REPORT
 183 TYPE1OUT031     'CAN FAT/LUNG'  1      19      REPORT
 184 TYPE1OUT032     'CAN FAT/LUNG'  1      21      REPORT
 185 TYPE1OUT033     'CAN FAT/LUNG'  1      26      REPORT
 186 TYPE1OUT034     'CAN FAT/LEUKEMIA'    1      26      REPORT
 187 TYPE1OUT035     'CAN FAT/BONE'  1      26      REPORT
 188 TYPE1OUT036     'CAN FAT/LIVER' 1      26      REPORT
 189 TYPE1OUT037     'CAN FAT/COLON' 1      26      REPORT
 190 TYPE1OUT038     'CAN FAT/RESIDUAL'    1      26      REPORT
    *
    * NUM2=0
 191 TYPE2NUMBER     0
    *
    * NUM2 - Number of results of type 2
 192 TYPE2NUMBER     1
********   RECORD NUMBER   192 REPLACES RECORD NUMBER   191   ********
    *
    * RISTHR, CCDF2  - Early-Fatality Radius
 193 TYPE2OUT001     0.      NONE
    *
    * NUM3=0
 194 TYPE3NUMBER     0
    *
    * NUM3 - Number of results of type 3
 195 TYPE3NUMBER     3
********   RECORD NUMBER   195 REPLACES RECORD NUMBER   194   ********
    *
    * NAME3, DOSTH3, CCDF3 - Population Exceeding a Dose Threshold
 196 TYPE3OUT001     'A-RED MARR'   2.32    NONE
 197 TYPE3OUT002     A-LUNGS 13.6    NONE
 198 TYPE3OUT003     A-STOMACH      6.5     NONE
    *
    * NUM4=0
 199 TYPE4NUMBER     0
    *
    * NUM5 =0
 200 TYPE5NUMBER     0
    *
    * NUM5 - Number of results of type 5
 201 TYPE5NUMBER     4
********   RECORD NUMBER   201 REPLACES RECORD NUMBER   200   ********
    *
    * NAME5, I1DIS5, CCDF5 - Population Dose
 202 TYPE5OUT001     L-ICRP60ED     1      12      REPORT
 203 TYPE5OUT002     L-ICRP60ED     1      19      REPORT
 204 TYPE5OUT003     L-ICRP60ED     1      21      REPORT
 205 TYPE5OUT004     L-ICRP60ED     1      26      REPORT
    *
    * NUM6 =0
 206 TYPE6NUMBER     0
    *
    * NUM7=0
 207 TYPE7NUMBER     0
    *
    * NUM8=0
 208 TYPE8NUMBER     0
    *
    * NUM8 - Number of results of type 8
 209 TYPE8NUMBER     17
********   RECORD NUMBER   209 REPLACES RECORD NUMBER   208   ********
    *
    * NAME8, I1DIS8, I2DIS8, CCDF8 - Population-Weighted Risk
 210 TYPE8OUT001     'CAN FAT/TOTAL' 1      12      REPORT
```

```
  211  TYPE8OUT002      'CAN FAT/TOTAL' 1       15      REPORT
  212  TYPE8OUT003      'CAN FAT/TOTAL' 1       17      REPORT
  213  TYPE8OUT004      'CAN FAT/TOTAL' 1       18      REPORT
  214  TYPE8OUT005      'CAN FAT/TOTAL' 1       19      REPORT
  215  TYPE8OUT006      'CAN FAT/TOTAL' 1       21      REPORT
  216  TYPE8OUT007      'CAN FAT/TOTAL' 1       23      REPORT
  217  TYPE8OUT008      'CAN FAT/TOTAL' 1       25      REPORT
  218  TYPE8OUT009      'CAN FAT/TOTAL' 1       26      REPORT
  219  TYPE8OUT010      'CAN FAT/TOTAL' 13      15      REPORT
  220  TYPE8OUT011      'CAN FAT/TOTAL' 16      17      REPORT
  221  TYPE8OUT012      'CAN FAT/TOTAL' 18      18      REPORT
  222  TYPE8OUT013      'CAN FAT/TOTAL' 19      19      REPORT
  223  TYPE8OUT014      'CAN FAT/TOTAL' 20      21      REPORT
  224  TYPE8OUT015      'CAN FAT/TOTAL' 22      23      REPORT
  225  TYPE8OUT016      'CAN FAT/TOTAL' 24      25      REPORT
  226  TYPE8OUT017      'CAN FAT/TOTAL' 26      26      REPORT
       *
       * NUMA=0
  227  TYPEANUMBER      0
       *
       * NUMA - Number of results of type A
  228  TYPEANUMBER      1
********   RECORD NUMBER   228 REPLACES RECORD NUMBER   227   ********
       *
       * NAMEA, I1DISA, I2DISA, CCDFA - Peak Dose vs Distance
  229  TYPEAOUT001      L-ICRP60ED      1       26      REPORT
       *
       * NUMB =0
  230  TYPEBNUMBER      0
       *
       * NUMC=0
  231  TYPECNUMBER      0
       *
       * Form 'Land Area Exceeding Dose' Comment:
       * Emergency Phase PAGs
       *
       * NUMC number of typeC output
  232  TYPECNUMBER      3
********   RECORD NUMBER   232 REPLACES RECORD NUMBER   231   ********
       *
       * ORGNAM8, ELEVDOSE, PRINT_FLAG_C - organs for typeC output
  233  TYPECOUT001      L-ICRP60ED      0.01    .FALSE.
  234  TYPECOUT002      L-ICRP60ED      0.05    .FALSE.
  235  TYPECOUT003      A-THYROID       0.05    .FALSE.
       *
       * NUMD = 0
  236  TYPEDNUMBER      0
       *
       * NUMD number of typeD output
  237  TYPEDNUMBER      16
********   RECORD NUMBER   237 REPLACES RECORD NUMBER   236   ********
       *
       * I1DISD, NUCLIDED, ELEVCONC, PRINT_FLAG_D
  238  TYPEDOUT001      12      Cs-137  37000.  .FALSE.
  239  TYPEDOUT002      12      Cs-137  1.85000E+05     .FALSE.
  240  TYPEDOUT003      12      Cs-137  5.55000E+05     .FALSE.
  241  TYPEDOUT004      12      Cs-137  1.480000E+06    .FALSE.
  242  TYPEDOUT005      19      Cs-137  37000.  .FALSE.
  243  TYPEDOUT006      19      Cs-137  1.85000E+05     .FALSE.
  244  TYPEDOUT007      19      Cs-137  5.55000E+05     .FALSE.
  245  TYPEDOUT008      19      Cs-137  1.480000E+06    .FALSE.
  246  TYPEDOUT009      21      Cs-137  37000.  .FALSE.
  247  TYPEDOUT010      21      Cs-137  1.85000E+05     .FALSE.
  248  TYPEDOUT011      21      Cs-137  5.55000E+05     .FALSE.
  249  TYPEDOUT012      21      Cs-137  1.480000E+06    .FALSE.
  250  TYPEDOUT013      25      Cs-137  37000.  .FALSE.
  251  TYPEDOUT014      25      Cs-137  1.85000E+05     .FALSE.
  252  TYPEDOUT015      25      Cs-137  5.55000E+05     .FALSE.
  253  TYPEDOUT016      25      Cs-137  1.480000E+06    .FALSE.
       *
       * DOSMOD, dose model, LNT, AT or PL
  254  LCDOSMOD001      LNT
       *
       * Form 'Annual Threshold' Comment:
       * Threshold values are from Health Physics Society position statement PS010-1 (August 2004).
       *
       * DTHNUM, Number of annual dose threshold values
  255  LCDTHNUM001      1
       *
       * DTHANN, Annual threshold values
  256  LCDTHANN001      1.E-04
       *
       * DTHLIF, Lifetime dose restriction
  257  LCDTHLIF001      10000.
       *
       * KIMODL, KI model
  258  EZKIMODL001      KI
       *
       * EFFACY_TH, KI Ingestion
  259  EZEFFACY001      0.7
       *
       * POPFRAC_TH, KI Ingestion, SLT
  260  EZPOPFRC001      1.
       *
       * FRACLD_FILE - popflg=FILE, dummy variable
  261  STFRACLD001      1.0
       *
       * NUME=0
  262  TYPEENUMBER      0
       .
******** TERMINATOR RECORD ENCOUNTERED -- END OF BASE CASE USER INPUT ********


USER INPUT PROCESSING SUMMARY - BASE CASE

NUMBER OF RECORDS READ                          =  465
NUMBER OF BLANK OR COMMENT RECORDS READ         =  202
NUMBER OF TERMINATOR RECORDS                    =    1
NUMBER OF RECORDS PROCESSED                     =  262
```

NUMBER OF PROCESSED RECORDS DUPLICATED = 8
NUMBER OF PROCESSED RECORDS SORTED = 254
****

THE KI MODEL IS IN EFFECT
READING DCF FILE:C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Data\FGR13GyEquivDCF.INP
DCF FILE is of type :FGR13DF
Am using a FGR13DCF dose factor file

The list of defined organs is as follows (A- is ACUTE and L- is LIFETIME):
A-SKIN
A-RED MARR
A-LUNGS
A-THYROID
A-STOMACH
A-LOWER LI
L-ICRP60ED
L-RED MARR
L-BONE SUR
L-BREAST
L-LUNGS
L-THYROID
L-LOWER LI
L-BLAD WAL
L-LIVER

READING FROM A DOSE CONVERSION FILE WITH THE FOLLOWING HEADER:
FGR13DF 5/13/2008 12:23:56 Version 1.03, Gy-Equivalent DCFs
Internal Dose Coefficients derived from FGR 13, EPA 402-R-99-001

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	4	4	2	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	2	1	2	1	2	2	1	4	4	4
13	1	1	1	1	1	2	1	4	4	4	1	2	1	4	4	4
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1
4	4	2	2	1	1	1	1	4	4	4	2	2	1	4	4	2
5	1	1	1	1	2	1	4	4	2	2	1	4	4	2	2	1
6	2	2	1	1	1	1	2	1	1	4	2	2	1	2	1	4
7	1	1	1	1	4	4	2	2	1	1	4	4	1	1	4	1
8	2	1	4	4	2	1	2	1	2	1	4	2	1	4	2	1
9	1	2	1	4	2	1	2	1	2	1	1	4	1	4	1	4
10	2	2	2	2	1	2	1	4	4	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	4	2	2	2	1	2	1	1	1	1	4	1	1	1	1	4
13	4	2	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1	1
4	1	4	4	2	1	1	4	2	1	4	4	4	4	4	1	1
5	4	4	2	2	1	4	4	2	1	1	1	1	1	1	1	1
6	4	2	2	1	4	4	4	4	4	4	4	1	1	1	1	1
7	4	4	4	1	4	4	4	4	4	4	1	1	1	1	1	1
8	1	4	4	1	4	4	4	2	2	1	1	1	1	1	2	2
9	4	4	4	1	4	2	1	4	1	4	1	1	1	2	2	2
10	4	2	2	1	1	1	1	1	4	4	4	2	2	1	1	1
11	4	4	4	2	2	2	1	4	4	4	2	2	1	4	2	2
12	4	4	2	2	2	2	1	4	2	1	1	2	2	1	4	2
13	4	4	2	2	2	1	1	4	2	1	4	2	1	4	1	1
14	1	4	4	2	2	1	1	4	1	4	1	1	1	1	1	1

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 16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1


USING THE FOLLOWING SITE DATA FILE:

 SECPOP2000 Version: 3.13.1 MACCS2 Formatted Site:  Peach Bottom Census: C:\Program Files\SecPOP_2000\Census\CENSUS00.DAT County: C:\Program Files\SecPOP_2000\Census\COUNTY2002RA.DAT* Created from P:\SFP Comparative Study\MACCS2 runs\SFPSS February\SFPSS base
seismic\Data\PBsite2011_64cp_26r rev1.inp using PopMod 1.0.4 3/14/2012 2:33:26 PM
 Lat: 39d45'32" Long:  76d16' 9" Population multiplier: 1.1051 Economic multiplier: 1.2500 Run Time: 3/12/2012 12:54:07 PM
  26 SPATIAL INTERVALS
  64 WIND DIRECTIONS
   7 CROP CATEGORIES
   4 WATER PATHWAY ISOTOPES
   1 WATERSHEDS
  97 ECONOMIC REGIONS
 SPATIAL DISTANCES     KILOMETERS
    0.1600    0.5200    1.2100    1.6100    2.1300    3.2200    4.0200    4.8300
    5.6300    8.0500   11.2700   16.0900   20.9200   25.7500   32.1900   40.2300
   48.2800   64.3700   80.4700  112.6500  160.9300  241.1400  321.8700  563.2700
  804.6700 1609.3400
 POPULATION
       0.       0.       0.       0.       0.       0.       0.      11.
       0.      78.      96.     462.    1022.    2382.   21591.   15081.
   14023.    8126.   15295.   24411.   56436.   61039.  151087.  323046.
    6284.       0.
       0.       0.       0.       0.       0.       0.       0.      11.
       0.      78.      96.     462.    1022.    2382.   21591.   15081.
   14023.    8126.   15295.   24411.   56436.   61039.  151087.  323046.
    6284.       0.
       0.       0.       0.       0.       0.       0.       3.       5.
       8.      68.     124.     573.    1396.    1791.   12148.    9216.
    9785.   11035.   36798.   26532.   58675.   94682.   99058.  360987.
   66812.       9.
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      16.      58.     152.     683.    1769.    1200.    2705.    3351.
    5547.   13945.   58301.   28654.   60915.  128324.   47030.  398928.
  127340.      17.
       0.       0.       0.       0.       0.       0.       7.       0.
      16.      58.     152.     683.    1769.    1200.    2705.    3351.
    5547.   13945.   58301.   28654.   60915.  128324.   47030.  398928.
  127340.      17.
       0.       0.       0.       0.       0.       0.       7.       0.
      16.      58.     152.     683.    1769.    1200.    2705.    3351.
    5547.   13945.   58301.   28654.   60915.  128324.   47030.  398928.
  127340.      17.
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  375600.   78169.
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       0.     111.     126.     262.     445.     692.    1367.    4120.
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REGION INDEX
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CROP SEASON AND SHARE
1 PASTURE 90. 270. 0.4100
2 STORED FORAGE 150. 240. 0.1300
3 GRAINS 150. 240. 0.2100
4 GRN LEAFY VEGETABLES 150. 240. 0.0020
5 OTHER FOOD CROPS 150. 240. 0.0040
6 LEGUMES AND SEEDS 150. 240. 0.1500
7 ROOTS AND TUBERS 150. 240. 0.0030

WATERSHED DEFINITION -- INITIAL AND ANNUAL WASHOFF AND INGESTION FACTORS

1	Sr-89	5.00E-06	0.0
2	Sr-90	5.00E-06	0.0
3	Cs-134	5.00E-06	0.0
4	Cs-137	5.00E-06	0.0

REGIONAL ECONOMIC DATA

1	EXCLUSION	.493	.164	1597.5	14841.3	267452.5
2	REGION_02	.493	.164	1597.5	14841.3	267452.5
3	REGION_03	.678	.333	5987.5	24571.3	269488.8
4	REGION_04	.678	.333	5987.5	24571.3	269488.8
5	REGION_05	.327	.186	2228.8	12328.8	244522.4
6	REGION_06	.322	.627	1031.5	4685.6	221426.7
7	REGION_07	.000	.000	0.0	0.0	0.0
8	REGION_08	.000	.000	0.0	0.0	0.0
9	REGION_09	.678	.333	5987.5	24571.3	269488.8
10	REGION_10	.678	.333	5987.5	24571.3	269488.8
11	REGION_11	.276	.142	2403.4	15362.4	263393.6
12	REGION_12	.183	.521	952.4	5380.4	232615.4
13	REGION_13	.000	.000	0.0	0.0	0.0
14	REGION_14	.678	.333	5987.5	24571.3	269488.8
15	REGION_15	.678	.333	5987.5	24571.3	269488.8
16	REGION_16	.602	.286	6202.0	26278.3	308768.9
17	REGION_17	.314	.127	2566.4	27195.2	383801.6
18	REGION_18	.091	.237	1409.4	12681.8	270060.0
19	REGION_19	.000	.000	0.0	0.0	0.0
20	REGION_20	.678	.333	5987.5	24571.3	269488.8
21	REGION_21	.678	.333	5987.5	24571.3	269488.8
22	REGION_22	.443	.186	6650.8	29850.9	390973.1
23	REGION_23	.191	.070	2708.6	42103.8	397659.4
24	REGION_24	.086	.046	9398.7	49875.9	357410.7
25	REGION_25	.000	.000	0.0	0.0	0.0
26	REGION_26	.678	.333	5987.5	24571.3	269488.8
27	REGION_27	.678	.333	5987.5	24571.3	269488.7
28	REGION_28	.382	.132	6302.6	29463.6	401859.2
29	REGION_29	.196	.040	3402.4	26358.8	318103.1
30	REGION_30	.033	.000	2801.1	44149.3	300457.7
31	REGION_31	.000	.000	0.0	0.0	0.0
32	REGION_32	.000	.000	0.0	0.0	0.0
33	REGION_33	.618	.273	5403.2	23369.9	270489.2
34	REGION_34	.350	.008	2915.4	18419.2	279887.2
35	REGION_35	.209	.027	4658.0	16837.1	283816.1
36	REGION_36	.084	.000	7993.7	17902.5	293675.5
37	REGION_37	.000	.000	0.0	0.0	0.0
38	REGION_38	.666	.322	5699.6	23933.2	269355.2
39	REGION_39	.346	.000	2748.8	17912.5	275033.8
40	REGION_40	.346	.001	2740.8	17900.1	275225.8
41	REGION_41	.444	.060	2976.9	12897.8	264007.0
42	REGION_42	.467	.013	4769.4	11589.4	249102.5
43	REGION_43	.000	.000	0.0	0.0	0.0
44	REGION_44	.493	.164	1597.5	14841.3	267452.5
45	REGION_45	.303	.315	1033.1	15123.5	314129.9
46	REGION_46	.289	.327	990.0	15145.0	317690.6
47	REGION_47	.515	.040	3131.8	10518.6	261393.7
48	REGION_48	.361	.001	4563.6	7907.1	230038.3
49	REGION_49	.000	.000	0.0	0.0	0.0
50	REGION_50	.493	.164	1597.5	14841.3	267452.5
51	REGION_51	.301	.317	1027.0	15126.5	314629.3
52	REGION_52	.289	.327	990.0	15145.0	317691.3
53	REGION_53	.455	.043	1411.0	10735.1	307246.6
54	REGION_54	.277	.002	1563.5	8142.3	228214.0
55	REGION_55	.000	.000	0.0	0.0	0.0
56	REGION_56	.493	.164	1597.5	14841.3	267452.5
57	REGION_57	.309	.310	1050.9	15114.5	312651.0
58	REGION_58	.257	.249	1514.8	16971.1	332311.6
59	REGION_59	.157	.022	833.4	16703.4	326956.9
60	REGION_60	.311	.043	1841.7	8978.3	227750.8
61	REGION_61	.000	.000	0.0	0.0	0.0
62	REGION_62	.000	.000	0.0	0.0	0.0
63	REGION_63	.329	.295	1108.2	15085.9	307920.1
64	REGION_64	.246	.223	1693.3	17592.2	337283.3
65	REGION_65	.202	.075	1465.8	20888.5	417597.2
66	REGION_66	.272	.049	1241.1	7876.9	214429.8
67	REGION_67	.000	.000	0.0	0.0	0.0
68	REGION_68	.493	.164	1597.5	14841.3	267452.5
69	REGION_69	.455	.194	1485.4	14897.3	276726.3
70	REGION_70	.236	.177	1988.8	18545.9	343449.3
71	REGION_71	.453	.304	1278.9	18274.9	324722.2
72	REGION_72	.463	.046	683.6	5306.7	198416.6
73	REGION_73	.000	.000	0.0	0.0	0.0
74	REGION_74	.493	.164	1597.5	14841.3	267452.5
75	REGION_75	.493	.164	1597.5	14841.3	267452.5
76	REGION_76	.462	.155	1704.8	15463.5	277354.7
77	REGION_77	.464	.393	1871.0	12193.9	259129.2
78	REGION_78	.699	.059	726.3	6501.2	227236.6
79	REGION_79	.000	.000	0.0	0.0	0.0
80	REGION_80	.493	.164	1597.5	14841.3	267452.5
81	REGION_81	.493	.164	1597.5	14841.3	267452.5
82	REGION_82	.493	.164	1597.5	14841.3	267452.6
83	REGION_83	.409	.423	1830.5	10962.2	248707.1
84	REGION_84	.542	.249	1185.0	7857.4	247338.3
85	REGION_85	.000	.000	0.0	0.0	0.0
86	REGION_86	.493	.164	1597.5	14841.3	267452.5
87	REGION_87	.500	.170	1768.2	15219.5	267531.7
88	REGION_88	.498	.168	1706.2	15082.1	267503.1
89	REGION_89	.369	.357	1947.2	11473.8	242836.9
90	REGION_90	.132	.268	512.4	5664.8	223019.3
91	REGION_91	.000	.000	0.0	0.0	0.0
92	REGION_92	.493	.164	1597.5	14841.3	267452.5
93	REGION_93	.678	.333	5987.5	24571.3	269488.8
94	REGION_94	.678	.333	5982.8	24561.0	269486.3
95	REGION_95	.406	.260	2724.2	13333.0	248126.0
96	REGION_96	.342	.453	1204.8	5227.4	229491.3
97	REGION_97	.000	.000	0.0	0.0	0.0

POPULATION

******** BEGINNING OF CHANGE CASE 1 USER INPUT ********

*

* Form 'Basic Parameters' Comment:

* Public (0-10)
*
* EANAM2 - Name of emergency response cohort
263 EZEANAM2001 'Group 2'
******** RECORD NUMBER 263 REPLACES RECORD NUMBER 42 ********
*
* WTFRAC - weighting fraction applied to results of emergency response cohort
264 EZWTFRAC001 0.417
******** RECORD NUMBER 264 REPLACES RECORD NUMBER 44 ********
*
* TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
265 TRAVELPOINT CENTERPOINT
******** RECORD NUMBER 265 REPLACES RECORD NUMBER 46 ********
*
* ESPEED - evacuee travel speed during the three phases of evacuation
266 EZESPEED001 2.235
******** RECORD NUMBER 266 REPLACES RECORD NUMBER 47 ********
267 EZESPEED002 0.894
******** RECORD NUMBER 267 REPLACES RECORD NUMBER 48 ********
268 EZESPEED003 8.941
******** RECORD NUMBER 268 REPLACES RECORD NUMBER 49 ********
*
* ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
269 EZESPMUL001 0.7
******** RECORD NUMBER 269 REPLACES RECORD NUMBER 50 ********
270 EZESPMUL002 0.7
******** RECORD NUMBER 270 REPLACES RECORD NUMBER 51 ********
271 EZESPMUL003 0.7
******** RECORD NUMBER 271 REPLACES RECORD NUMBER 52 ********
*
* REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
272 EZREFPNT001 ALARM
******** RECORD NUMBER 272 REPLACES RECORD NUMBER 53 ********
*
* DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
273 EZDURBEG001 900.
******** RECORD NUMBER 273 REPLACES RECORD NUMBER 54 ********
*
* DURMID - duration of middle phase of evacuation, in seconds.
274 EZDURMID001 10800.
******** RECORD NUMBER 274 REPLACES RECORD NUMBER 55 ********
*
* NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
275 EZNUMEVA001 12
******** RECORD NUMBER 275 REPLACES RECORD NUMBER 56 ********
*
* DLTSHL - delay from reference time point to when individual takes shelter. DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
276 EZDLTSHL001 7200.
******** RECORD NUMBER 276 REPLACES RECORD NUMBER 57 ********
277 EZDLTSHL002 7200.
******** RECORD NUMBER 277 REPLACES RECORD NUMBER 58 ********
278 EZDLTSHL003 7200.
******** RECORD NUMBER 278 REPLACES RECORD NUMBER 59 ********
279 EZDLTSHL004 7200.
******** RECORD NUMBER 279 REPLACES RECORD NUMBER 60 ********
280 EZDLTSHL005 7200.
******** RECORD NUMBER 280 REPLACES RECORD NUMBER 61 ********
281 EZDLTSHL006 7200.
******** RECORD NUMBER 281 REPLACES RECORD NUMBER 62 ********
282 EZDLTSHL007 7200.
******** RECORD NUMBER 282 REPLACES RECORD NUMBER 63 ********
283 EZDLTSHL008 7200.
******** RECORD NUMBER 283 REPLACES RECORD NUMBER 64 ********
284 EZDLTSHL009 7200.
******** RECORD NUMBER 284 REPLACES RECORD NUMBER 65 ********
285 EZDLTSHL010 7200.
******** RECORD NUMBER 285 REPLACES RECORD NUMBER 66 ********
286 EZDLTSHL011 7200.
******** RECORD NUMBER 286 REPLACES RECORD NUMBER 67 ********
287 EZDLTSHL012 7200.
******** RECORD NUMBER 287 REPLACES RECORD NUMBER 68 ********
*
* DLTEVA -Delay time to begin evacuation
288 EZDLTEVA001 3600.
******** RECORD NUMBER 288 REPLACES RECORD NUMBER 72 ********
289 EZDLTEVA002 3600.
******** RECORD NUMBER 289 REPLACES RECORD NUMBER 73 ********
290 EZDLTEVA003 3600.
******** RECORD NUMBER 290 REPLACES RECORD NUMBER 74 ********
291 EZDLTEVA004 3600.
******** RECORD NUMBER 291 REPLACES RECORD NUMBER 75 ********
292 EZDLTEVA005 3600.
******** RECORD NUMBER 292 REPLACES RECORD NUMBER 76 ********
293 EZDLTEVA006 3600.
******** RECORD NUMBER 293 REPLACES RECORD NUMBER 77 ********
294 EZDLTEVA007 3600.
******** RECORD NUMBER 294 REPLACES RECORD NUMBER 78 ********
295 EZDLTEVA008 3600.
******** RECORD NUMBER 295 REPLACES RECORD NUMBER 79 ********
296 EZDLTEVA009 3600.
******** RECORD NUMBER 296 REPLACES RECORD NUMBER 80 ********
297 EZDLTEVA010 3600.
******** RECORD NUMBER 297 REPLACES RECORD NUMBER 81 ********
298 EZDLTEVA011 3600.
******** RECORD NUMBER 298 REPLACES RECORD NUMBER 82 ********
299 EZDLTEVA012 3600.
******** RECORD NUMBER 299 REPLACES RECORD NUMBER 83 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
300 EZESPGRD001 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 300 REPLACES RECORD NUMBER 87 ********
301 EZESPGRD002 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 301 REPLACES RECORD NUMBER 88 ********
302 EZESPGRD003 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 302 REPLACES RECORD NUMBER 89 ********
303 EZESPGRD004 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.

1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 303 REPLACES RECORD NUMBER 90 ********
304 EZESPGRD005 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 304 REPLACES RECORD NUMBER 91 ********
305 EZESPGRD006 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 305 REPLACES RECORD NUMBER 92 ********
306 EZESPGRD007 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 306 REPLACES RECORD NUMBER 93 ********
307 EZESPGRD008 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 307 REPLACES RECORD NUMBER 94 ********
308 EZESPGRD009 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 308 REPLACES RECORD NUMBER 95 ********
309 EZESPGRD010 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 309 REPLACES RECORD NUMBER 96 ********
310 EZESPGRD011 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 310 REPLACES RECORD NUMBER 97 ********
311 EZESPGRD012 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 311 REPLACES RECORD NUMBER 98 ********
312 EZESPGRD013 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 312 REPLACES RECORD NUMBER 99 ********
313 EZESPGRD014 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 313 REPLACES RECORD NUMBER 100 ********
314 EZESPGRD015 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 314 REPLACES RECORD NUMBER 101 ********
315 EZESPGRD016 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 315 REPLACES RECORD NUMBER 102 ********
316 EZESPGRD017 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 316 REPLACES RECORD NUMBER 103 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
317 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 317 REPLACES RECORD NUMBER 104 ********
318 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 318 REPLACES RECORD NUMBER 105 ********
319 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 4 1 1
******** RECORD NUMBER 319 REPLACES RECORD NUMBER 106 ********
320 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2
1 4 4 4 4 4 1 1
******** RECORD NUMBER 320 REPLACES RECORD NUMBER 107 ********
321 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2
1 1 1 1 1 1 1 1
******** RECORD NUMBER 321 REPLACES RECORD NUMBER 108 ********
322 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4
4 4 4 1 1 1 1 1
******** RECORD NUMBER 322 REPLACES RECORD NUMBER 109 ********
323 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 2 2 1 4 4 4 4
4 4 1 1 1 1 1 1
******** RECORD NUMBER 323 REPLACES RECORD NUMBER 110 ********
324 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 2 1 4 4 4 2
2 1 1 1 1 1 2 2
******** RECORD NUMBER 324 REPLACES RECORD NUMBER 111 ********
325 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 2 2 1 4 2 1 4
1 4 1 1 1 2 2 2
******** RECORD NUMBER 325 REPLACES RECORD NUMBER 112 ********
326 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 2 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1
4 4 1 1 1 1 1 1
******** RECORD NUMBER 326 REPLACES RECORD NUMBER 113 ********
327 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 2 1 4 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4
4 4 2 2 1 4 2 2
******** RECORD NUMBER 327 REPLACES RECORD NUMBER 114 ********
328 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 4 1 2 1 2 2 1 4 4 2 2 1 1 4 4 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4
2 1 1 2 2 1 4 2
******** RECORD NUMBER 328 REPLACES RECORD NUMBER 115 ********
329 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 2 2 1 1 2 1 4 2 2 1 4 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4
2 1 4 2 1 4 1 1
******** RECORD NUMBER 329 REPLACES RECORD NUMBER 116 ********
330 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4
1 4 1 1 1 1 1 1
******** RECORD NUMBER 330 REPLACES RECORD NUMBER 117 ********
331 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2
1 1 2 2 2 1 2 1
******** RECORD NUMBER 331 REPLACES RECORD NUMBER 118 ********
332 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 332 REPLACES RECORD NUMBER 119 ********
333 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 333 REPLACES RECORD NUMBER 120 ********
*
* LASMOV - The outermost spatial interval of the evacuation movement zone.
334 EZLASMOV001 17
******** RECORD NUMBER 334 REPLACES RECORD NUMBER 121 ********
*
* EFFACY, KI Ingestion
335 EZEFFACY001 0.7
******** RECORD NUMBER 335 REPLACES RECORD NUMBER 259 ********
*
* POPFRAC, KI Ingestion
336 EZPOPFRC001 0.
******** RECORD NUMBER 336 REPLACES RECORD NUMBER 260 ********
.
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE 1 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE 1
NUMBER OF RECORDS CHANGED = 74
NUMBER OF RECORDS ADDED = 0
****

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	2	1	4	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	4	1	2	1	2	2	1	4	4	2
13	1	1	1	1	1	2	1	2	2	1	1	2	1	4	2	2
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1
4	4	2	2	1	1	1	1	4	4	4	2	2	1	4	4	2
5	1	1	1	1	2	1	4	4	2	2	1	4	4	2	2	1
6	2	2	1	1	1	1	2	1	1	4	2	2	1	2	1	4
7	1	1	1	1	4	4	2	2	1	1	4	4	1	1	4	1
8	2	1	4	4	2	1	2	1	2	1	4	2	1	4	2	1
9	1	2	1	4	2	1	2	1	2	1	1	4	1	4	1	4
10	2	2	2	2	1	2	1	4	2	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	2	1	1	4	4	2	1	1	1	1	4	1	1	1	1	4
13	1	4	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1	1
4	1	4	4	2	1	1	4	2	1	4	4	4	4	4	1	1
5	4	4	2	2	1	4	4	2	1	1	1	1	1	1	1	1
6	4	2	2	1	4	4	4	4	4	4	4	1	1	1	1	1
7	4	2	2	1	4	4	4	4	4	4	1	1	1	1	1	1
8	1	4	2	1	4	4	4	2	2	1	1	1	1	1	2	2
9	4	2	2	1	4	2	1	4	1	4	1	1	1	2	2	2
10	4	2	2	1	1	1	1	1	4	4	1	1	1	1	1	1
11	4	4	4	2	2	2	1	4	4	4	2	2	1	4	2	2
12	4	4	2	2	2	2	1	4	2	1	1	2	2	1	4	2
13	4	4	2	2	2	1	1	4	2	1	4	2	1	4	1	1
14	1	4	4	2	2	1	1	4	1	4	1	1	1	1	1	1
15	1	4	2	1	2	1	1	2	1	1	2	2	2	1	2	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

THE KI MODEL IS IN EFFECT

******** BEGINNING OF CHANGE CASE 2 USER INPUT ********
*
* CSFACT - Cloudshine shielding factor
337 SECSFACT001 1.
******** RECORD NUMBER 337 REPLACES RECORD NUMBER 25 ********
338 SECSFACT002 0.31
******** RECORD NUMBER 338 REPLACES RECORD NUMBER 26 ********
339 SECSFACT003 0.31
******** RECORD NUMBER 339 REPLACES RECORD NUMBER 27 ********
*
* PROTIN - Inhalation protection factor
340 SEPROTIN001 0.98
******** RECORD NUMBER 340 REPLACES RECORD NUMBER 28 ********
341 SEPROTIN002 0.33
******** RECORD NUMBER 341 REPLACES RECORD NUMBER 29 ********
342 SEPROTIN003 0.33
******** RECORD NUMBER 342 REPLACES RECORD NUMBER 30 ********
*
* BRRATE - Breathing rates

343 SEBRRATE001 2.66E-04
******** RECORD NUMBER 343 REPLACES RECORD NUMBER 31 ********
344 SEBRRATE002 2.66E-04
******** RECORD NUMBER 344 REPLACES RECORD NUMBER 32 ********
345 SEBRRATE003 2.66E-04
******** RECORD NUMBER 345 REPLACES RECORD NUMBER 33 ********
*
* SKPFAC - skin protection factors
346 SESKPFAC001 0.98
******** RECORD NUMBER 346 REPLACES RECORD NUMBER 34 ********
347 SESKPFAC002 0.33
******** RECORD NUMBER 347 REPLACES RECORD NUMBER 35 ********
348 SESKPFAC003 0.33
******** RECORD NUMBER 348 REPLACES RECORD NUMBER 36 ********
*
* GSHFAC - groundshine shielding factors
349 SEGSHFAC001 0.5
******** RECORD NUMBER 349 REPLACES RECORD NUMBER 37 ********
350 SEGSHFAC002 0.05
******** RECORD NUMBER 350 REPLACES RECORD NUMBER 38 ********
351 SEGSHFAC003 0.05
******** RECORD NUMBER 351 REPLACES RECORD NUMBER 39 ********
*
* Form 'Basic Parameters' Comment:
* Special Facilities (0-10 miles)
*
* EANAM2 - Name of emergency response cohort
352 EZEANAM2001 'Group 3'
******** RECORD NUMBER 352 REPLACES RECORD NUMBER 42 ********
*
* WTFRAC - weighting fraction applied to results of emergency response cohort
353 EZWTFRAC001 0.006
******** RECORD NUMBER 353 REPLACES RECORD NUMBER 44 ********
*
* TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
354 TRAVELPOINT CENTERPOINT
******** RECORD NUMBER 354 REPLACES RECORD NUMBER 46 ********
*
* ESPEED - evacuee travel speed during the three phases of evacuation
355 EZESPEED001 0.894
******** RECORD NUMBER 355 REPLACES RECORD NUMBER 47 ********
356 EZESPEED002 6.706
******** RECORD NUMBER 356 REPLACES RECORD NUMBER 48 ********
357 EZESPEED003 8.941
******** RECORD NUMBER 357 REPLACES RECORD NUMBER 49 ********
*
* ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
358 EZESPMUL001 0.7
******** RECORD NUMBER 358 REPLACES RECORD NUMBER 50 ********
359 EZESPMUL002 0.7
******** RECORD NUMBER 359 REPLACES RECORD NUMBER 51 ********
360 EZESPMUL003 0.7
******** RECORD NUMBER 360 REPLACES RECORD NUMBER 52 ********
*
* REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
361 EZREFPNT001 ALARM
******** RECORD NUMBER 361 REPLACES RECORD NUMBER 53 ********
*
* DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
362 EZDURBEG001 1800.
******** RECORD NUMBER 362 REPLACES RECORD NUMBER 54 ********
*
* DURMID - duration of middle phase of evacuation, in seconds.
363 EZDURMID001 1800.
******** RECORD NUMBER 363 REPLACES RECORD NUMBER 55 ********
*
* NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
364 EZNUMEVA001 12
******** RECORD NUMBER 364 REPLACES RECORD NUMBER 56 ********
*
* DLTSHL - delay from reference time point to when individual takes shelter. DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
365 EZDLTSHL001 0.
******** RECORD NUMBER 365 REPLACES RECORD NUMBER 57 ********
366 EZDLTSHL002 0.
******** RECORD NUMBER 366 REPLACES RECORD NUMBER 58 ********
367 EZDLTSHL003 0.
******** RECORD NUMBER 367 REPLACES RECORD NUMBER 59 ********
368 EZDLTSHL004 0.
******** RECORD NUMBER 368 REPLACES RECORD NUMBER 60 ********
369 EZDLTSHL005 0.
******** RECORD NUMBER 369 REPLACES RECORD NUMBER 61 ********
370 EZDLTSHL006 0.
******** RECORD NUMBER 370 REPLACES RECORD NUMBER 62 ********
371 EZDLTSHL007 0.
******** RECORD NUMBER 371 REPLACES RECORD NUMBER 63 ********
372 EZDLTSHL008 0.
******** RECORD NUMBER 372 REPLACES RECORD NUMBER 64 ********
373 EZDLTSHL009 0.
******** RECORD NUMBER 373 REPLACES RECORD NUMBER 65 ********
374 EZDLTSHL010 0.
******** RECORD NUMBER 374 REPLACES RECORD NUMBER 66 ********
375 EZDLTSHL011 0.
******** RECORD NUMBER 375 REPLACES RECORD NUMBER 67 ********
376 EZDLTSHL012 0.
******** RECORD NUMBER 376 REPLACES RECORD NUMBER 68 ********
*
* DLTEVA -Delay time to begin evacuation
377 EZDLTEVA001 14400.
******** RECORD NUMBER 377 REPLACES RECORD NUMBER 72 ********
378 EZDLTEVA002 14400.
******** RECORD NUMBER 378 REPLACES RECORD NUMBER 73 ********
379 EZDLTEVA003 14400.
******** RECORD NUMBER 379 REPLACES RECORD NUMBER 74 ********
380 EZDLTEVA004 14400.
******** RECORD NUMBER 380 REPLACES RECORD NUMBER 75 ********
381 EZDLTEVA005 14400.
******** RECORD NUMBER 381 REPLACES RECORD NUMBER 76 ********
382 EZDLTEVA006 14400.
******** RECORD NUMBER 382 REPLACES RECORD NUMBER 77 ********

383 EZDLTEVA007 14400.
******** RECORD NUMBER 383 REPLACES RECORD NUMBER 78 ********
384 EZDLTEVA008 14400.
******** RECORD NUMBER 384 REPLACES RECORD NUMBER 79 ********
385 EZDLTEVA009 14400.
******** RECORD NUMBER 385 REPLACES RECORD NUMBER 80 ********
386 EZDLTEVA010 14400.
******** RECORD NUMBER 386 REPLACES RECORD NUMBER 81 ********
387 EZDLTEVA011 14400.
******** RECORD NUMBER 387 REPLACES RECORD NUMBER 82 ********
388 EZDLTEVA012 14400.
******** RECORD NUMBER 388 REPLACES RECORD NUMBER 83 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
389 EZESPGRD001 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 389 REPLACES RECORD NUMBER 87 ********
390 EZESPGRD002 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 390 REPLACES RECORD NUMBER 88 ********
391 EZESPGRD003 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 391 REPLACES RECORD NUMBER 89 ********
392 EZESPGRD004 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 392 REPLACES RECORD NUMBER 90 ********
393 EZESPGRD005 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 393 REPLACES RECORD NUMBER 91 ********
394 EZESPGRD006 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 394 REPLACES RECORD NUMBER 92 ********
395 EZESPGRD007 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 395 REPLACES RECORD NUMBER 93 ********
396 EZESPGRD008 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 396 REPLACES RECORD NUMBER 94 ********
397 EZESPGRD009 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 397 REPLACES RECORD NUMBER 95 ********
398 EZESPGRD010 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 398 REPLACES RECORD NUMBER 96 ********
399 EZESPGRD011 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 399 REPLACES RECORD NUMBER 97 ********
400 EZESPGRD012 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 400 REPLACES RECORD NUMBER 98 ********
401 EZESPGRD013 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 401 REPLACES RECORD NUMBER 99 ********
402 EZESPGRD014 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 402 REPLACES RECORD NUMBER 100 ********
403 EZESPGRD015 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 403 REPLACES RECORD NUMBER 101 ********
404 EZESPGRD016 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 404 REPLACES RECORD NUMBER 102 ********
405 EZESPGRD017 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 405 REPLACES RECORD NUMBER 103 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
406 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 406 REPLACES RECORD NUMBER 104 ********
407 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 407 REPLACES RECORD NUMBER 105 ********
408 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 4 1 1
******** RECORD NUMBER 408 REPLACES RECORD NUMBER 106 ********
409 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2
1 4 4 4 4 4 1 1
******** RECORD NUMBER 409 REPLACES RECORD NUMBER 107 ********
410 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2
1 1 1 1 1 1 1 1
******** RECORD NUMBER 410 REPLACES RECORD NUMBER 108 ********
411 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4
4 4 4 1 1 1 1 1
******** RECORD NUMBER 411 REPLACES RECORD NUMBER 109 ********
412 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 2 2 1 4 4 4 4
4 4 1 1 1 1 1 1
******** RECORD NUMBER 412 REPLACES RECORD NUMBER 110 ********
413 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 2 1 4 4 4 2
2 1 1 1 1 1 2 2
******** RECORD NUMBER 413 REPLACES RECORD NUMBER 111 ********
414 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 2 2 1 4 2 1 4
1 4 1 1 1 2 2 2
******** RECORD NUMBER 414 REPLACES RECORD NUMBER 112 ********
415 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 2 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1
4 4 1 1 1 1 1 1
******** RECORD NUMBER 415 REPLACES RECORD NUMBER 113 ********
416 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 2 1 4 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4
4 4 2 2 1 4 2 2
******** RECORD NUMBER 416 REPLACES RECORD NUMBER 114 ********
417 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 4 1 2 1 2 2 1 4 4 2 2 1 1 4 4 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4
2 1 1 2 2 1 4 2
******** RECORD NUMBER 417 REPLACES RECORD NUMBER 115 ********
418 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 2 2 1 1 2 1 4 2 2 1 4 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4
2 1 4 2 1 4 1 1
******** RECORD NUMBER 418 REPLACES RECORD NUMBER 116 ********
419 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4
1 4 1 1 1 1 1 1
******** RECORD NUMBER 419 REPLACES RECORD NUMBER 117 ********

```
 420  EZIDIREC015    1    1    4    2    2    2    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    4    4    2    2    2    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    1    2    1    1    1    4    2    1    2    1    1    2
1    1    2    2    2    1    2    1
********   RECORD NUMBER   420 REPLACES RECORD NUMBER   118   ********
 421  EZIDIREC016    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1
1    1    1    1    1    1    1    1
********   RECORD NUMBER   421 REPLACES RECORD NUMBER   119   ********
 422  EZIDIREC017    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1
1    1    1    1    1    1    1    1
********   RECORD NUMBER   422 REPLACES RECORD NUMBER   120   ********
     *
     * LASMOV - The outermost spatial interval of the evacuation movement zone.
 423  EZLASMOV001      17
********   RECORD NUMBER   423 REPLACES RECORD NUMBER   121   ********
     *
     * EFFACY,  KI Ingestion
 424  EZEFFACY001      0.7
********   RECORD NUMBER   424 REPLACES RECORD NUMBER   259   ********
     *
     * POPFRAC, KI Ingestion
 425  EZPOPFRC001      1.
********   RECORD NUMBER   425 REPLACES RECORD NUMBER   260   ********
     .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE   2 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE   2
NUMBER OF RECORDS CHANGED            =   89
NUMBER OF RECORDS ADDED              =    0
****************************************************************************************************
```

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	2	1	4	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	4	1	2	1	2	2	1	4	4	2
13	1	1	1	1	1	2	1	2	2	1	1	2	1	4	2	2
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1
4	4	2	2	1	1	1	1	4	4	4	2	2	1	4	4	2
5	1	1	1	1	2	1	4	4	2	2	1	4	4	2	2	1
6	2	2	1	1	1	1	2	1	1	4	2	2	1	2	1	4
7	1	1	1	1	4	4	2	2	1	1	4	4	1	1	4	1
8	2	1	4	4	2	1	2	1	2	1	4	2	1	4	2	1
9	1	2	1	4	2	1	2	1	2	1	1	4	1	4	1	4
10	2	2	2	2	1	2	1	4	2	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	2	1	1	4	4	2	1	1	1	1	4	1	1	1	1	4
13	1	4	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1	1
4	1	4	4	2	1	1	4	2	1	4	4	4	4	4	1	1
5	4	4	2	2	1	4	4	2	1	1	1	1	1	1	1	1
6	4	2	2	1	4	4	4	4	4	4	4	1	1	1	1	1
7	4	2	2	1	4	4	4	4	4	4	1	1	1	1	1	1
8	1	4	2	1	4	4	4	2	2	1	1	1	1	1	2	2
9	4	2	2	1	4	2	1	4	1	4	1	1	1	2	2	2
10	4	2	2	1	1	1	1	1	4	4	1	1	1	1	1	1
11	4	4	4	2	2	2	1	4	4	4	2	2	1	4	2	2
12	4	4	2	2	2	2	1	4	2	1	1	2	2	1	4	2
13	4	4	2	2	2	1	1	4	2	1	4	2	1	4	1	1
14	1	4	4	2	2	1	1	4	1	4	1	1	1	1	1	1
15	1	4	2	1	2	1	1	2	1	1	2	2	2	1	2	1

```
  16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE   3 USER INPUT ********
      *
      * CSFACT - Cloudshine shielding factor
  426  SECSFACT001      1.
********   RECORD NUMBER   426 REPLACES RECORD NUMBER    25   ********
  427  SECSFACT002      0.6
********   RECORD NUMBER   427 REPLACES RECORD NUMBER    26   ********
  428  SECSFACT003      0.5
********   RECORD NUMBER   428 REPLACES RECORD NUMBER    27   ********
      *
      * PROTIN - Inhalation protection factor
  429  SEPROTIN001      0.98
********   RECORD NUMBER   429 REPLACES RECORD NUMBER    28   ********
  430  SEPROTIN002      0.46
********   RECORD NUMBER   430 REPLACES RECORD NUMBER    29   ********
  431  SEPROTIN003      0.33
********   RECORD NUMBER   431 REPLACES RECORD NUMBER    30   ********
      *
      * BRRATE - Breathing rates
  432  SEBRRATE001      2.66E-04
********   RECORD NUMBER   432 REPLACES RECORD NUMBER    31   ********
  433  SEBRRATE002      2.66E-04
********   RECORD NUMBER   433 REPLACES RECORD NUMBER    32   ********
  434  SEBRRATE003      2.66E-04
********   RECORD NUMBER   434 REPLACES RECORD NUMBER    33   ********
      *
      * SKPFAC - skin protection factors
  435  SESKPFAC001      0.98
********   RECORD NUMBER   435 REPLACES RECORD NUMBER    34   ********
  436  SESKPFAC002      0.46
********   RECORD NUMBER   436 REPLACES RECORD NUMBER    35   ********
  437  SESKPFAC003  0.33
********   RECORD NUMBER   437 REPLACES RECORD NUMBER    36   ********
      *
      * GSHFAC - groundshine shielding factors
  438  SEGSHFAC001      0.5
********   RECORD NUMBER   438 REPLACES RECORD NUMBER    37   ********
  439  SEGSHFAC002      0.18
********   RECORD NUMBER   439 REPLACES RECORD NUMBER    38   ********
  440  SEGSHFAC003      0.1
********   RECORD NUMBER   440 REPLACES RECORD NUMBER    39   ********
      *
      * Form 'Basic Parameters' Comment:
      * Tail (0-10)
      *
      * EANAM2 - Name of emergency response cohort
  441  EZEANAM2001      'Group 4'
********   RECORD NUMBER   441 REPLACES RECORD NUMBER    42   ********
      *
      * WTFRAC - weighting fraction applied to results of emergency response cohort
  442  EZWTFRAC001      0.1
********   RECORD NUMBER   442 REPLACES RECORD NUMBER    44   ********
      *
      * TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
  443  TRAVELPOINT      CENTERPOINT
********   RECORD NUMBER   443 REPLACES RECORD NUMBER    46   ********
      *
      * ESPEED - evacuee travel speed during the three phases of evacuation
  444  EZESPEED001      0.8941
********   RECORD NUMBER   444 REPLACES RECORD NUMBER    47   ********
  445  EZESPEED002      6.706
********   RECORD NUMBER   445 REPLACES RECORD NUMBER    48   ********
  446  EZESPEED003      8.941
********   RECORD NUMBER   446 REPLACES RECORD NUMBER    49   ********
      *
      * ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
  447  EZESPMUL001      0.7
********   RECORD NUMBER   447 REPLACES RECORD NUMBER    50   ********
  448  EZESPMUL002      0.7
********   RECORD NUMBER   448 REPLACES RECORD NUMBER    51   ********
  449  EZESPMUL003      0.7
********   RECORD NUMBER   449 REPLACES RECORD NUMBER    52   ********
      *
      * REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
  450  EZREFPNT001      ALARM
********   RECORD NUMBER   450 REPLACES RECORD NUMBER    53   ********
      *
      * DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
  451  EZDURBEG001      1800.
********   RECORD NUMBER   451 REPLACES RECORD NUMBER    54   ********
      *
      * DURMID - duration of middle phase of evacuation, in seconds.
  452  EZDURMID001      1800.
********   RECORD NUMBER   452 REPLACES RECORD NUMBER    55   ********
      *
      * NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
  453  EZNUMEVA001      12
********   RECORD NUMBER   453 REPLACES RECORD NUMBER    56   ********
      *
      * DLTSHL - delay from reference time point to when individual takes shelter.  DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
  454  EZDLTSHL001      7200.
********   RECORD NUMBER   454 REPLACES RECORD NUMBER    57   ********
  455  EZDLTSHL002      7200.
********   RECORD NUMBER   455 REPLACES RECORD NUMBER    58   ********
  456  EZDLTSHL003      7200.
********   RECORD NUMBER   456 REPLACES RECORD NUMBER    59   ********
  457  EZDLTSHL004      7200.
********   RECORD NUMBER   457 REPLACES RECORD NUMBER    60   ********
  458  EZDLTSHL005      7200.
********   RECORD NUMBER   458 REPLACES RECORD NUMBER    61   ********
  459  EZDLTSHL006      7200.
********   RECORD NUMBER   459 REPLACES RECORD NUMBER    62   ********
  460  EZDLTSHL007      7200.
********   RECORD NUMBER   460 REPLACES RECORD NUMBER    63   ********
```

461 EZDLTSHL008 7200.
******** RECORD NUMBER 461 REPLACES RECORD NUMBER 64 ********
462 EZDLTSHL009 7200.
******** RECORD NUMBER 462 REPLACES RECORD NUMBER 65 ********
463 EZDLTSHL010 7200.
******** RECORD NUMBER 463 REPLACES RECORD NUMBER 66 ********
464 EZDLTSHL011 7200.
******** RECORD NUMBER 464 REPLACES RECORD NUMBER 67 ********
465 EZDLTSHL012 7200.
******** RECORD NUMBER 465 REPLACES RECORD NUMBER 68 ********
*
* DLTEVA -Delay time to begin evacuation
466 EZDLTEVA001 10800.
******** RECORD NUMBER 466 REPLACES RECORD NUMBER 72 ********
467 EZDLTEVA002 10800.
******** RECORD NUMBER 467 REPLACES RECORD NUMBER 73 ********
468 EZDLTEVA003 10800.
******** RECORD NUMBER 468 REPLACES RECORD NUMBER 74 ********
469 EZDLTEVA004 10800.
******** RECORD NUMBER 469 REPLACES RECORD NUMBER 75 ********
470 EZDLTEVA005 10800.
******** RECORD NUMBER 470 REPLACES RECORD NUMBER 76 ********
471 EZDLTEVA006 10800.
******** RECORD NUMBER 471 REPLACES RECORD NUMBER 77 ********
472 EZDLTEVA007 10800.
******** RECORD NUMBER 472 REPLACES RECORD NUMBER 78 ********
473 EZDLTEVA008 10800.
******** RECORD NUMBER 473 REPLACES RECORD NUMBER 79 ********
474 EZDLTEVA009 10800.
******** RECORD NUMBER 474 REPLACES RECORD NUMBER 80 ********
475 EZDLTEVA010 10800.
******** RECORD NUMBER 475 REPLACES RECORD NUMBER 81 ********
476 EZDLTEVA011 10800.
******** RECORD NUMBER 476 REPLACES RECORD NUMBER 82 ********
477 EZDLTEVA012 10800.
******** RECORD NUMBER 477 REPLACES RECORD NUMBER 83 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
478 EZESPGRD001 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 478 REPLACES RECORD NUMBER 87 ********
479 EZESPGRD002 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 479 REPLACES RECORD NUMBER 88 ********
480 EZESPGRD003 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 480 REPLACES RECORD NUMBER 89 ********
481 EZESPGRD004 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 481 REPLACES RECORD NUMBER 90 ********
482 EZESPGRD005 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 482 REPLACES RECORD NUMBER 91 ********
483 EZESPGRD006 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 483 REPLACES RECORD NUMBER 92 ********
484 EZESPGRD007 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 484 REPLACES RECORD NUMBER 93 ********
485 EZESPGRD008 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 485 REPLACES RECORD NUMBER 94 ********
486 EZESPGRD009 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 486 REPLACES RECORD NUMBER 95 ********
487 EZESPGRD010 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 487 REPLACES RECORD NUMBER 96 ********
488 EZESPGRD011 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 488 REPLACES RECORD NUMBER 97 ********
489 EZESPGRD012 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 489 REPLACES RECORD NUMBER 98 ********
490 EZESPGRD013 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 490 REPLACES RECORD NUMBER 99 ********
491 EZESPGRD014 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 491 REPLACES RECORD NUMBER 100 ********
492 EZESPGRD015 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 492 REPLACES RECORD NUMBER 101 ********
493 EZESPGRD016 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 493 REPLACES RECORD NUMBER 102 ********
494 EZESPGRD017 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 494 REPLACES RECORD NUMBER 103 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
495 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 495 REPLACES RECORD NUMBER 104 ********
496 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 496 REPLACES RECORD NUMBER 105 ********
497 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 4 1 1
******** RECORD NUMBER 497 REPLACES RECORD NUMBER 106 ********
498 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2
1 4 4 4 4 4 1 1
******** RECORD NUMBER 498 REPLACES RECORD NUMBER 107 ********
499 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2
1 1 1 1 1 1 1 1
******** RECORD NUMBER 499 REPLACES RECORD NUMBER 108 ********
500 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4
4 4 4 1 1 1 1 1
******** RECORD NUMBER 500 REPLACES RECORD NUMBER 109 ********

```
 501  EZIDIREC007   2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 2 2 1 4 4 4 4
4 4 1 1 1 1 1 1
********  RECORD NUMBER  501 REPLACES RECORD NUMBER  110  ********
 502  EZIDIREC008   1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 2 1 4 4 4 2
2 1 1 1 1 1 2 2
********  RECORD NUMBER  502 REPLACES RECORD NUMBER  111  ********
 503  EZIDIREC009   1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 2 2 1 4 2 1 4
1 4 1 1 1 2 2 2
********  RECORD NUMBER  503 REPLACES RECORD NUMBER  112  ********
 504  EZIDIREC010   1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 2 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1
4 4 1 1 1 1 1 1
********  RECORD NUMBER  504 REPLACES RECORD NUMBER  113  ********
 505  EZIDIREC011 1 1 4 2 1 4 2 1 4 4 2 1 4 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4 4
4 2 2 1 4 2 2
********  RECORD NUMBER  505 REPLACES RECORD NUMBER  114  ********
 506  EZIDIREC012   2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 4 1 2 1 2 2 1 4 4 2 2 1 1 4 4 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4
2 1 1 2 2 1 4 2
********  RECORD NUMBER  506 REPLACES RECORD NUMBER  115  ********
 507  EZIDIREC013   1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 2 2 1 1 2 1 4 2 2 1 4 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4
2 1 4 2 1 4 1 1
********  RECORD NUMBER  507 REPLACES RECORD NUMBER  116  ********
 508  EZIDIREC014   1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4
1 4 1 1 1 1 1 1
********  RECORD NUMBER  508 REPLACES RECORD NUMBER  117  ********
 509  EZIDIREC015   1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2
1 1 2 2 2 1 2 1
********  RECORD NUMBER  509 REPLACES RECORD NUMBER  118  ********
 510  EZIDIREC016   1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
********  RECORD NUMBER  510 REPLACES RECORD NUMBER  119  ********
 511  EZIDIREC017   1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
********  RECORD NUMBER  511 REPLACES RECORD NUMBER  120  ********
    *
    * LASMOV - The outermost spatial interval of the evacuation movement zone.
 512  EZLASMOV001     17
********  RECORD NUMBER  512 REPLACES RECORD NUMBER  121  ********
    *
    * EFFACY,  KI Ingestion
 513  EZEFFACY001     0.7
********  RECORD NUMBER  513 REPLACES RECORD NUMBER  259  ********
    *
    * POPFRAC, KI Ingestion
 514  EZPOPFRC001     0.
********  RECORD NUMBER  514 REPLACES RECORD NUMBER  260  ********
    .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE   3 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE   3
NUMBER OF RECORDS CHANGED          =   89
NUMBER OF RECORDS ADDED            =    0
*****************************************************************************************************
```

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

```
IRAD     1  2  3  4  5  6  7  8  9 10 11 12 13 14 15 16
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    1  1  1  1  1  1  1  1  1  1  1  1  1  2  1  1
  4    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  5    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4
  6    1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  1
  7    2  2  1  2  2  1  2  2  1  4  2  1  4  2  2  1
  8    1  4  1  1  4  2  1  4  2  1  1  4  2  2  1  1
  9    1  1  4  2  1  1  2  1  1  4  1  4  1  4  1  1
 10    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 11    1  1  4  2  1  4  2  1  4  4  2  1  4  2  1  4
 12    2  1  1  4  1  1  4  1  4  4  2  1  4  2  1  1
 13    1  1  4  1  4  2  1  1  2  1  4  4  2  2  1  1
 14    1  1  4  1  1  1  2  1  2  1  2  1  1  2  1  1
 15    1  1  4  2  2  2  1  1  1  1  1  1  1  1  1  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD    17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    1  1  1  1  1  1  1  1  2  1  4  4  4  4  2  2
  4    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  5    1  1  1  1  1  1  1  1  1  2  2  2  2  2  2  1
  6    1  1  1  1  1  1  1  1  1  1  2  2  2  2  2  2
  7    1  1  1  1  1  1  1  1  1  1  2  2  1  4  4  2
  8    2  1  1  1  4  4  4  1  1  1  1  1  1  4  1  4
  9    1  2  1  1  1  4  4  4  1  1  4  1  2  2  2  1
 10    1  1  1  1  1  1  1  1  1  1  1  1  1  4  4  4
 11    4  2  2  2  2  2  2  1  4  4  1  1  4  2  2  1
 12    4  4  2  2  2  1  4  1  2  1  2  2  1  4  4  2
 13    1  1  1  1  1  2  1  2  2  1  1  2  1  4  2  2
 14    1  1  1  1  1  4  1  1  1  1  2  1  4  1  1  1
 15    1  1  1  1  1  1  1  4  4  4  2  2  2  1  1  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD    33 34 35 36 37 38 39 40 41 42 43 44 45 46 47 48
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    2  2  2  1  1  1  4  4  4  4  2  2  2  1  1  1
  4    4  2  2  1  1  1  1  4  4  4  2  2  1  4  4  2
  5    1  1  1  1  2  1  4  4  2  2  1  4  4  2  2  1
  6    2  2  1  1  1  1  2  1  1  4  2  2  1  2  1  4
  7    1  1  1  1  4  4  2  2  1  1  4  4  1  1  4  1
  8    2  1  4  4  2  1  2  1  2  1  4  2  1  4  2  1
  9    1  2  1  4  2  1  2  1  2  1  1  4  1  4  1  4
 10    2  2  2  2  1  2  1  4  2  1  1  1  4  2  1  4
```

```
 11    1  1  4  2  1  4  1  4  2  1  4  1  1  2  1  1
 12    2  1  1  4  4  2  1  1  1  1  4  1  1  1  1  4
 13    1  4  2  1  4  4  1  1  4  2  1  1  2  1  2  1
 14    1  1  1  1  1  1  1  1  1  1  1  1  1  1  2  1
 15    1  1  1  1  1  1  1  1  1  1  1  4  1  2  1  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD     49  50  51  52  53  54  55  56  57  58  59  60  61  62  63  64
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  1
  4    1  4  4  2  1  1  4  2  1  4  4  4  4  4  1  1
  5    4  4  2  2  1  4  4  2  1  1  1  1  1  1  1  1
  6    4  2  2  1  4  4  4  4  4  4  4  1  1  1  1  1
  7    4  2  2  1  4  4  4  4  4  4  1  1  1  1  1  1
  8    1  4  2  1  4  4  4  2  2  1  1  1  1  1  2  2
  9    4  2  2  1  4  2  1  4  1  4  1  1  1  2  2  2
 10    4  2  2  1  1  1  1  1  4  4  1  1  1  1  1  1
 11    4  4  4  2  2  2  1  4  4  4  2  2  1  4  2  2
 12    4  4  2  2  2  2  1  4  2  1  1  2  2  1  4  2
 13    4  4  2  2  2  1  1  4  2  1  4  2  1  4  1  1
 14    1  4  4  2  2  1  1  4  1  4  1  1  1  1  1  1
 15    1  4  2  1  2  1  1  2  1  1  2  2  2  1  2  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE   4 USER INPUT ********
     *
     * Form 'Basic Parameters' Comment:
     * Schools (0-10)
     *
     * EANAM2 - Name of emergency response cohort
 515  EZEANAM2001      'Group 5'
********   RECORD NUMBER   515 REPLACES RECORD NUMBER    42   ********
     *
     * WTFRAC - weighting fraction applied to results of emergency response cohort
 516  EZWTFRAC001      0.172
********   RECORD NUMBER   516 REPLACES RECORD NUMBER    44   ********
     *
     * TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
 517  TRAVELPOINT      CENTERPOINT
********   RECORD NUMBER   517 REPLACES RECORD NUMBER    46   ********
     *
     * ESPEED - evacuee travel speed during the three phases of evacuation
 518  EZESPEED001       8.941
********   RECORD NUMBER   518 REPLACES RECORD NUMBER    47   ********
 519  EZESPEED002      6.706
********   RECORD NUMBER   519 REPLACES RECORD NUMBER    48   ********
 520  EZESPEED003      8.941
********   RECORD NUMBER   520 REPLACES RECORD NUMBER    49   ********
     *
     * ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
 521  EZESPMUL001      0.7
********   RECORD NUMBER   521 REPLACES RECORD NUMBER    50   ********
 522  EZESPMUL002      0.7
********   RECORD NUMBER   522 REPLACES RECORD NUMBER    51   ********
 523  EZESPMUL003      0.7
********   RECORD NUMBER   523 REPLACES RECORD NUMBER    52   ********
     *
     * REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
 524  EZREFPNT001      ALARM
********   RECORD NUMBER   524 REPLACES RECORD NUMBER    53   ********
     *
     * DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
 525  EZDURBEG001      3600.
********   RECORD NUMBER   525 REPLACES RECORD NUMBER    54   ********
     *
     * DURMID - duration of middle phase of evacuation, in seconds.
 526  EZDURMID001      1800.
********   RECORD NUMBER   526 REPLACES RECORD NUMBER    55   ********
     *
     * NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
 527  EZNUMEVA001      12
********   RECORD NUMBER   527 REPLACES RECORD NUMBER    56   ********
     *
     * DLTSHL - delay from reference time point to when individual takes shelter.  DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
 528  EZDLTSHL001      0.
********   RECORD NUMBER   528 REPLACES RECORD NUMBER    57   ********
 529  EZDLTSHL002      0.
********   RECORD NUMBER   529 REPLACES RECORD NUMBER    58   ********
 530  EZDLTSHL003      0.
********   RECORD NUMBER   530 REPLACES RECORD NUMBER    59   ********
 531  EZDLTSHL004      0.
********   RECORD NUMBER   531 REPLACES RECORD NUMBER    60   ********
 532  EZDLTSHL005      0.
********   RECORD NUMBER   532 REPLACES RECORD NUMBER    61   ********
 533  EZDLTSHL006      0.
********   RECORD NUMBER   533 REPLACES RECORD NUMBER    62   ********
 534  EZDLTSHL007      0.
********   RECORD NUMBER   534 REPLACES RECORD NUMBER    63   ********
 535  EZDLTSHL008      0.
********   RECORD NUMBER   535 REPLACES RECORD NUMBER    64   ********
 536  EZDLTSHL009      0.
********   RECORD NUMBER   536 REPLACES RECORD NUMBER    65   ********
 537  EZDLTSHL010      0.
********   RECORD NUMBER   537 REPLACES RECORD NUMBER    66   ********
 538  EZDLTSHL011      0.
********   RECORD NUMBER   538 REPLACES RECORD NUMBER    67   ********
 539  EZDLTSHL012      0.
********   RECORD NUMBER   539 REPLACES RECORD NUMBER    68   ********
     *
     * DLTEVA -Delay time to begin evacuation
 540  EZDLTEVA001      1800.
********   RECORD NUMBER   540 REPLACES RECORD NUMBER    72   ********
 541  EZDLTEVA002      1800.
********   RECORD NUMBER   541 REPLACES RECORD NUMBER    73   ********
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 542 EZDLTEVA003 1800.
******** RECORD NUMBER 542 REPLACES RECORD NUMBER 74 ********
 543 EZDLTEVA004 1800.
******** RECORD NUMBER 543 REPLACES RECORD NUMBER 75 ********
 544 EZDLTEVA005 1800.
******** RECORD NUMBER 544 REPLACES RECORD NUMBER 76 ********
 545 EZDLTEVA006 1800.
******** RECORD NUMBER 545 REPLACES RECORD NUMBER 77 ********
 546 EZDLTEVA007 1800.
******** RECORD NUMBER 546 REPLACES RECORD NUMBER 78 ********
 547 EZDLTEVA008 1800.
******** RECORD NUMBER 547 REPLACES RECORD NUMBER 79 ********
 548 EZDLTEVA009 1800.
******** RECORD NUMBER 548 REPLACES RECORD NUMBER 80 ********
 549 EZDLTEVA010 1800.
******** RECORD NUMBER 549 REPLACES RECORD NUMBER 81 ********
 550 EZDLTEVA011 1800.
******** RECORD NUMBER 550 REPLACES RECORD NUMBER 82 ********
 551 EZDLTEVA012 1800.
******** RECORD NUMBER 551 REPLACES RECORD NUMBER 83 ********
 *
 * ESPGRD_NET - Evacuation speed multiplier for network evacuation
 552 EZESPGRD001 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 552 REPLACES RECORD NUMBER 87 ********
 553 EZESPGRD002 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 553 REPLACES RECORD NUMBER 88 ********
 554 EZESPGRD003 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 554 REPLACES RECORD NUMBER 89 ********
 555 EZESPGRD004 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 555 REPLACES RECORD NUMBER 90 ********
 556 EZESPGRD005 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 556 REPLACES RECORD NUMBER 91 ********
 557 EZESPGRD006 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 557 REPLACES RECORD NUMBER 92 ********
 558 EZESPGRD007 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 558 REPLACES RECORD NUMBER 93 ********
 559 EZESPGRD008 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 559 REPLACES RECORD NUMBER 94 ********
 560 EZESPGRD009 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 560 REPLACES RECORD NUMBER 95 ********
 561 EZESPGRD010 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 561 REPLACES RECORD NUMBER 96 ********
 562 EZESPGRD011 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 562 REPLACES RECORD NUMBER 97 ********
 563 EZESPGRD012 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 563 REPLACES RECORD NUMBER 98 ********
 564 EZESPGRD013 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 564 REPLACES RECORD NUMBER 99 ********
 565 EZESPGRD014 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 565 REPLACES RECORD NUMBER 100 ********
 566 EZESPGRD015 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. .75
1. 1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 566 REPLACES RECORD NUMBER 101 ********
 567 EZESPGRD016 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 567 REPLACES RECORD NUMBER 102 ********
 568 EZESPGRD017 .75 .75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. .75 .75 .75
******** RECORD NUMBER 568 REPLACES RECORD NUMBER 103 ********
 *
 * IDIREC - destination direction of every spatial element in the evacuation and sheltering region
 569 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 569 REPLACES RECORD NUMBER 104 ********
 570 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 570 REPLACES RECORD NUMBER 105 ********
 571 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 4 1 1
******** RECORD NUMBER 571 REPLACES RECORD NUMBER 106 ********
 572 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2
1 4 4 4 4 4 1 1
******** RECORD NUMBER 572 REPLACES RECORD NUMBER 107 ********
 573 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2
1 1 1 1 1 1 1 1
******** RECORD NUMBER 573 REPLACES RECORD NUMBER 108 ********
 574 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4
4 4 4 1 1 1 1 1
******** RECORD NUMBER 574 REPLACES RECORD NUMBER 109 ********
 575 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 2 2 1 4 4 4 4
4 4 1 1 1 1 1 1
******** RECORD NUMBER 575 REPLACES RECORD NUMBER 110 ********
 576 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 2 1 4 4 4 2
2 1 1 1 1 1 2 2
******** RECORD NUMBER 576 REPLACES RECORD NUMBER 111 ********
 577 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 2 2 1 4 2 1 4
1 4 1 1 1 2 2 2
******** RECORD NUMBER 577 REPLACES RECORD NUMBER 112 ********
 578 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 2 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1
4 4 1 1 1 1 1 1
******** RECORD NUMBER 578 REPLACES RECORD NUMBER 113 ********
 579 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 2 1 4 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4
4 4 2 2 1 4 2 2
******** RECORD NUMBER 579 REPLACES RECORD NUMBER 114 ********
 580 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 4 1 2 1 2 2 1 4 4 2 2 1 1 4 4 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4
```

2 1 1 2 2 1 4 2
******** RECORD NUMBER 580 REPLACES RECORD NUMBER 115 ********
581 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 2 2 1 1 2 1 4 2 2 1 4 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4
2 1 4 2 1 4 1 1
******** RECORD NUMBER 581 REPLACES RECORD NUMBER 116 ********
582 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4
1 4 1 1 1 1 1 1
******** RECORD NUMBER 582 REPLACES RECORD NUMBER 117 ********
583 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2
1 1 2 2 2 1 2 1
******** RECORD NUMBER 583 REPLACES RECORD NUMBER 118 ********
584 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 584 REPLACES RECORD NUMBER 119 ********
585 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1 1
******** RECORD NUMBER 585 REPLACES RECORD NUMBER 120 ********
*
* LASMOV - The outermost spatial interval of the evacuation movement zone.
586 EZLASMOV001 17
******** RECORD NUMBER 586 REPLACES RECORD NUMBER 121 ********
*
* EFFACY, KI Ingestion
587 EZEFFACY001 0.7
******** RECORD NUMBER 587 REPLACES RECORD NUMBER 259 ********
*
* POPFRAC, KI Ingestion
588 EZPOPFRC001 0.
******** RECORD NUMBER 588 REPLACES RECORD NUMBER 260 ********
.
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE 4 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE 4
NUMBER OF RECORDS CHANGED = 74
NUMBER OF RECORDS ADDED = 0
****

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	2	1	4	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	4	1	2	1	2	2	1	4	4	2
13	1	1	1	1	1	2	1	2	2	1	1	2	1	4	2	2
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1
4	4	2	2	1	1	1	1	4	4	4	2	2	1	4	4	2
5	1	1	1	1	2	1	4	4	2	2	1	4	4	2	2	1
6	2	2	1	1	1	1	2	1	1	4	2	2	1	2	1	4
7	1	1	1	1	4	4	2	2	1	1	4	4	1	1	4	1
8	2	1	4	4	2	1	2	1	2	1	4	2	1	4	2	1
9	1	2	1	4	2	1	2	1	2	1	1	4	1	4	1	4
10	2	2	2	2	1	2	1	4	2	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	2	1	1	4	4	2	1	1	1	1	4	1	1	1	1	4
13	1	4	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1	1
4	1	4	4	2	1	1	4	2	1	4	4	4	4	4	1	1
5	4	4	2	2	1	4	4	2	1	1	1	1	1	1	1	1
6	4	2	2	1	4	4	4	4	4	4	4	1	1	1	1	1
7	4	2	2	1	4	4	4	4	4	4	1	1	1	1	1	1

```
  8    1  4  2  1  4  4  4  2  2  1  1  1  1  1  2  2
  9    4  2  2  1  4  2  1  4  1  4  1  1  1  2  2  2
 10    4  2  2  1  1  1  1  1  4  4  1  1  1  1  1  1
 11    4  4  4  2  2  2  1  4  4  4  2  2  1  4  2  2
 12    4  4  2  2  2  2  1  4  2  1  1  2  2  1  4  2
 13    4  4  2  2  2  1  1  4  2  1  4  2  1  4  1  1
 14    1  4  4  2  2  1  1  4  1  4  1  1  1  1  1  1
 15    1  4  2  1  2  1  1  2  1  1  2  2  2  1  2  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE   5 USER INPUT ********
     *
     * Form 'Basic Parameters' Comment:
     * Non-evacuation (0-10)
     *
     * EANAM2 - Name of emergency response cohort
  589  EZEANAM2001     'Group 6'
********   RECORD NUMBER   589 REPLACES RECORD NUMBER    42   ********
     *
     * WTFRAC - weighting fraction applied to results of emergency response cohort
  590  EZWTFRAC001     0.005
********   RECORD NUMBER   590 REPLACES RECORD NUMBER    44   ********
     *
     * LASMOV2  (used for no evacuation), always 0
  591  EZLASMOV001     0
********   RECORD NUMBER   591 REPLACES RECORD NUMBER   121   ********
     *
     * EFFACY,  KI Ingestion
  592  EZEFFACY001     0.7
********   RECORD NUMBER   592 REPLACES RECORD NUMBER   259   ********
     *
     * POPFRAC, KI Ingestion
  593  EZPOPFRC001     0.
********   RECORD NUMBER   593 REPLACES RECORD NUMBER   260   ********
     .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE   5 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE   5
NUMBER OF RECORDS CHANGED           =    5
NUMBER OF RECORDS ADDED             =    0
****************************************************************************************************


NO EVACUATION REQUESTED
THE KI MODEL IS IN EFFECT



******** WARNING -- THE FOLLOWING RECORDS WERE NEVER ACCESSED ********

     LCDTHNUM001     1
     LCDTHANN001     1.E-04
     LCDTHLIF001     10000.
     STFRACLD001     1.0


USER INPUT IS READ FROM UNIT 26
RECORD IDENTIFIER FIELDS  11 CHARACTERS LONG ARE EXPECTED.
THE FIRST 499 COLUMNS OF EACH INPUT RECORD ARE PROCESSED.

RECORD
NUMBER                          RECORD

     * File created using WinMACCS version 3.7.0 11/13/2012 4:58:24 PM
     *
     * CHNAME - description
   1  CHCHNAME001     'OCP3 low density no spray'
     *
     * EVACST - daily cost
   2  CHEVACST001     172.
     *
     * RELCST - daily cost due to intermediate
   3  CHRELCST001     172.
     *
     * DUR_INTPHAS, intermediate-phase period
   4  DUR_INTPHAS     0.E+00
     *
     * TMPACT - long term dose period
   5  CHTMPACT001     3.16E+07
     *
     * Form 'Long Term Dose Criterion' Comment:
     * Value of DSCRLT (0.005) from Pennsylvania Bureau of Radiation Protection.
     *
     * DSCRTI - dose criterion for phase
   6  CHDSCRTI001     1.00000E+05
     *
     * DSCRLT - dose criterion for habitation
   7  CHDSCRLT001     .005
     *
     * EXPTIM - long term exposure period
   8  CHEXPTIM001     1.58E+09
     *
     * CRTOCR - critical organ
   9  CHCRTOCR001     L-ICRP60ED
     *
     * Form 'Number of Plan Levels' Comment:
     * From NUREG-1150.
     *
     * LVLDEC - number of decontamination levels
  10  CHLVLDEC001     2
     *
     * TIMDEC - time for each level
  11  CHTIMDEC001     3.15E+07
  12  CHTIMDEC002     3.15E+07
     *
     * DSRFCT - effectiveness of decontamination
```

```
13  CHDSRFCT001      3.
14  CHDSRFCT002      15.
   *
   * CDFRM - farmland decontamination cost
15  CHCDFRM0001      1330.
16  CHCDFRM0002      2960.
   *
   * CDNFRM - nonfarmland decontamination cost
17  CHCDNFRM001      7110.
18  CHCDNFRM002      19000.
   *
   * FRFDL - fraction farmland cost due labor
19  CHFRFDL0001      .3
20  CHFRFDL0002      .35
   *
   * FRNFDL - fraction nonfarmland cost due labor
21  CHFRNFDL001      .7
22  CHFRNFDL002      .5
   *
   * TFWKF - fraction time farmland worker
23  CHTFWKF0001      0.1
24  CHTFWKF0002      0.33
   *
   * TFWKNF - fraction time nonfarmland worker
25  CHTFWKNF001      0.33
26  CHTFWKNF002      0.33
   *
   * DLBCST - labor cost decontamination worker
27  CHDLBCST001      84000.
   *
   * DPRATE - depreciation rate applies to improvements
28  CHDPRATE001      .2
   *
   * DSRATE - rate of return
29  CHDSRATE001      .12
   *
   * POPCST - Per capita removal cost
30  CHPOPCST001      12000.
   *
   * NGWTRM - number weathering terms
31  CHNGWTRM001      2
   *
   * GWCOEF - groundshine coefficient
32  CHGWCOEF001      0.5
33  CHGWCOEF002      0.5
   *
   * TGWHLF - groundshine half lives
34  CHTGWHLF001      1.6E7
35  CHTGWHLF002      2.8E9
   *
   * NRWTRM - number resuspension terms
36  CHNRWTRM001      3
   *
   * RWCOEF - resuspension coefficient
37  CHRWCOEF001      1.0E-5
38  CHRWCOEF002      1.0E-7
39  CHRWCOEF003      1.0E-9
   *
   * TRWHLF - resuspension half lives
40  CHTRWHLF001      1.6E7
41  CHTRWHLF002      1.6E8
42  CHTRWHLF003      1.6E9
   *
   * VALWF - value of farm wealth
43  CHVALWF0001      9040.
   *
   * FRFIM - fraction of farm wealth due improvements
44  CHFRFIM0001      .25
   *
   * VALWNF - value of nonfarm wealth
45  CHVALWNF001      2.10000E+05
   *
   * FRNFIM - fraction nonfarm wealth due improvements
46  CHFRNFIM001      .8
   *
   * FDPATH, value = OLD, NEW or OFF to use models MACCS food, Comida2 or no food model respectively
47  CHFDPATH001      NEW
   *
   * DOSEMILK
48  DOSEMILK001      0.025
49  DOSEMILK002  0.075
   *
   * DOSEOTHR
50  DOSEOTHR001      0.025
51  DOSEOTHR002      0.075
   *
   * DOSELONG
52  DOSELONG001      0.005
53  DOSELONG002      0.015
   *
   * Form 'Water Ingestion Radionuclides' Comment:
   *
   *
   * NUMWPI - size of array NAMWPI
54  CHNUMWPI001      4
   *
   * popflg=FILE,NAMWPI, WSHFRI, WSHRTA, WINGF - water ingestion data
55  CHWTRISO001      Sr-89   0.01    0.004   0.
56  CHWTRISO002      Sr-90   0.01    0.004   0.
57  CHWTRISO003      Cs-134  0.005   0.001   0.
58  CHWTRISO004      Cs-137  0.005   0.001   0.
   *
   * KSWTCH - chronc output diagnostic switch
59  CHKSWTCH001      0
   *
   * FRCFRM_FILE - popflg = FILE, dummy variable
60  CHFRCFRM001      1.0
   *
   * FRMPRD_FILE - popflg=FILE, dummy variable
```

```
  61  CHFRMPRD001      0.0
      *
      * DPFRCT_FILE - popflg=FILE, dummy variable
  62  CHDPFRCT001      0.0
      *
      * Form 'Shielding and Exposure' Comment:
      * Data are taken directly from NUREG-1150 for normal activity.
      *
      * LPROTIN - Inhalation protection factor used in CHRONC
  63  CHLPROTIN01      .46
      *
      * LBRRATE - Breathing rate used in CHRONC
  64  CHLBRRATE01      2.66E-04
      *
      * LGSHFAC - groundshine shielding factor used in CHRONC
  65  CHLGSHFAC01      .18
      *
      * NXUM9=0
  66  TYPE9NUMBER      0
      *
      * NXUM9, number of type9 results
  67  TYPE9NUMBER      4
********   RECORD NUMBER    67 REPLACES RECORD NUMBER    66   ********
      *
      * ORGNAM7, IX1DS9, IX2DS9, CCDF9 - Population Dose
  68  TYPE9OUT001      L-ICRP60ED      1      12      NONE
  69  TYPE9OUT002      L-ICRP60ED      1      19      NONE
  70  TYPE9OUT003      L-ICRP60ED      1      21      NONE
  71  TYPE9OUT004      L-ICRP60ED      1      26      NONE
      *
      * NXUM10=0
  72  TYP10NUMBER      0
      *
      * NXUM10, number of type10 results
  73  TYP10NUMBER      10
********   RECORD NUMBER    73 REPLACES RECORD NUMBER    72   ********
      *
      *  I1DS10, I2DS10, CCDF10 - Economic Cost
  74  TYP10OUT001      1      26      NONE
  75  TYP10OUT002      1      12      NONE
  76  TYP10OUT003      13      15      NONE
  77  TYP10OUT004      16      17      NONE
  78  TYP10OUT005      18      18      NONE
  79  TYP10OUT006      19      19      NONE
  80  TYP10OUT007      20      21      NONE
  81  TYP10OUT008      22      23      NONE
  82  TYP10OUT009      24      25      NONE
  83  TYP10OUT010      26      26      NONE
      *
      * FLAG11 - Action Distance
  84  TYP11FLAG11      .TRUE.  NONE
      *
      * NUM12=0
  85  TYP12NUMBER      0
      *
      * NUM12, number of type 12 results
  86  TYP12NUMBER      10
********   RECORD NUMBER    86 REPLACES RECORD NUMBER    85   ********
      *
      * I1DS12, I2DS12, Impacted Area/Population
  87  TYP12OUT001      1      26      NONE
  88  TYP12OUT002      1      12      NONE
  89  TYP12OUT003      13      15      NONE
  90  TYP12OUT004      16      17      NONE
  91  TYP12OUT005      18      18      NONE
  92  TYP12OUT006      19      19      NONE
  93  TYP12OUT007      20      21      NONE
  94  TYP12OUT008      22      23      NONE
  95  TYP12OUT009      24      25      NONE
  96  TYP12OUT010      26      26      NONE
      *
      * NUM13=0
  97  TYP13NUMBER      0
      *
      * NUM13, number of type 13 results
  98  TYP13NUMBER      18
********   RECORD NUMBER    98 REPLACES RECORD NUMBER    97   ********
      *
      * IRAD13, ORGN13, Max Individual Food Ingestion Dose at a Distance
  99  TYP13OUT001      12      EFFECTIVE      NONE
 100  TYP13OUT002      15      EFFECTIVE      NONE
 101  TYP13OUT003      17      EFFECTIVE      NONE
 102  TYP13OUT004      18      EFFECTIVE      NONE
 103  TYP13OUT005      19      EFFECTIVE      NONE
 104  TYP13OUT006      21      EFFECTIVE      NONE
 105  TYP13OUT007      23      EFFECTIVE      NONE
 106  TYP13OUT008      25      EFFECTIVE      NONE
 107  TYP13OUT009      26      EFFECTIVE      NONE
 108  TYP13OUT010      12      THYROID NONE
 109  TYP13OUT011      15      THYROID NONE
 110  TYP13OUT012      17      THYROID NONE
 111  TYP13OUT013      18      THYROID NONE
 112  TYP13OUT014      19      THYROID NONE
 113  TYP13OUT015      21      THYROID NONE
 114  TYP13OUT016      23      THYROID NONE
 115  TYP13OUT017      25      THYROID NONE
 116  TYP13OUT018      26      THYROID NONE
      *
      * COMIDA2_TH - use for premade comida2, dose AT or PL models
 117  BIN_FILE001      'C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Data\samp_a_FGR13GyEquivDCF.bin'
      .
******** TERMINATOR RECORD ENCOUNTERED -- END OF BASE CASE USER INPUT ********


USER INPUT PROCESSING SUMMARY - BASE CASE

NUMBER OF RECORDS READ                       =  249
NUMBER OF BLANK OR COMMENT RECORDS READ        =  131
NUMBER OF TERMINATOR RECORDS                  =    1
```

NUMBER OF RECORDS PROCESSED = 117
NUMBER OF PROCESSED RECORDS DUPLICATED = 4
NUMBER OF PROCESSED RECORDS SORTED = 113
****

READING COMIDA2 FILE: C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\10-mile evac\3.4 LowDensity\Data\samp_a_FGR13GyEquivDCF.bin
COMIDA2 binary file header =
COMIDA2 20120302 19:05:30 Version 1.13.0.1, 06/20/07

COMIDA2 descriptive title =
FGR13DF 5/13/2008 12:23:56 Version 1.03, Gy-Equivalent DCFs

Internal Dose Coefficients derived from FGR 13, EPA 402-R-99-001

COMIDA2 LASTSTOR = 9

A SITE DATA FILE IS BEING USED FOR BOTH "EARLY" AND "CHRONC"

8 CANCER EFFECTS ARE DEFINED IN THE MODEL.

INDEX	CANCER EFFECT	ORGAN	ALPHA	BETA	CFRISK	CIRISK
1	LEUKEMIA	L-RED MARR	1.000E+00	0.000E+00	1.110E-02	1.130E-02
2	BONE	L-BONE SUR	1.000E+00	0.000E+00	1.900E-04	2.710E-04
3	BREAST	L-BREAST	1.000E+00	0.000E+00	5.060E-03	1.010E-02
4	LUNG	L-LUNGS	1.000E+00	0.000E+00	1.980E-02	2.080E-02
5	THYROID	L-THYROID	1.000E+00	0.000E+00	6.480E-04	6.480E-03
6	LIVER	L-LIVER	1.000E+00	0.000E+00	3.000E-03	3.160E-03
7	COLON	L-LOWER LI	1.000E+00	0.000E+00	2.080E-02	3.780E-02
8	RESIDUAL	L-BLAD WAL	1.000E+00	0.000E+00	4.930E-02	1.690E-01

TIME OF HOTSPOT RELOCATION IS 1.4400E+04.
TIME OF NORMAL RETURN IS 2.880E+04 AND THE EMERGENCY PHASE ENDS AT 6.048E+05.

GROUNDSHINE SHIELDING FACTOR = 0.180

RESUSPENSION PROTECTION FACTOR = 0.460

BREATHING RATE (CUBIC M/S) = 2.660E-04

DISPERSION MODEL FLAG IS 3

WINDROSE PROBABILITIES BY WIND DIRECTION AND MET BIN NUMBER

BIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	0.0169	0.0099	0.0042	0.0113	0.0042	0.0000	0.0028	0.0085	0.0042	0.0099	0.0099	0.0071	0.0042	0.0071	0.0113	0.0155
2	0.0167	0.0143	0.0119	0.0167	0.0048	0.0072	0.0143	0.0167	0.0095	0.0119	0.0048	0.0167	0.0095	0.0143	0.0263	0.0286
3	0.0000	0.0000	0.0000	0.0000	0.0000	0.0111	0.0111	0.0111	0.0000	0.0222	0.0111	0.0222	0.0222	0.0000	0.0000	0.0333
4	0.0172	0.0210	0.0134	0.0095	0.0115	0.0095	0.0095	0.0076	0.0057	0.0076	0.0076	0.0115	0.0115	0.0210	0.0191	0.0134
5	0.0124	0.0212	0.0106	0.0124	0.0106	0.0071	0.0088	0.0053	0.0106	0.0053	0.0088	0.0053	0.0124	0.0071	0.0124	0.0265
6	0.0040	0.0054	0.0027	0.0040	0.0081	0.0027	0.0027	0.0054	0.0108	0.0040	0.0081	0.0108	0.0135	0.0108	0.0135	0.0148
7	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0048	0.0000	0.0096	0.0048	0.0385	0.0721
8	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
9	0.0226	0.0288	0.0309	0.0041	0.0082	0.0370	0.0123	0.0062	0.0165	0.0103	0.0185	0.0144	0.0041	0.0082	0.0103	0.0103
10	0.0282	0.0301	0.0214	0.0107	0.0136	0.0107	0.0117	0.0136	0.0097	0.0136	0.0175	0.0253	0.0224	0.0146	0.0224	0.0301
11	0.0103	0.0129	0.0078	0.0091	0.0052	0.0039	0.0039	0.0052	0.0091	0.0220	0.0272	0.0298	0.0310	0.0233	0.0336	0.0401
12	0.0085	0.0113	0.0028	0.0056	0.0056	0.0000	0.0085	0.0056	0.0113	0.0085	0.0113	0.0085	0.0282	0.0169	0.0339	0.0565
13	0.0176	0.0118	0.0412	0.0216	0.0137	0.0235	0.0314	0.0098	0.0275	0.0314	0.0255	0.0235	0.0196	0.0216	0.0275	0.0255
14	0.0053	0.0040	0.0160	0.0053	0.0093	0.0187	0.0120	0.0267	0.0293	0.0573	0.0600	0.0773	0.0960	0.0547	0.0667	0.0560
15	0.0000	0.0073	0.0000	0.0000	0.0073	0.0000	0.0073	0.0000	0.0219	0.0584	0.0803	0.0657	0.1168	0.1387	0.0949	0.1022
16	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.2500	0.2500	0.0000	0.2500	0.0000	0.2500	0.0000
17	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
18	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
19	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
20	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
21	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
22	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
23	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
24	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
25	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
26	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
27	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
28	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
29	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
30	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
31	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
32	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
33	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
34	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
35	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
36	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
37	0.0146	0.0162	0.0135	0.0094	0.0088	0.0102	0.0107	0.0100	0.0116	0.0177	0.0195	0.0210	0.0235	0.0186	0.0243	0.0288
38	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
39	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
40	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
41	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000

WINDROSE PROBABILITIES BY WIND DIRECTION AND MET BIN NUMBER

BIN	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	0.0028	0.0127	0.0071	0.0056	0.0099	0.0099	0.0071	0.0056	0.0071	0.0155	0.0099	0.0268	0.0395	0.0353	0.0226	0.0381
2	0.0048	0.0215	0.0239	0.0382	0.0549	0.0430	0.0406	0.0597	0.0358	0.0501	0.0740	0.0883	0.0644	0.0191	0.0143	0.0048
3	0.0333	0.0222	0.0000	0.0222	0.0000	0.0000	0.0222	0.0222	0.0111	0.0111	0.0000	0.0222	0.0111	0.0111	0.0000	0.0333
4	0.0153	0.0115	0.0191	0.0115	0.0248	0.0153	0.0172	0.0210	0.0248	0.0095	0.0229	0.0191	0.0248	0.0286	0.0172	0.0286
5	0.0106	0.0159	0.0177	0.0212	0.0389	0.0336	0.0319	0.0177	0.0195	0.0478	0.0602	0.0549	0.0460	0.0319	0.0283	0.0478
6	0.0175	0.0229	0.0565	0.0350	0.0579	0.0485	0.0848	0.0713	0.0646	0.0808	0.0619	0.0390	0.0565	0.0296	0.0094	0.0108
7	0.0192	0.0240	0.0337	0.0529	0.0721	0.0769	0.0577	0.1154	0.1250	0.0721	0.0962	0.0337	0.0337	0.0144	0.0000	0.0000
8	0.0313	0.0000	0.0313	0.0313	0.0313	0.0000	0.0000	0.0000	0.0000	0.4688	0.3750	0.0000	0.0000	0.0000	0.0000	0.0000
9	0.0062	0.0062	0.0144	0.0123	0.0082	0.0041	0.0123	0.0185	0.0144	0.0165	0.0082	0.0041	0.0041	0.0144	0.0041	0.0350
10	0.0233	0.0194	0.0262	0.0214	0.0204	0.0126	0.0224	0.0204	0.0117	0.0214	0.0262	0.0097	0.0175	0.0087	0.0126	0.0233
11	0.0453	0.0505	0.0582	0.0440	0.0388	0.0440	0.0310	0.0246	0.0207	0.0401	0.0285	0.0310	0.0207	0.0142	0.0246	0.0155
12	0.0311	0.0226	0.0452	0.0254	0.0678	0.0706	0.0650	0.0537	0.0537	0.0452	0.0282	0.0113	0.0113	0.0198	0.0141	0.0254
13	0.0157	0.0157	0.0137	0.0216	0.0098	0.0157	0.0216	0.0137	0.0196	0.0157	0.0333	0.0020	0.0000	0.0020	0.0020	0.0314
14	0.0307	0.0307	0.0280	0.0173	0.0293	0.0107	0.0200	0.0147	0.0227	0.0133	0.0133	0.0120	0.0133	0.0013	0.0040	0.0253
15	0.0438	0.0438	0.0146	0.0146	0.0219	0.0073	0.0073	0.0292	0.0000	0.0146	0.0000	0.0073	0.0073	0.0000	0.0073	0.0073
16	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
17	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
18	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
19	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
20	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
21	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266

22	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
23	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
24	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
25	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
26	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
27	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
28	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
29	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
30	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
31	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
32	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
33	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
34	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
35	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
36	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
37	0.0177	0.0201	0.0248	0.0215	0.0282	0.0234	0.0282	0.0263	0.0250	0.0318	0.0322	0.0236	0.0251	0.0172	0.0131	0.0249
38	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
39	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
40	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
41	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000

WINDROSE PROBABILITIES BY WIND DIRECTION AND MET BIN NUMBER

BIN	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	0.0240	0.0480	0.0452	0.0494	0.0212	0.0184	0.0169	0.0085	0.0071	0.0198	0.0240	0.0184	0.0240	0.0198	0.0226	0.0282
2	0.0000	0.0191	0.0095	0.0095	0.0024	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
3	0.0000	0.0222	0.0000	0.0444	0.0333	0.0222	0.0333	0.0111	0.0000	0.0222	0.0333	0.0222	0.0444	0.0222	0.0444	0.0556
4	0.0191	0.0439	0.0401	0.0305	0.0286	0.0134	0.0153	0.0057	0.0057	0.0153	0.0095	0.0076	0.0134	0.0038	0.0134	0.0134
5	0.0265	0.0159	0.0319	0.0071	0.0071	0.0053	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
6	0.0108	0.0121	0.0242	0.0013	0.0013	0.0000	0.0013	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
7	0.0000	0.0048	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
8	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
9	0.0123	0.0082	0.0103	0.0226	0.0062	0.0041	0.0062	0.0062	0.0082	0.0123	0.0082	0.0041	0.0267	0.0185	0.0288	0.0247
10	0.0107	0.0107	0.0165	0.0058	0.0087	0.0078	0.0029	0.0019	0.0000	0.0049	0.0019	0.0019	0.0019	0.0010	0.0019	0.0078
11	0.0091	0.0039	0.0078	0.0065	0.0039	0.0013	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
12	0.0169	0.0113	0.0056	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
13	0.0000	0.0059	0.0020	0.0176	0.0020	0.0000	0.0020	0.0000	0.0000	0.0020	0.0078	0.0020	0.0137	0.0039	0.0157	0.0216
14	0.0013	0.0000	0.0000	0.0027	0.0013	0.0013	0.0000	0.0013	0.0000	0.0000	0.0013	0.0040	0.0000	0.0000	0.0013	0.0040
15	0.0073	0.0000	0.0000	0.0073	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
16	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
17	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
18	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
19	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
20	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
21	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
22	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
23	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
24	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
25	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
26	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
27	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
28	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
29	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
30	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
31	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
32	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
33	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
34	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
35	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
36	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
37	0.0122	0.0191	0.0174	0.0132	0.0088	0.0056	0.0051	0.0034	0.0023	0.0058	0.0064	0.0047	0.0072	0.0047	0.0084	0.0112
38	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
39	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
40	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
41	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000

WINDROSE PROBABILITIES BY WIND DIRECTION AND MET BIN NUMBER

BIN	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	0.0113	0.0311	0.0212	0.0099	0.0198	0.0071	0.0169	0.0099	0.0127	0.0099	0.0099	0.0141	0.0085	0.0155	0.0085	0.0155
2	0.0000	0.0000	0.0000	0.0000	0.0000	0.0024	0.0024	0.0000	0.0000	0.0024	0.0000	0.0048	0.0048	0.0191	0.0358	0.0263
3	0.0000	0.0444	0.0222	0.0111	0.0111	0.0111	0.0333	0.0111	0.0000	0.0111	0.0222	0.0111	0.0222	0.0000	0.0111	0.0000
4	0.0172	0.0057	0.0019	0.0095	0.0095	0.0115	0.0172	0.0153	0.0115	0.0115	0.0172	0.0191	0.0191	0.0153	0.0134	0.0191
5	0.0000	0.0000	0.0000	0.0000	0.0000	0.0053	0.0088	0.0000	0.0159	0.0265	0.0212	0.0230	0.0283	0.0177	0.0283	0.0301
6	0.0000	0.0000	0.0000	0.0000	0.0000	0.0013	0.0013	0.0027	0.0000	0.0081	0.0013	0.0013	0.0094	0.0202	0.0148	0.0202
7	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0096	0.0048	0.0144	0.0096
8	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0313	0.0000	0.0000
9	0.0309	0.0267	0.0206	0.0185	0.0144	0.0165	0.0309	0.0226	0.0103	0.0165	0.0412	0.0226	0.0267	0.0267	0.0309	0.0103
10	0.0049	0.0175	0.0107	0.0107	0.0097	0.0117	0.0155	0.0214	0.0204	0.0204	0.0292	0.0253	0.0311	0.0282	0.0311	0.0330
11	0.0000	0.0026	0.0000	0.0000	0.0026	0.0026	0.0013	0.0013	0.0103	0.0181	0.0194	0.0194	0.0194	0.0246	0.0246	0.0155
12	0.0000	0.0000	0.0000	0.0000	0.0000	0.0028	0.0028	0.0000	0.0056	0.0028	0.0028	0.0085	0.0141	0.0141	0.0339	0.0650
13	0.0451	0.0353	0.0431	0.0059	0.0608	0.0118	0.0255	0.0039	0.0000	0.0078	0.0118	0.0235	0.0059	0.0059	0.0118	0.0000
14	0.0040	0.0133	0.0093	0.0067	0.0107	0.0053	0.0093	0.0040	0.0013	0.0093	0.0000	0.0053	0.0067	0.0053	0.0067	0.0027
15	0.0000	0.0000	0.0000	0.0000	0.0073	0.0000	0.0000	0.0000	0.0000	0.0146	0.0000	0.0000	0.0146	0.0073	0.0073	0.0073
16	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
17	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
18	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
19	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
20	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
21	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
22	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
23	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
24	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
25	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
26	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
27	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
28	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
29	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
30	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
31	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
32	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
33	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
34	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
35	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
36	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
37	0.0104	0.0116	0.0095	0.0073	0.0126	0.0075	0.0121	0.0110	0.0112	0.0155	0.0161	0.0170	0.0183	0.0188	0.0227	0.0213
38	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
39	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
40	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
41	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000

Processing a Site Data File with Header: SECPOP2000 Version: 3.13.1 MACCS2 Formatted Site: Peach Bottom Census: C:\Prog
Lat: 39d45'32" Long: 76d16' 9" Population multiplier: 1.1051 Economic multip

THIS PROGRAM CURRENTLY ALLOWS THE GENERATION OF UP TO 3394 RESULTS

YOU HAVE REQUESTED 108 RESULTS FROM "EARLY" COMPOSED OF:

38 RESULTS OF TYPE 1
1 RESULTS OF TYPE 2
3 RESULTS OF TYPE 3
0 RESULTS OF TYPE 4
4 RESULTS OF TYPE 5
0 RESULTS OF TYPE 6
0 RESULTS OF TYPE 7
17 RESULTS OF TYPE 8
26 RESULTS OF TYPE A
0 RESULTS OF TYPE B
3 RESULTS OF TYPE C
16 RESULTS OF TYPE D
0 RESULTS OF TYPE E

YOU HAVE REQUESTED 304 RESULTS FROM "CHRONC" COMPOSED OF:

68 RESULTS OF TYPE 9
130 RESULTS OF TYPE 10
8 RESULTS OF TYPE 11
80 RESULTS OF TYPE 12
18 RESULTS OF TYPE 13

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TRIAL    DAY   PERIOD    BIN   PRBMET
    1    152      3        9  1.13E-03
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WARNING!!

THE TOTAL RELEASE DURATION EXCEEDS 2 HOURS.

THIS MAY CAUSE ERRONEOUS RESULTS TO BE PRODUCED

WHEN USING THE Regulatory Guide 1.145 model.

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For Julian Day 152, selecting COMIDA2 results # 4 of 9
    2    152     10        1  1.14E-03
For Julian Day 152, selecting COMIDA2 results # 4 of 9
    3    152     15       36  1.43E-04
For Julian Day 152, selecting COMIDA2 results # 4 of 9
    4    152     16       35  1.14E-04
For Julian Day 152, selecting COMIDA2 results # 4 of 9
    5    152     17       34  1.14E-04
For Julian Day 152, selecting COMIDA2 results # 4 of 9
    6    152     18       32  3.23E-04
For Julian Day 152, selecting COMIDA2 results # 4 of 9
    7    153      1       10  1.14E-03
For Julian Day 153, selecting COMIDA2 results # 4 of 9
    8    153      6       36  1.43E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
    9    153      7       36  1.43E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
   10    153      9       36  1.43E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
   11    153     10       35  1.14E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
   12    153     11       35  1.14E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
   13    153     12       34  1.14E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
   14    153     13       34  1.14E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
   15    154     12        6  1.15E-03
For Julian Day 154, selecting COMIDA2 results # 4 of 9
   16    154     23       10  1.14E-03
For Julian Day 154, selecting COMIDA2 results # 4 of 9
   17    154     24       11  1.15E-03
For Julian Day 154, selecting COMIDA2 results # 4 of 9
   18    155     10        4  1.15E-03
For Julian Day 155, selecting COMIDA2 results # 4 of 9
   19    155     23       11  1.15E-03
For Julian Day 155, selecting COMIDA2 results # 4 of 9
   20    156      7        5  1.13E-03
For Julian Day 156, selecting COMIDA2 results # 4 of 9
   21    156     19       10  1.14E-03
For Julian Day 156, selecting COMIDA2 results # 4 of 9
   22    156     24        9  1.13E-03
For Julian Day 156, selecting COMIDA2 results # 4 of 9
   23    157     12        1  1.14E-03
For Julian Day 157, selecting COMIDA2 results # 4 of 9
   24    157     18        3  8.56E-04
For Julian Day 157, selecting COMIDA2 results # 4 of 9
   25    158     10       21  1.13E-03
For Julian Day 158, selecting COMIDA2 results # 4 of 9
   26    158     14       25  1.52E-04
For Julian Day 158, selecting COMIDA2 results # 4 of 9
   27    158     15       24  1.14E-04
For Julian Day 158, selecting COMIDA2 results # 4 of 9
   28    158     19       12  1.15E-03
For Julian Day 158, selecting COMIDA2 results # 4 of 9
   29    159      1       17  1.14E-03
For Julian Day 159, selecting COMIDA2 results # 4 of 9
   30    159      4       14  1.14E-03
For Julian Day 159, selecting COMIDA2 results # 4 of 9
   31    159     10        4  1.15E-03
For Julian Day 159, selecting COMIDA2 results # 4 of 9
   32    159     17       26  2.38E-04
For Julian Day 159, selecting COMIDA2 results # 4 of 9
   33    159     18       25  1.52E-04
For Julian Day 159, selecting COMIDA2 results # 4 of 9
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34 159 19 24 1.14E-04
For Julian Day 159, selecting COMIDA2 results # 4 of 9
35 159 20 22 1.09E-03
For Julian Day 159, selecting COMIDA2 results # 4 of 9
36 159 24 18 5.99E-04
For Julian Day 159, selecting COMIDA2 results # 4 of 9
37 160 6 14 1.14E-03
For Julian Day 160, selecting COMIDA2 results # 4 of 9
38 160 10 5 1.13E-03
For Julian Day 160, selecting COMIDA2 results # 4 of 9
39 160 12 20 1.12E-03
For Julian Day 160, selecting COMIDA2 results # 4 of 9
40 160 13 19 1.11E-03
For Julian Day 160, selecting COMIDA2 results # 4 of 9
41 161 1 11 1.15E-03
For Julian Day 161, selecting COMIDA2 results # 4 of 9
42 161 12 2 1.14E-03
For Julian Day 161, selecting COMIDA2 results # 4 of 9
43 161 17 6 1.15E-03
For Julian Day 161, selecting COMIDA2 results # 4 of 9
44 161 20 11 1.15E-03
For Julian Day 161, selecting COMIDA2 results # 4 of 9
45 161 23 15 1.12E-03
For Julian Day 161, selecting COMIDA2 results # 4 of 9
46 163 7 10 1.14E-03
For Julian Day 163, selecting COMIDA2 results # 4 of 9
47 163 12 1 1.14E-03
For Julian Day 163, selecting COMIDA2 results # 4 of 9
48 164 3 14 1.14E-03
For Julian Day 164, selecting COMIDA2 results # 4 of 9
49 165 11 1 1.14E-03
For Julian Day 165, selecting COMIDA2 results # 4 of 9
50 165 12 1 1.14E-03
For Julian Day 165, selecting COMIDA2 results # 4 of 9

TRIAL DAY PERIOD BIN PRBMET
51 165 21 11 1.15E-03
For Julian Day 165, selecting COMIDA2 results # 4 of 9
52 166 3 11 1.15E-03
For Julian Day 166, selecting COMIDA2 results # 4 of 9
53 166 12 2 1.14E-03
For Julian Day 166, selecting COMIDA2 results # 4 of 9
54 167 16 4 1.15E-03
For Julian Day 167, selecting COMIDA2 results # 5 of 9
55 167 21 14 1.14E-03
For Julian Day 167, selecting COMIDA2 results # 5 of 9
56 167 24 13 1.14E-03
For Julian Day 167, selecting COMIDA2 results # 5 of 9
57 168 6 13 1.14E-03
For Julian Day 168, selecting COMIDA2 results # 5 of 9
58 169 9 4 1.15E-03
For Julian Day 169, selecting COMIDA2 results # 5 of 9
59 169 14 1 1.14E-03
For Julian Day 169, selecting COMIDA2 results # 5 of 9
60 169 18 6 1.15E-03
For Julian Day 169, selecting COMIDA2 results # 5 of 9
61 169 23 10 1.14E-03
For Julian Day 169, selecting COMIDA2 results # 5 of 9
62 170 13 26 2.38E-04
For Julian Day 170, selecting COMIDA2 results # 5 of 9
63 170 15 24 1.14E-04
For Julian Day 170, selecting COMIDA2 results # 5 of 9
64 170 19 10 1.14E-03
For Julian Day 170, selecting COMIDA2 results # 5 of 9
65 171 7 17 1.14E-03
For Julian Day 171, selecting COMIDA2 results # 5 of 9
66 171 13 5 1.13E-03
For Julian Day 171, selecting COMIDA2 results # 5 of 9
67 171 21 14 1.14E-03
For Julian Day 171, selecting COMIDA2 results # 5 of 9
68 172 7 9 1.13E-03
For Julian Day 172, selecting COMIDA2 results # 5 of 9
69 173 2 21 1.13E-03
For Julian Day 173, selecting COMIDA2 results # 5 of 9
70 173 9 10 1.14E-03
For Julian Day 173, selecting COMIDA2 results # 5 of 9
71 174 3 14 1.14E-03
For Julian Day 174, selecting COMIDA2 results # 5 of 9
72 174 8 4 1.15E-03
For Julian Day 174, selecting COMIDA2 results # 5 of 9
73 174 11 1 1.14E-03
For Julian Day 174, selecting COMIDA2 results # 5 of 9
74 174 12 5 1.13E-03
For Julian Day 174, selecting COMIDA2 results # 5 of 9
75 174 22 19 1.11E-03
For Julian Day 174, selecting COMIDA2 results # 5 of 9
76 175 9 36 1.43E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
77 175 10 36 1.43E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
78 175 11 35 1.14E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
79 175 12 35 1.14E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
80 175 13 34 1.14E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
81 175 16 32 3.23E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
82 175 17 27 3.71E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
83 175 20 20 1.12E-03
For Julian Day 175, selecting COMIDA2 results # 5 of 9
84 175 24 17 1.14E-03
For Julian Day 175, selecting COMIDA2 results # 5 of 9
85 176 2 10 1.14E-03
For Julian Day 176, selecting COMIDA2 results # 5 of 9
86 176 4 26 2.38E-04
For Julian Day 176, selecting COMIDA2 results # 5 of 9

87 176 7 25 1.52E-04
For Julian Day 176, selecting COMIDA2 results # 5 of 9
88 176 9 24 1.14E-04
For Julian Day 176, selecting COMIDA2 results # 5 of 9
89 176 22 27 3.71E-04
For Julian Day 176, selecting COMIDA2 results # 5 of 9
90 177 2 32 3.23E-04
For Julian Day 177, selecting COMIDA2 results # 5 of 9
91 177 5 32 3.23E-04
For Julian Day 177, selecting COMIDA2 results # 5 of 9
92 177 8 17 1.14E-03
For Julian Day 177, selecting COMIDA2 results # 5 of 9
93 177 18 32 3.23E-04
For Julian Day 177, selecting COMIDA2 results # 5 of 9
94 177 21 25 1.52E-04
For Julian Day 177, selecting COMIDA2 results # 5 of 9
95 177 24 25 1.52E-04
For Julian Day 177, selecting COMIDA2 results # 5 of 9
96 178 3 32 3.23E-04
For Julian Day 178, selecting COMIDA2 results # 5 of 9
97 178 6 22 1.09E-03
For Julian Day 178, selecting COMIDA2 results # 5 of 9
98 178 7 17 1.14E-03
For Julian Day 178, selecting COMIDA2 results # 5 of 9
99 179 3 27 3.71E-04
For Julian Day 179, selecting COMIDA2 results # 5 of 9
100 179 16 1 1.14E-03
For Julian Day 179, selecting COMIDA2 results # 5 of 9

TRIAL DAY PERIOD BIN PRBMET
101 179 19 11 1.15E-03
For Julian Day 179, selecting COMIDA2 results # 5 of 9
102 179 22 20 1.12E-03
For Julian Day 179, selecting COMIDA2 results # 5 of 9
103 180 1 19 1.11E-03
For Julian Day 180, selecting COMIDA2 results # 5 of 9
104 180 13 4 1.15E-03
For Julian Day 180, selecting COMIDA2 results # 5 of 9
105 180 16 31 1.14E-04
For Julian Day 180, selecting COMIDA2 results # 5 of 9
106 180 17 30 1.14E-04
For Julian Day 180, selecting COMIDA2 results # 5 of 9
107 180 18 30 1.14E-04
For Julian Day 180, selecting COMIDA2 results # 5 of 9
108 180 19 29 1.14E-04
For Julian Day 180, selecting COMIDA2 results # 5 of 9
109 181 20 14 1.14E-03
For Julian Day 181, selecting COMIDA2 results # 5 of 9
110 182 3 13 1.14E-03
For Julian Day 182, selecting COMIDA2 results # 5 of 9
111 182 7 9 1.13E-03
For Julian Day 182, selecting COMIDA2 results # 5 of 9
112 182 17 5 1.13E-03
For Julian Day 182, selecting COMIDA2 results # 5 of 9
113 182 23 10 1.14E-03
For Julian Day 182, selecting COMIDA2 results # 5 of 9
114 183 3 10 1.14E-03
For Julian Day 183, selecting COMIDA2 results # 5 of 9
115 183 18 11 1.15E-03
For Julian Day 183, selecting COMIDA2 results # 5 of 9
116 185 18 9 1.13E-03
For Julian Day 185, selecting COMIDA2 results # 5 of 9
117 185 24 21 1.13E-03
For Julian Day 185, selecting COMIDA2 results # 5 of 9
118 186 4 17 1.14E-03
For Julian Day 186, selecting COMIDA2 results # 5 of 9
119 186 7 10 1.14E-03
For Julian Day 186, selecting COMIDA2 results # 5 of 9
120 186 11 5 1.13E-03
For Julian Day 186, selecting COMIDA2 results # 5 of 9
121 187 2 32 3.23E-04
For Julian Day 187, selecting COMIDA2 results # 5 of 9
122 187 4 17 1.14E-03
For Julian Day 187, selecting COMIDA2 results # 5 of 9
123 187 5 10 1.14E-03
For Julian Day 187, selecting COMIDA2 results # 5 of 9
124 188 17 4 1.15E-03
For Julian Day 188, selecting COMIDA2 results # 5 of 9
125 188 23 14 1.14E-03
For Julian Day 188, selecting COMIDA2 results # 5 of 9
126 189 14 4 1.15E-03
For Julian Day 189, selecting COMIDA2 results # 5 of 9
127 189 17 10 1.14E-03
For Julian Day 189, selecting COMIDA2 results # 5 of 9
128 189 21 14 1.14E-03
For Julian Day 189, selecting COMIDA2 results # 5 of 9
129 190 17 6 1.15E-03
For Julian Day 190, selecting COMIDA2 results # 5 of 9
130 191 2 9 1.13E-03
For Julian Day 191, selecting COMIDA2 results # 5 of 9
131 192 4 10 1.14E-03
For Julian Day 192, selecting COMIDA2 results # 6 of 9
132 192 6 9 1.13E-03
For Julian Day 192, selecting COMIDA2 results # 6 of 9
133 192 10 1 1.14E-03
For Julian Day 192, selecting COMIDA2 results # 6 of 9
134 192 15 5 1.13E-03
For Julian Day 192, selecting COMIDA2 results # 6 of 9
135 192 18 5 1.13E-03
For Julian Day 192, selecting COMIDA2 results # 6 of 9
136 193 18 26 2.38E-04
For Julian Day 193, selecting COMIDA2 results # 6 of 9
137 193 20 25 1.52E-04
For Julian Day 193, selecting COMIDA2 results # 6 of 9
138 193 21 24 1.14E-04
For Julian Day 193, selecting COMIDA2 results # 6 of 9
139 195 1 19 1.11E-03
For Julian Day 195, selecting COMIDA2 results # 6 of 9

140 195 6 13 1.14E-03
For Julian Day 195, selecting COMIDA2 results # 6 of 9
141 195 10 1 1.14E-03
For Julian Day 195, selecting COMIDA2 results # 6 of 9
142 196 1 10 1.14E-03
For Julian Day 196, selecting COMIDA2 results # 6 of 9
143 196 6 10 1.14E-03
For Julian Day 196, selecting COMIDA2 results # 6 of 9
144 196 7 36 1.43E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
145 196 9 36 1.43E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
146 196 10 35 1.14E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
147 196 11 35 1.14E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
148 196 12 34 1.14E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
149 196 15 18 5.99E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
150 196 18 3 8.56E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9

TRIAL DAY PERIOD BIN PRBMET
151 197 2 14 1.14E-03
For Julian Day 197, selecting COMIDA2 results # 6 of 9
152 197 7 11 1.15E-03
For Julian Day 197, selecting COMIDA2 results # 6 of 9
153 197 15 5 1.13E-03
For Julian Day 197, selecting COMIDA2 results # 6 of 9
154 197 17 11 1.15E-03
For Julian Day 197, selecting COMIDA2 results # 6 of 9
155 198 6 14 1.14E-03
For Julian Day 198, selecting COMIDA2 results # 6 of 9
156 198 13 14 1.14E-03
For Julian Day 198, selecting COMIDA2 results # 6 of 9
157 199 7 9 1.13E-03
For Julian Day 199, selecting COMIDA2 results # 6 of 9
158 199 14 36 1.43E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
159 199 15 36 1.43E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
160 199 16 36 1.43E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
161 199 17 35 1.14E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
162 199 18 34 1.14E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
163 199 24 20 1.12E-03
For Julian Day 199, selecting COMIDA2 results # 6 of 9
164 200 15 4 1.15E-03
For Julian Day 200, selecting COMIDA2 results # 6 of 9
165 201 8 4 1.15E-03
For Julian Day 201, selecting COMIDA2 results # 6 of 9
166 201 24 10 1.14E-03
For Julian Day 201, selecting COMIDA2 results # 6 of 9
167 202 2 9 1.13E-03
For Julian Day 202, selecting COMIDA2 results # 6 of 9
168 202 8 3 8.56E-04
For Julian Day 202, selecting COMIDA2 results # 6 of 9
169 202 18 10 1.14E-03
For Julian Day 202, selecting COMIDA2 results # 6 of 9
170 202 21 14 1.14E-03
For Julian Day 202, selecting COMIDA2 results # 6 of 9
171 203 11 5 1.13E-03
For Julian Day 203, selecting COMIDA2 results # 6 of 9
172 203 17 26 2.38E-04
For Julian Day 203, selecting COMIDA2 results # 6 of 9
173 203 19 22 1.09E-03
For Julian Day 203, selecting COMIDA2 results # 6 of 9
174 203 20 18 5.99E-04
For Julian Day 203, selecting COMIDA2 results # 6 of 9
175 204 4 10 1.14E-03
For Julian Day 204, selecting COMIDA2 results # 6 of 9
176 204 14 1 1.14E-03
For Julian Day 204, selecting COMIDA2 results # 6 of 9
177 204 17 4 1.15E-03
For Julian Day 204, selecting COMIDA2 results # 6 of 9
178 205 15 4 1.15E-03
For Julian Day 205, selecting COMIDA2 results # 6 of 9
179 206 5 10 1.14E-03
For Julian Day 206, selecting COMIDA2 results # 6 of 9
180 206 15 2 1.14E-03
For Julian Day 206, selecting COMIDA2 results # 6 of 9
181 206 20 11 1.15E-03
For Julian Day 206, selecting COMIDA2 results # 6 of 9
182 207 6 9 1.13E-03
For Julian Day 207, selecting COMIDA2 results # 6 of 9
183 207 12 1 1.14E-03
For Julian Day 207, selecting COMIDA2 results # 6 of 9
184 207 17 5 1.13E-03
For Julian Day 207, selecting COMIDA2 results # 6 of 9
185 208 15 6 1.15E-03
For Julian Day 208, selecting COMIDA2 results # 6 of 9
186 209 2 9 1.13E-03
For Julian Day 209, selecting COMIDA2 results # 6 of 9
187 209 11 26 2.38E-04
For Julian Day 209, selecting COMIDA2 results # 6 of 9
188 209 12 25 1.52E-04
For Julian Day 209, selecting COMIDA2 results # 6 of 9
189 209 23 14 1.14E-03
For Julian Day 209, selecting COMIDA2 results # 6 of 9
190 209 24 15 1.12E-03
For Julian Day 209, selecting COMIDA2 results # 6 of 9
191 210 4 14 1.14E-03
For Julian Day 210, selecting COMIDA2 results # 6 of 9
192 210 16 4 1.15E-03
For Julian Day 210, selecting COMIDA2 results # 6 of 9

193 211 6 10 1.14E-03
For Julian Day 211, selecting COMIDA2 results # 6 of 9
194 211 21 13 1.14E-03
For Julian Day 211, selecting COMIDA2 results # 6 of 9
195 211 22 14 1.14E-03
For Julian Day 211, selecting COMIDA2 results # 6 of 9
196 212 11 26 2.38E-04
For Julian Day 212, selecting COMIDA2 results # 6 of 9
197 212 12 25 1.52E-04
For Julian Day 212, selecting COMIDA2 results # 6 of 9
198 212 13 24 1.14E-04
For Julian Day 212, selecting COMIDA2 results # 6 of 9
199 212 14 23 1.14E-04
For Julian Day 212, selecting COMIDA2 results # 6 of 9
200 212 19 10 1.14E-03
For Julian Day 212, selecting COMIDA2 results # 6 of 9

TRIAL DAY PERIOD BIN PRBMET
201 212 20 9 1.13E-03
For Julian Day 212, selecting COMIDA2 results # 6 of 9
202 213 9 3 8.56E-04
For Julian Day 213, selecting COMIDA2 results # 6 of 9
203 213 13 5 1.13E-03
For Julian Day 213, selecting COMIDA2 results # 6 of 9
204 214 2 14 1.14E-03
For Julian Day 214, selecting COMIDA2 results # 6 of 9
205 214 3 14 1.14E-03
For Julian Day 214, selecting COMIDA2 results # 6 of 9
206 214 14 5 1.13E-03
For Julian Day 214, selecting COMIDA2 results # 6 of 9
207 215 18 11 1.15E-03
For Julian Day 215, selecting COMIDA2 results # 6 of 9
208 215 20 14 1.14E-03
For Julian Day 215, selecting COMIDA2 results # 6 of 9
209 215 21 13 1.14E-03
For Julian Day 215, selecting COMIDA2 results # 6 of 9
210 216 12 6 1.15E-03
For Julian Day 216, selecting COMIDA2 results # 6 of 9
211 217 2 14 1.14E-03
For Julian Day 217, selecting COMIDA2 results # 6 of 9
212 217 8 4 1.15E-03
For Julian Day 217, selecting COMIDA2 results # 6 of 9
213 217 11 1 1.14E-03
For Julian Day 217, selecting COMIDA2 results # 6 of 9
214 218 2 14 1.14E-03
For Julian Day 218, selecting COMIDA2 results # 6 of 9
215 218 6 10 1.14E-03
For Julian Day 218, selecting COMIDA2 results # 6 of 9
216 218 10 1 1.14E-03
For Julian Day 218, selecting COMIDA2 results # 6 of 9
217 218 23 31 1.14E-04
For Julian Day 218, selecting COMIDA2 results # 6 of 9
218 218 24 30 1.14E-04
For Julian Day 218, selecting COMIDA2 results # 6 of 9
219 219 1 29 1.14E-04
For Julian Day 219, selecting COMIDA2 results # 6 of 9
220 219 2 28 1.14E-04
For Julian Day 219, selecting COMIDA2 results # 6 of 9
221 219 4 17 1.14E-03
For Julian Day 219, selecting COMIDA2 results # 6 of 9
222 219 14 1 1.14E-03
For Julian Day 219, selecting COMIDA2 results # 6 of 9
223 220 1 10 1.14E-03
For Julian Day 220, selecting COMIDA2 results # 6 of 9
224 220 5 12 1.15E-03
For Julian Day 220, selecting COMIDA2 results # 6 of 9
225 220 14 5 1.13E-03
For Julian Day 220, selecting COMIDA2 results # 6 of 9
226 220 22 11 1.15E-03
For Julian Day 220, selecting COMIDA2 results # 6 of 9
227 221 18 4 1.15E-03
For Julian Day 221, selecting COMIDA2 results # 6 of 9
228 221 21 14 1.14E-03
For Julian Day 221, selecting COMIDA2 results # 6 of 9
229 222 20 10 1.14E-03
For Julian Day 222, selecting COMIDA2 results # 7 of 9
230 222 23 9 1.13E-03
For Julian Day 222, selecting COMIDA2 results # 7 of 9
231 222 24 14 1.14E-03
For Julian Day 222, selecting COMIDA2 results # 7 of 9
232 223 5 12 1.15E-03
For Julian Day 223, selecting COMIDA2 results # 7 of 9
233 223 12 1 1.14E-03
For Julian Day 223, selecting COMIDA2 results # 7 of 9
234 224 24 14 1.14E-03
For Julian Day 224, selecting COMIDA2 results # 7 of 9
235 225 15 1 1.14E-03
For Julian Day 225, selecting COMIDA2 results # 7 of 9
236 225 19 10 1.14E-03
For Julian Day 225, selecting COMIDA2 results # 7 of 9
237 226 23 11 1.15E-03
For Julian Day 226, selecting COMIDA2 results # 7 of 9
238 227 8 10 1.14E-03
For Julian Day 227, selecting COMIDA2 results # 7 of 9
239 227 11 1 1.14E-03
For Julian Day 227, selecting COMIDA2 results # 7 of 9
240 228 1 14 1.14E-03
For Julian Day 228, selecting COMIDA2 results # 7 of 9
241 228 3 15 1.12E-03
For Julian Day 228, selecting COMIDA2 results # 7 of 9
242 228 9 4 1.15E-03
For Julian Day 228, selecting COMIDA2 results # 7 of 9
243 228 15 1 1.14E-03
For Julian Day 228, selecting COMIDA2 results # 7 of 9
244 229 6 10 1.14E-03
For Julian Day 229, selecting COMIDA2 results # 7 of 9
245 229 15 1 1.14E-03
For Julian Day 229, selecting COMIDA2 results # 7 of 9

246 229 20 14 1.14E-03
For Julian Day 229, selecting COMIDA2 results # 7 of 9
247 230 7 9 1.13E-03
For Julian Day 230, selecting COMIDA2 results # 7 of 9
248 230 17 4 1.15E-03
For Julian Day 230, selecting COMIDA2 results # 7 of 9
249 231 8 4 1.15E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9
250 231 10 1 1.14E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9

TRIAL DAY PERIOD BIN PRBMET
251 231 15 6 1.15E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9
252 231 18 5 1.13E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9
253 231 21 11 1.15E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9
254 233 4 11 1.15E-03
For Julian Day 233, selecting COMIDA2 results # 7 of 9
255 233 18 10 1.14E-03
For Julian Day 233, selecting COMIDA2 results # 7 of 9
256 233 24 14 1.14E-03
For Julian Day 233, selecting COMIDA2 results # 7 of 9
257 234 6 14 1.14E-03
For Julian Day 234, selecting COMIDA2 results # 7 of 9
258 234 11 1 1.14E-03
For Julian Day 234, selecting COMIDA2 results # 7 of 9
259 234 14 1 1.14E-03
For Julian Day 234, selecting COMIDA2 results # 7 of 9
260 235 16 5 1.13E-03
For Julian Day 235, selecting COMIDA2 results # 7 of 9
261 236 2 9 1.13E-03
For Julian Day 236, selecting COMIDA2 results # 7 of 9
262 236 3 13 1.14E-03
For Julian Day 236, selecting COMIDA2 results # 7 of 9
263 236 11 1 1.14E-03
For Julian Day 236, selecting COMIDA2 results # 7 of 9
264 236 17 10 1.14E-03
For Julian Day 236, selecting COMIDA2 results # 7 of 9
265 237 15 1 1.14E-03
For Julian Day 237, selecting COMIDA2 results # 7 of 9
266 238 8 4 1.15E-03
For Julian Day 238, selecting COMIDA2 results # 7 of 9
267 239 4 32 3.23E-04
For Julian Day 239, selecting COMIDA2 results # 7 of 9
268 239 7 21 1.13E-03
For Julian Day 239, selecting COMIDA2 results # 7 of 9
269 239 10 17 1.14E-03
For Julian Day 239, selecting COMIDA2 results # 7 of 9
270 239 18 11 1.15E-03
For Julian Day 239, selecting COMIDA2 results # 7 of 9
271 240 6 9 1.13E-03
For Julian Day 240, selecting COMIDA2 results # 7 of 9
272 240 9 4 1.15E-03
For Julian Day 240, selecting COMIDA2 results # 7 of 9
273 240 15 5 1.13E-03
For Julian Day 240, selecting COMIDA2 results # 7 of 9
274 240 18 10 1.14E-03
For Julian Day 240, selecting COMIDA2 results # 7 of 9
275 241 15 31 1.14E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
276 241 16 31 1.14E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
277 241 17 30 1.14E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
278 241 18 29 1.14E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
279 241 19 27 3.71E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
280 241 20 27 3.71E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
281 241 22 20 1.12E-03
For Julian Day 241, selecting COMIDA2 results # 7 of 9
282 242 16 10 1.14E-03
For Julian Day 242, selecting COMIDA2 results # 7 of 9
283 242 19 9 1.13E-03
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284 243 5 10 1.14E-03
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285 243 24 9 1.13E-03
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286 244 4 21 1.13E-03
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287 244 20 22 1.09E-03
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288 245 3 17 1.14E-03
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289 245 11 17 1.14E-03
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290 245 20 19 1.11E-03
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291 246 2 14 1.14E-03
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292 246 8 3 8.56E-04
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293 246 18 10 1.14E-03
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294 246 23 13 1.14E-03
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295 247 10 1 1.14E-03
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296 247 23 20 1.12E-03
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297 248 7 17 1.14E-03
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298 248 8 27 3.71E-04
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299 248 9 32 3.23E-04
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300 249 2 10 1.14E-03
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TRIAL DAY PERIOD BIN PRBMET
301 249 15 4 1.15E-03
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302 249 16 4 1.15E-03
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303 250 9 3 8.56E-04
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304 250 18 9 1.13E-03
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305 250 19 13 1.14E-03
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306 251 14 26 2.38E-04
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307 251 16 24 1.14E-04
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308 251 20 14 1.14E-03
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309 252 2 13 1.14E-03
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310 252 22 13 1.14E-03
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311 253 9 4 1.15E-03
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312 253 21 19 1.11E-03
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313 253 22 18 5.99E-04
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314 254 7 9 1.13E-03
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315 254 12 1 1.14E-03
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316 254 24 13 1.14E-03
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317 255 4 9 1.13E-03
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318 255 7 10 1.14E-03
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319 255 18 10 1.14E-03
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320 256 10 21 1.13E-03
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321 256 11 20 1.12E-03
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322 256 12 19 1.11E-03
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323 256 14 17 1.14E-03
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324 256 22 11 1.15E-03
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325 256 24 26 2.38E-04
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326 257 1 25 1.52E-04
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327 257 2 23 1.14E-04
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328 258 2 10 1.14E-03
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329 258 15 17 1.14E-03
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330 258 21 11 1.15E-03
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331 259 19 10 1.14E-03
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332 259 23 14 1.14E-03
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333 260 3 14 1.14E-03
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334 260 11 5 1.13E-03
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335 261 19 15 1.12E-03
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336 261 21 14 1.14E-03
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337 262 13 5 1.13E-03
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338 262 15 4 1.15E-03
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339 263 23 11 1.15E-03
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345 266 12 2 1.14E-03
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346 266 17 4 1.15E-03
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347 266 21 10 1.14E-03
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350 267 18 11 1.15E-03
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TRIAL DAY PERIOD BIN PRBMET
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353 269 6 13 1.14E-03
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354 269 13 5 1.13E-03
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356 269 18 9 1.13E-03
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357 269 22 13 1.14E-03
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359 271 16 7 1.13E-03
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360 271 17 30 1.14E-04
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361 271 18 27 3.71E-04
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362 271 21 22 1.09E-03
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363 272 9 6 1.15E-03
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364 272 22 14 1.14E-03
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369 274 10 1 1.14E-03
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370 274 14 5 1.13E-03
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371 274 15 10 1.14E-03
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372 274 22 15 1.12E-03
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373 275 3 14 1.14E-03
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374 276 4 9 1.13E-03
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375 276 12 10 1.14E-03
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377 276 24 13 1.14E-03
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378 277 5 14 1.14E-03
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379 278 2 5 1.13E-03
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380 278 4 6 1.15E-03
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381 278 8 2 1.14E-03
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382 278 24 18 5.99E-04
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383 279 7 17 1.14E-03
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384 279 11 17 1.14E-03
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385 280 5 2 1.14E-03
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386 280 11 1 1.14E-03
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387 281 1 1 1.14E-03
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388 281 19 1 1.14E-03
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389 282 2 1 1.14E-03
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390 282 8 1 1.14E-03
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391 282 22 20 1.12E-03
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392 282 24 19 1.11E-03
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393 283 8 1 1.14E-03
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394 283 17 9 1.13E-03
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395 283 18 13 1.14E-03
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396 284 6 9 1.13E-03
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397 284 19 18 5.99E-04
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398 284 21 17 1.14E-03
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TRIAL DAY PERIOD BIN PRBMET
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407 289 24 11 1.15E-03
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408 290 1 11 1.15E-03
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409 290 13 27 3.71E-04
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414 292 7 26 2.38E-04
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428 296 16 6 1.15E-03
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430 297 18 6 1.15E-03
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438 299 24 13 1.14E-03
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439 300 1 9 1.13E-03
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440 300 7 21 1.13E-03
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441 300 8 21 1.13E-03
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442 300 20 22 1.09E-03
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444 302 17 12 1.15E-03
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446 305 1 19 1.11E-03
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447 305 7 13 1.14E-03
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448 306 9 6 1.15E-03
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449 306 16 11 1.15E-03
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450 306 20 11 1.15E-03
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TRIAL DAY PERIOD BIN PRBMET
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453 308 4 15 1.12E-03
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454 308 10 2 1.14E-03
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455 308 24 14 1.14E-03
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456 309 9 3 8.56E-04

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457 309 10 4 1.15E-03
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458 309 13 1 1.14E-03
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464 311 16 20 1.12E-03
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466 312 9 22 1.09E-03
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469 314 4 14 1.14E-03
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470 314 15 3 8.56E-04
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471 314 21 13 1.14E-03
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472 315 19 10 1.14E-03
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475 316 12 7 1.13E-03
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477 316 17 7 1.13E-03
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TRIAL DAY PERIOD BIN PRBMET
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TRIAL DAY PERIOD BIN PRBMET
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562 349 7 9 1.13E-03

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571 352 24 11 1.15E-03
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572 353 7 10 1.14E-03
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573 353 11 5 1.13E-03
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574 353 15 6 1.15E-03
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575 353 18 11 1.15E-03
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576 354 4 14 1.14E-03
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577 355 7 20 1.12E-03
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578 355 24 14 1.14E-03
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579 356 10 19 1.11E-03
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580 357 1 22 1.09E-03
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581 357 3 10 1.14E-03
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582 357 4 10 1.14E-03
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583 357 19 11 1.15E-03
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584 357 23 15 1.12E-03
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585 358 10 6 1.15E-03
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586 359 6 21 1.13E-03
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587 359 11 18 5.99E-04
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588 359 20 17 1.14E-03
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589 360 21 6 1.15E-03
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590 361 4 6 1.15E-03
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591 361 18 11 1.15E-03
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592 362 18 9 1.13E-03
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593 363 11 1 1.14E-03
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594 363 14 4 1.15E-03
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595 363 21 14 1.14E-03
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596 364 1 9 1.13E-03
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597 364 7 9 1.13E-03
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598 364 15 5 1.13E-03
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599 365 2 10 1.14E-03
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600 365 7 10 1.14E-03
For Julian Day 365, selecting COMIDA2 results # 1 of 9

TRIAL DAY PERIOD BIN PRBMET
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602 1 13 6 1.15E-03
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603 1 15 5 1.13E-03
For Julian Day 1, selecting COMIDA2 results # 1 of 9
604 1 21 14 1.14E-03
For Julian Day 1, selecting COMIDA2 results # 1 of 9
605 1 23 13 1.14E-03
For Julian Day 1, selecting COMIDA2 results # 1 of 9
606 2 4 21 1.13E-03
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607 2 6 20 1.12E-03
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608 2 11 17 1.14E-03
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609 2 14 17 1.14E-03
For Julian Day 2, selecting COMIDA2 results # 1 of 9
610 2 15 27 3.71E-04
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611 3 14 6 1.15E-03
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612 3 22 5 1.13E-03
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613 4 6 4 1.15E-03
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614 5 8 11 1.15E-03
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615 6 8 6 1.15E-03

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616 6 14 6 1.15E-03
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617 7 15 6 1.15E-03
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618 7 20 5 1.13E-03
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619 7 22 4 1.15E-03
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620 8 1 10 1.14E-03
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621 8 4 10 1.14E-03
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622 8 21 12 1.15E-03
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623 9 9 11 1.15E-03
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624 9 10 6 1.15E-03
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625 9 22 10 1.14E-03
For Julian Day 9, selecting COMIDA2 results # 1 of 9
626 11 14 23 1.14E-04
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627 11 15 22 1.09E-03
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628 12 5 11 1.15E-03
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629 12 13 2 1.14E-03
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630 13 9 10 1.14E-03
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631 13 22 21 1.13E-03
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632 14 15 8 3.04E-04
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633 14 17 8 3.04E-04
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634 14 19 8 3.04E-04
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635 14 23 8 3.04E-04
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636 15 2 8 3.04E-04
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637 15 5 8 3.04E-04
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638 15 7 8 3.04E-04
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639 15 9 8 3.04E-04
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640 15 13 8 3.04E-04
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641 15 15 8 3.04E-04
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642 15 21 12 1.15E-03
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643 16 9 5 1.13E-03
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644 16 12 1 1.14E-03
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645 16 14 4 1.15E-03
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646 16 16 3 8.56E-04
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647 17 18 20 1.12E-03
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648 18 3 17 1.14E-03
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649 18 6 22 1.09E-03
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650 18 15 8 3.04E-04
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TRIAL DAY PERIOD BIN PRBMET
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652 19 8 9 1.13E-03
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653 19 11 10 1.14E-03
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654 19 18 11 1.15E-03
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655 19 19 12 1.15E-03
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656 20 1 10 1.14E-03
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657 20 21 13 1.14E-03
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658 21 1 14 1.14E-03
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659 21 12 6 1.15E-03
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660 21 16 7 1.13E-03
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661 21 20 6 1.15E-03
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662 22 6 11 1.15E-03
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663 22 8 4 1.15E-03
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664 23 2 17 1.14E-03
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665 23 4 19 1.11E-03
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666 23 22 10 1.14E-03
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667 24 12 1 1.14E-03
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668 24 24 11 1.15E-03

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669 25 2 16 1.14E-04
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670 25 18 7 1.13E-03
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671 25 24 7 1.13E-03
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672 26 13 7 1.13E-03
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673 27 2 6 1.15E-03
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674 27 11 5 1.13E-03
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675 27 18 14 1.14E-03
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676 28 10 10 1.14E-03
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677 28 21 13 1.14E-03
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678 28 23 13 1.14E-03
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679 29 9 20 1.12E-03
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680 29 10 19 1.11E-03
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681 29 16 6 1.15E-03
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682 29 18 9 1.13E-03
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683 30 5 18 5.99E-04
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684 30 12 4 1.15E-03
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685 31 4 10 1.14E-03
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686 31 7 21 1.13E-03
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687 31 17 7 1.13E-03
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688 31 23 6 1.15E-03
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689 32 11 1 1.14E-03
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690 33 2 10 1.14E-03
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691 33 12 5 1.13E-03
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692 33 22 17 1.14E-03
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693 34 6 17 1.14E-03
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694 34 10 6 1.15E-03
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695 34 16 6 1.15E-03
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696 34 21 9 1.13E-03
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697 35 1 13 1.14E-03
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698 35 17 18 5.99E-04
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699 35 21 22 1.09E-03
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700 35 22 17 1.14E-03
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TRIAL DAY PERIOD BIN PRBMET
701 36 3 11 1.15E-03
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702 36 19 12 1.15E-03
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703 37 3 12 1.15E-03
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704 37 10 7 1.13E-03
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705 37 19 11 1.15E-03
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706 38 17 6 1.15E-03
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707 39 5 11 1.15E-03
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708 40 6 11 1.15E-03
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709 40 11 2 1.14E-03
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710 40 15 6 1.15E-03
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711 41 4 11 1.15E-03
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712 41 15 5 1.13E-03
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713 42 3 10 1.14E-03
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714 42 9 21 1.13E-03
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715 43 6 17 1.14E-03
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716 43 19 12 1.15E-03
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717 44 6 16 1.14E-04
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718 44 11 2 1.14E-03
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719 44 19 14 1.14E-03
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720 45 7 14 1.14E-03
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721 45 9 4 1.15E-03

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722 45 18 10 1.14E-03
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723 46 8 13 1.14E-03
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724 46 17 9 1.13E-03
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725 47 8 13 1.14E-03
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726 47 19 15 1.12E-03
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727 48 16 6 1.15E-03
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728 48 19 6 1.15E-03
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729 49 3 10 1.14E-03
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730 49 20 7 1.13E-03
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731 50 6 12 1.15E-03
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732 50 7 5 1.13E-03
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733 50 17 5 1.13E-03
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734 50 21 11 1.15E-03
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735 51 5 10 1.14E-03
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736 51 11 1 1.14E-03
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737 52 7 14 1.14E-03
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738 52 16 6 1.15E-03
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739 53 3 13 1.14E-03
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740 53 17 4 1.15E-03
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741 54 5 21 1.13E-03
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742 54 17 12 1.15E-03
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743 54 18 11 1.15E-03
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744 54 20 15 1.12E-03
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745 55 7 7 1.13E-03
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746 55 10 8 3.04E-04
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747 55 12 2 1.14E-03
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748 56 5 11 1.15E-03
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749 56 6 10 1.14E-03
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750 56 10 2 1.14E-03
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TRIAL DAY PERIOD BIN PRBMET
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752 57 8 7 1.13E-03
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753 58 14 6 1.15E-03
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754 58 16 6 1.15E-03
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755 59 7 5 1.13E-03
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756 59 11 2 1.14E-03
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757 60 1 10 1.14E-03
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758 60 4 6 1.15E-03
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759 61 5 20 1.12E-03
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760 61 7 20 1.12E-03
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761 61 21 7 1.13E-03
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762 62 23 7 1.13E-03
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763 63 3 6 1.15E-03
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764 63 10 2 1.14E-03
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765 63 20 12 1.15E-03
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766 64 7 6 1.15E-03
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767 64 23 12 1.15E-03
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768 64 24 11 1.15E-03
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769 65 8 4 1.15E-03
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770 65 12 1 1.14E-03
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771 66 10 6 1.15E-03
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772 66 11 2 1.14E-03
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773 66 14 2 1.14E-03
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774 66 19 10 1.14E-03

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775 67 1 10 1.14E-03
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776 67 5 14 1.14E-03
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777 67 16 5 1.13E-03
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778 68 6 5 1.13E-03
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779 68 17 6 1.15E-03
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780 69 6 12 1.15E-03
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781 69 14 7 1.13E-03
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782 70 1 11 1.15E-03
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783 70 11 1 1.14E-03
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784 70 22 11 1.15E-03
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785 71 10 17 1.14E-03
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786 71 11 3 8.56E-04
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787 71 13 4 1.15E-03
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788 72 4 18 5.99E-04
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789 72 9 9 1.13E-03
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790 72 14 2 1.14E-03
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791 73 3 12 1.15E-03
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792 74 5 12 1.15E-03
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793 74 10 7 1.13E-03
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794 75 15 2 1.14E-03
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795 75 21 11 1.15E-03
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796 76 6 5 1.13E-03
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797 76 23 12 1.15E-03
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798 77 8 6 1.15E-03
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799 77 9 2 1.14E-03
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800 77 14 2 1.14E-03
For Julian Day 77, selecting COMIDA2 results # 2 of 9

TRIAL DAY PERIOD BIN PRBMET
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802 78 19 6 1.15E-03
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803 78 21 7 1.13E-03
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804 79 13 2 1.14E-03
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805 80 8 1 1.14E-03
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806 80 19 4 1.15E-03
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807 81 5 6 1.15E-03
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808 81 15 6 1.15E-03
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809 82 3 6 1.15E-03
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810 83 3 5 1.13E-03
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811 83 10 2 1.14E-03
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812 83 20 11 1.15E-03
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813 84 1 5 1.13E-03
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814 84 12 1 1.14E-03
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815 84 17 5 1.13E-03
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816 84 19 10 1.14E-03
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817 85 2 11 1.15E-03
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818 86 3 12 1.15E-03
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819 86 7 6 1.15E-03
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820 86 12 2 1.14E-03
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821 86 16 6 1.15E-03
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822 86 18 11 1.15E-03
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823 86 20 14 1.14E-03
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824 87 1 13 1.14E-03
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825 87 15 1 1.14E-03
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826 87 21 4 1.15E-03
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827 87 24 21 1.13E-03

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828 88 6 19 1.11E-03
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829 89 5 13 1.14E-03
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830 89 11 10 1.14E-03
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831 90 19 12 1.15E-03
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832 91 8 5 1.13E-03
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833 91 16 7 1.13E-03
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834 92 13 1 1.14E-03
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835 92 22 13 1.14E-03
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836 93 3 10 1.14E-03
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837 93 7 5 1.13E-03
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838 93 11 21 1.13E-03
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839 93 16 36 1.43E-04
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840 93 17 35 1.14E-04
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841 93 18 34 1.14E-04
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842 93 19 33 1.14E-04
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843 93 21 27 3.71E-04
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844 94 1 12 1.15E-03
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845 94 10 2 1.14E-03
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846 94 22 11 1.15E-03
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847 95 9 7 1.13E-03
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848 95 13 2 1.14E-03
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849 95 16 6 1.15E-03
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850 96 6 10 1.14E-03
For Julian Day 96, selecting COMIDA2 results # 3 of 9

TRIAL DAY PERIOD BIN PRBMET
851 97 8 17 1.14E-03
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852 97 10 17 1.14E-03
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853 97 15 2 1.14E-03
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854 98 16 6 1.15E-03
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855 98 20 11 1.15E-03
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856 98 24 12 1.15E-03
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857 99 24 14 1.14E-03
For Julian Day 99, selecting COMIDA2 results # 3 of 9
858 100 15 1 1.14E-03
For Julian Day 100, selecting COMIDA2 results # 3 of 9
859 100 22 9 1.13E-03
For Julian Day 100, selecting COMIDA2 results # 3 of 9
860 101 6 13 1.14E-03
For Julian Day 101, selecting COMIDA2 results # 3 of 9
861 101 21 11 1.15E-03
For Julian Day 101, selecting COMIDA2 results # 3 of 9
862 102 12 2 1.14E-03
For Julian Day 102, selecting COMIDA2 results # 3 of 9
863 102 13 6 1.15E-03
For Julian Day 102, selecting COMIDA2 results # 3 of 9
864 103 8 9 1.13E-03
For Julian Day 103, selecting COMIDA2 results # 3 of 9
865 104 3 20 1.12E-03
For Julian Day 104, selecting COMIDA2 results # 3 of 9
866 104 4 20 1.12E-03
For Julian Day 104, selecting COMIDA2 results # 3 of 9
867 104 12 10 1.14E-03
For Julian Day 104, selecting COMIDA2 results # 3 of 9
868 104 21 19 1.11E-03
For Julian Day 104, selecting COMIDA2 results # 3 of 9
869 105 4 14 1.14E-03
For Julian Day 105, selecting COMIDA2 results # 3 of 9
870 105 9 4 1.15E-03
For Julian Day 105, selecting COMIDA2 results # 3 of 9
871 105 14 2 1.14E-03
For Julian Day 105, selecting COMIDA2 results # 3 of 9
872 105 22 12 1.15E-03
For Julian Day 105, selecting COMIDA2 results # 3 of 9
873 106 2 11 1.15E-03
For Julian Day 106, selecting COMIDA2 results # 3 of 9
874 106 14 2 1.14E-03
For Julian Day 106, selecting COMIDA2 results # 3 of 9
875 106 16 1 1.14E-03
For Julian Day 106, selecting COMIDA2 results # 3 of 9
876 106 24 10 1.14E-03
For Julian Day 106, selecting COMIDA2 results # 3 of 9
877 107 1 5 1.13E-03
For Julian Day 107, selecting COMIDA2 results # 3 of 9
878 107 3 10 1.14E-03
For Julian Day 107, selecting COMIDA2 results # 3 of 9
879 107 13 1 1.14E-03
For Julian Day 107, selecting COMIDA2 results # 3 of 9
880 108 18 5 1.13E-03

For Julian Day 108, selecting COMIDA2 results # 3 of 9
881 109 13 2 1.14E-03
For Julian Day 109, selecting COMIDA2 results # 3 of 9
882 109 17 6 1.15E-03
For Julian Day 109, selecting COMIDA2 results # 3 of 9
883 110 2 11 1.15E-03
For Julian Day 110, selecting COMIDA2 results # 3 of 9
884 110 14 1 1.14E-03
For Julian Day 110, selecting COMIDA2 results # 3 of 9
885 110 22 13 1.14E-03
For Julian Day 110, selecting COMIDA2 results # 3 of 9
886 111 23 17 1.14E-03
For Julian Day 111, selecting COMIDA2 results # 3 of 9
887 112 6 17 1.14E-03
For Julian Day 112, selecting COMIDA2 results # 3 of 9
888 112 10 23 1.14E-04
For Julian Day 112, selecting COMIDA2 results # 3 of 9
889 112 19 32 3.23E-04
For Julian Day 112, selecting COMIDA2 results # 3 of 9
890 112 21 27 3.71E-04
For Julian Day 112, selecting COMIDA2 results # 3 of 9
891 113 1 17 1.14E-03
For Julian Day 113, selecting COMIDA2 results # 3 of 9
892 113 20 21 1.13E-03
For Julian Day 113, selecting COMIDA2 results # 3 of 9
893 114 14 2 1.14E-03
For Julian Day 114, selecting COMIDA2 results # 3 of 9
894 115 9 4 1.15E-03
For Julian Day 115, selecting COMIDA2 results # 3 of 9
895 115 11 1 1.14E-03
For Julian Day 115, selecting COMIDA2 results # 3 of 9
896 115 19 12 1.15E-03
For Julian Day 115, selecting COMIDA2 results # 3 of 9
897 116 20 10 1.14E-03
For Julian Day 116, selecting COMIDA2 results # 3 of 9
898 116 24 9 1.13E-03
For Julian Day 116, selecting COMIDA2 results # 3 of 9
899 117 1 14 1.14E-03
For Julian Day 117, selecting COMIDA2 results # 3 of 9
900 117 12 1 1.14E-03
For Julian Day 117, selecting COMIDA2 results # 3 of 9

TRIAL DAY PERIOD BIN PRBMET
901 117 13 1 1.14E-03
For Julian Day 117, selecting COMIDA2 results # 3 of 9
902 117 20 14 1.14E-03
For Julian Day 117, selecting COMIDA2 results # 3 of 9
903 117 22 13 1.14E-03
For Julian Day 117, selecting COMIDA2 results # 3 of 9
904 118 15 2 1.14E-03
For Julian Day 118, selecting COMIDA2 results # 3 of 9
905 119 16 1 1.14E-03
For Julian Day 119, selecting COMIDA2 results # 3 of 9
906 119 22 10 1.14E-03
For Julian Day 119, selecting COMIDA2 results # 3 of 9
907 119 23 11 1.15E-03
For Julian Day 119, selecting COMIDA2 results # 3 of 9
908 120 4 9 1.13E-03
For Julian Day 120, selecting COMIDA2 results # 3 of 9
909 121 10 1 1.14E-03
For Julian Day 121, selecting COMIDA2 results # 3 of 9
910 121 23 10 1.14E-03
For Julian Day 121, selecting COMIDA2 results # 3 of 9
911 122 9 4 1.15E-03
For Julian Day 122, selecting COMIDA2 results # 3 of 9
912 122 15 1 1.14E-03
For Julian Day 122, selecting COMIDA2 results # 3 of 9
913 122 24 10 1.14E-03
For Julian Day 122, selecting COMIDA2 results # 3 of 9
914 123 10 1 1.14E-03
For Julian Day 123, selecting COMIDA2 results # 3 of 9
915 124 3 14 1.14E-03
For Julian Day 124, selecting COMIDA2 results # 3 of 9
916 124 6 14 1.14E-03
For Julian Day 124, selecting COMIDA2 results # 3 of 9
917 124 14 2 1.14E-03
For Julian Day 124, selecting COMIDA2 results # 3 of 9
918 125 1 15 1.12E-03
For Julian Day 125, selecting COMIDA2 results # 3 of 9
919 125 8 5 1.13E-03
For Julian Day 125, selecting COMIDA2 results # 3 of 9
920 125 12 1 1.14E-03
For Julian Day 125, selecting COMIDA2 results # 3 of 9
921 125 22 13 1.14E-03
For Julian Day 125, selecting COMIDA2 results # 3 of 9
922 126 10 4 1.15E-03
For Julian Day 126, selecting COMIDA2 results # 3 of 9
923 126 16 6 1.15E-03
For Julian Day 126, selecting COMIDA2 results # 3 of 9
924 127 14 1 1.14E-03
For Julian Day 127, selecting COMIDA2 results # 3 of 9
925 128 2 18 5.99E-04
For Julian Day 128, selecting COMIDA2 results # 3 of 9
926 128 6 10 1.14E-03
For Julian Day 128, selecting COMIDA2 results # 3 of 9
927 128 12 4 1.15E-03
For Julian Day 128, selecting COMIDA2 results # 3 of 9
928 129 1 9 1.13E-03
For Julian Day 129, selecting COMIDA2 results # 3 of 9
929 129 16 1 1.14E-03
For Julian Day 129, selecting COMIDA2 results # 3 of 9
930 130 1 10 1.14E-03
For Julian Day 130, selecting COMIDA2 results # 3 of 9
931 130 9 10 1.14E-03
For Julian Day 130, selecting COMIDA2 results # 3 of 9
932 130 19 11 1.15E-03
For Julian Day 130, selecting COMIDA2 results # 3 of 9
933 131 2 9 1.13E-03

For Julian Day 131, selecting COMIDA2 results # 3 of 9
934 131 19 19 1.11E-03
For Julian Day 131, selecting COMIDA2 results # 3 of 9
935 132 2 17 1.14E-03
For Julian Day 132, selecting COMIDA2 results # 3 of 9
936 132 10 1 1.14E-03
For Julian Day 132, selecting COMIDA2 results # 3 of 9
937 132 20 10 1.14E-03
For Julian Day 132, selecting COMIDA2 results # 3 of 9
938 133 24 21 1.13E-03
For Julian Day 133, selecting COMIDA2 results # 3 of 9
939 134 1 21 1.13E-03
For Julian Day 134, selecting COMIDA2 results # 3 of 9
940 134 4 20 1.12E-03
For Julian Day 134, selecting COMIDA2 results # 3 of 9
941 134 5 20 1.12E-03
For Julian Day 134, selecting COMIDA2 results # 3 of 9
942 134 7 19 1.11E-03
For Julian Day 134, selecting COMIDA2 results # 3 of 9
943 135 1 18 5.99E-04
For Julian Day 135, selecting COMIDA2 results # 3 of 9
944 135 5 17 1.14E-03
For Julian Day 135, selecting COMIDA2 results # 3 of 9
945 135 24 20 1.12E-03
For Julian Day 135, selecting COMIDA2 results # 3 of 9
946 136 18 6 1.15E-03
For Julian Day 136, selecting COMIDA2 results # 3 of 9
947 136 22 14 1.14E-03
For Julian Day 136, selecting COMIDA2 results # 3 of 9
948 137 3 14 1.14E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9
949 137 7 4 1.15E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9
950 137 8 5 1.13E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9

TRIAL DAY PERIOD BIN PRBMET
951 137 14 17 1.14E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9
952 137 19 10 1.14E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9
953 137 21 11 1.15E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9
954 138 7 3 8.56E-04
For Julian Day 138, selecting COMIDA2 results # 4 of 9
955 138 15 5 1.13E-03
For Julian Day 138, selecting COMIDA2 results # 4 of 9
956 139 19 12 1.15E-03
For Julian Day 139, selecting COMIDA2 results # 4 of 9
957 140 11 2 1.14E-03
For Julian Day 140, selecting COMIDA2 results # 4 of 9
958 141 2 14 1.14E-03
For Julian Day 141, selecting COMIDA2 results # 4 of 9
959 141 4 14 1.14E-03
For Julian Day 141, selecting COMIDA2 results # 4 of 9
960 141 6 13 1.14E-03
For Julian Day 141, selecting COMIDA2 results # 4 of 9
961 141 22 12 1.15E-03
For Julian Day 141, selecting COMIDA2 results # 4 of 9
962 142 7 6 1.15E-03
For Julian Day 142, selecting COMIDA2 results # 4 of 9
963 142 12 2 1.14E-03
For Julian Day 142, selecting COMIDA2 results # 4 of 9
964 142 20 11 1.15E-03
For Julian Day 142, selecting COMIDA2 results # 4 of 9
965 143 3 15 1.12E-03
For Julian Day 143, selecting COMIDA2 results # 4 of 9
966 144 12 2 1.14E-03
For Julian Day 144, selecting COMIDA2 results # 4 of 9
967 144 13 1 1.14E-03
For Julian Day 144, selecting COMIDA2 results # 4 of 9
968 145 23 21 1.13E-03
For Julian Day 145, selecting COMIDA2 results # 4 of 9
969 146 18 5 1.13E-03
For Julian Day 146, selecting COMIDA2 results # 4 of 9
970 146 20 10 1.14E-03
For Julian Day 146, selecting COMIDA2 results # 4 of 9
971 147 6 14 1.14E-03
For Julian Day 147, selecting COMIDA2 results # 4 of 9
972 147 14 6 1.15E-03
For Julian Day 147, selecting COMIDA2 results # 4 of 9
973 147 18 11 1.15E-03
For Julian Day 147, selecting COMIDA2 results # 4 of 9
974 148 2 10 1.14E-03
For Julian Day 148, selecting COMIDA2 results # 4 of 9
975 148 4 14 1.14E-03
For Julian Day 148, selecting COMIDA2 results # 4 of 9
976 148 10 4 1.15E-03
For Julian Day 148, selecting COMIDA2 results # 4 of 9
977 148 15 1 1.14E-03
For Julian Day 148, selecting COMIDA2 results # 4 of 9
978 149 1 13 1.14E-03
For Julian Day 149, selecting COMIDA2 results # 4 of 9
979 149 7 9 1.13E-03
For Julian Day 149, selecting COMIDA2 results # 4 of 9
980 149 16 4 1.15E-03
For Julian Day 149, selecting COMIDA2 results # 4 of 9
981 150 5 13 1.14E-03
For Julian Day 150, selecting COMIDA2 results # 4 of 9
982 150 9 5 1.13E-03
For Julian Day 150, selecting COMIDA2 results # 4 of 9
983 150 23 10 1.14E-03
For Julian Day 150, selecting COMIDA2 results # 4 of 9
984 151 9 1 1.14E-03
For Julian Day 151, selecting COMIDA2 results # 4 of 9

"ATMOS" DESCRIPTION = OCP3 low density no spray
PROB QUANTILES PEAK PEAK PEAK

	NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONS	PROB	TRIAL
Source Term 1: Plume 1, at 0-0.2 km										
Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	2.51E+10	1.89E+10	4.59E+10	6.05E+10	****	****	7.92E+10	2.97E-02	127
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	9.50E+09	7.51E+09	1.60E+10	2.06E+10	2.84E+10	3.63E+10	4.31E+10	3.40E-03	315
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	7.29E+07	5.80E+07	1.22E+08	1.45E+08	2.27E+08	2.96E+08	3.40E+08	3.23E-04	91
Total Center Ground Conc. (Bq/m2)	1.0000	3.50E+08	2.82E+08	6.10E+08	7.43E+08	1.03E+09	1.12E+09	1.59E+09	3.23E-04	91
Ground-Level Dilution, X/Q (s/m3)	1.0000	4.21E-05	3.35E-05	7.33E-05	8.21E-05	1.20E-04	1.62E-04	1.93E-04	3.40E-03	315
Cs-137 Adjusted Source, Q (Bq)	1.0000	2.26E+14	2.03E+14	2.09E+14	2.11E+14	2.17E+14	2.20E+14	2.26E+14	1.15E-03	443
Plume Sigma-y (m)	1.0000	3.52E+01	3.35E+01	****	****	****	****	4.46E+01	1.60E-01	22
Plume Sigma-z (m)	1.0000	3.06E+01	2.77E+01	5.11E+01	****	****	****	5.14E+01	9.21E-02	2
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.50E+05	1.11E+05	1.43E+05	****	****	****	1.50E+05	7.19E-02	57
Source Term 1: Plume 1, at 0.2-0.5 km										
Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	6.53E+09	5.47E+09	1.24E+10	1.69E+10	****	****	2.15E+10	2.97E-02	127
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	4.25E+09	3.38E+09	7.87E+09	9.76E+09	1.24E+10	1.36E+10	1.66E+10	1.15E-03	457
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	3.19E+07	2.63E+07	5.75E+07	7.07E+07	8.74E+07	9.58E+07	1.28E+08	3.23E-04	91
Total Center Ground Conc. (Bq/m2)	1.0000	1.53E+08	1.25E+08	2.91E+08	3.30E+08	4.22E+08	4.69E+08	5.99E+08	3.23E-04	91
Ground-Level Dilution, X/Q (s/m3)	1.0000	1.91E-05	1.43E-05	3.52E-05	4.65E-05	5.72E-05	6.10E-05	7.66E-05	1.15E-03	457
Cs-137 Adjusted Source, Q (Bq)	1.0000	2.24E+14	2.02E+14	2.07E+14	2.09E+14	2.14E+14	2.16E+14	2.26E+14	3.04E-04	639
Plume Sigma-y (m)	1.0000	1.03E+02	1.07E+02	****	****	****	****	1.39E+02	1.60E-01	22
Plume Sigma-z (m)	1.0000	5.56E+01	4.51E+01	****	****	****	****	1.60E+02	1.22E-01	2
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.50E+05	1.11E+05	1.43E+05	****	****	****	1.50E+05	7.19E-02	57
Source Term 1: Plume 1, at 0.5-1.2 km										
Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	2.50E+09	2.23E+09	4.90E+09	5.90E+09	****	****	8.62E+09	1.06E-02	317
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	2.33E+09	2.04E+09	4.42E+09	5.43E+09	7.15E+09	7.38E+09	7.92E+09	1.13E-03	441
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	1.69E+07	1.36E+07	3.17E+07	3.81E+07	5.31E+07	5.71E+07	8.86E+07	3.71E-04	280
Total Center Ground Conc. (Bq/m2)	1.0000	8.12E+07	7.11E+07	1.39E+08	1.73E+08	2.36E+08	2.60E+08	4.18E+08	3.71E-04	280
Ground-Level Dilution, X/Q (s/m3)	1.0000	1.08E-05	8.80E-06	2.09E-05	2.49E-05	3.24E-05	3.44E-05	3.83E-05	1.50E-03	280
Cs-137 Adjusted Source, Q (Bq)	1.0000	2.19E+14	2.03E+14	2.09E+14	2.12E+14	2.19E+14	2.22E+14	2.25E+14	2.29E-03	800
Plume Sigma-y (m)	1.0000	2.01E+02	1.52E+02	****	****	****	****	2.66E+02	1.40E-01	2
Plume Sigma-z (m)	1.0000	1.11E+02	4.55E+01	3.24E+02	3.79E+02	****	****	4.35E+02	2.72E-02	33
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.50E+05	1.10E+05	1.36E+05	1.49E+05	****	****	1.51E+05	4.40E-02	57
Source Term 1: Plume 1, at 1.2-1.6 km										
Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	1.56E+09	1.23E+09	3.00E+09	3.70E+09	5.14E+09	5.30E+09	5.65E+09	1.14E-03	262
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	1.66E+09	1.26E+09	3.15E+09	3.84E+09	5.22E+09	5.46E+09	6.00E+09	1.14E-03	262
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	1.16E+07	1.07E+07	2.12E+07	2.57E+07	3.47E+07	3.81E+07	5.24E+07	3.23E-04	91
Total Center Ground Conc. (Bq/m2)	1.0000	5.57E+07	5.30E+07	1.03E+08	1.15E+08	1.51E+08	1.69E+08	2.45E+08	3.23E-04	91
Ground-Level Dilution, X/Q (s/m3)	1.0000	7.92E-06	7.01E-06	1.62E-05	2.17E-05	****	****	3.01E-05	1.86E-02	111
Cs-137 Adjusted Source, Q (Bq)	1.0000	2.15E+14	2.02E+14	2.08E+14	2.10E+14	2.16E+14	2.19E+14	2.25E+14	1.15E-03	849
Plume Sigma-y (m)	1.0000	2.74E+02	2.48E+02	****	****	****	****	4.06E+02	1.38E-01	2
Plume Sigma-z (m)	1.0000	1.72E+02	4.60E+01	7.03E+02	7.08E+02	7.21E+02	7.27E+02	7.55E+02	1.52E-04	415
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.51E+05	1.10E+05	1.37E+05	1.50E+05	****	****	1.53E+05	4.40E-02	57
Source Term 1: Plume 1, at 1.6-2.1 km										
Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	1.25E+09	1.09E+09	2.59E+09	3.14E+09	3.76E+09	4.07E+09	4.81E+09	1.13E-03	201
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	1.39E+09	1.12E+09	2.90E+09	3.30E+09	4.23E+09	4.71E+09	5.38E+09	1.13E-03	201
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	9.44E+06	8.22E+06	1.95E+07	2.23E+07	2.93E+07	3.18E+07	4.28E+07	3.23E-04	91
Total Center Ground Conc. (Bq/m2)	1.0000	4.53E+07	3.69E+07	9.48E+07	1.08E+08	1.31E+08	1.43E+08	2.00E+08	3.23E-04	91
Ground-Level Dilution, X/Q (s/m3)	1.0000	6.82E-06	5.66E-06	1.46E-05	2.02E-05	2.39E-05	2.57E-05	2.68E-05	3.41E-03	111
Cs-137 Adjusted Source, Q (Bq)	1.0000	2.11E+14	2.02E+14	2.08E+14	2.10E+14	2.16E+14	2.19E+14	2.25E+14	1.15E-03	849
Plume Sigma-y (m)	1.0000	3.20E+02	2.74E+02	****	****	****	****	5.17E+02	1.34E-01	2
Plume Sigma-z (m)	1.0000	2.28E+02	6.63E+01	1.00E+03	1.01E+03	1.04E+03	1.05E+03	1.06E+03	2.54E-03	50
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.51E+05	1.10E+05	1.37E+05	1.51E+05	****	****	1.54E+05	4.40E-02	57
Source Term 1: Plume 1, at 2.1-3.2 km										
Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	9.20E+08	7.84E+08	2.01E+09	2.33E+09	3.10E+09	3.27E+09	3.65E+09	1.13E-03	597
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	1.03E+09	9.34E+08	2.26E+09	3.00E+09	3.45E+09	3.66E+09	4.16E+09	1.13E-03	597
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	6.73E+06	6.50E+06	1.24E+07	1.48E+07	2.09E+07	2.25E+07	3.05E+07	3.23E-04	91
Total Center Ground Conc. (Bq/m2)	1.0000	3.23E+07	3.14E+07	6.85E+07	7.73E+07	9.92E+07	1.07E+08	1.42E+08	3.23E-04	91
Ground-Level Dilution, X/Q (s/m3)	1.0000	5.31E-06	4.06E-06	1.11E-05	1.37E-05	****	****	2.16E-05	1.47E-02	201
Cs-137 Adjusted Source, Q (Bq)	1.0000	2.06E+14	2.01E+14	2.07E+14	2.10E+14	2.16E+14	2.19E+14	2.24E+14	1.15E-03	849
Plume Sigma-y (m)	1.0000	3.97E+02	3.54E+02	****	****	****	****	7.01E+02	1.25E-01	2
Plume Sigma-z (m)	1.0000	3.31E+02	6.76E+01	1.04E+03	1.13E+03	1.38E+03	1.51E+03	1.65E+03	2.39E-03	50
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.52E+05	1.09E+05	1.34E+05	1.46E+05	****	****	1.55E+05	3.04E-02	110
Source Term 1: Plume 1, at 3.2-4.0 km										
Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	6.78E+08	5.79E+08	1.46E+09	1.95E+09	2.18E+09	2.26E+09	2.61E+09	3.71E-04	279
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	7.56E+08	6.61E+08	1.56E+09	2.01E+09	2.28E+09	2.41E+09	2.95E+09	3.71E-04	279
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	4.72E+06	4.16E+06	9.74E+06	1.09E+07	1.37E+07	1.51E+07	2.98E+07	3.71E-04	279
Total Center Ground Conc. (Bq/m2)	1.0000	2.26E+07	2.14E+07	4.66E+07	5.34E+07	6.60E+07	7.22E+07	1.41E+08	3.71E-04	279
Ground-Level Dilution, X/Q (s/m3)	1.0000	4.06E-06	3.29E-06	9.26E-06	1.07E-05	1.30E-05	1.42E-05	1.70E-05	1.13E-03	517
Cs-137 Adjusted Source, Q (Bq)	1.0000	2.01E+14	2.01E+14	2.07E+14	2.09E+14	2.16E+14	2.18E+14	2.24E+14	1.15E-03	849
Plume Sigma-y (m)	1.0000	4.85E+02	3.64E+02	7.40E+02	8.23E+02	****	****	9.11E+02	2.58E-02	62
Plume Sigma-z (m)	1.0000	4.57E+02	9.64E+01	2.04E+03	2.15E+03	****	****	2.39E+03	1.30E-02	50
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.52E+05	1.09E+05	1.35E+05	1.47E+05	****	****	1.57E+05	3.04E-02	110
Source Term 1: Plume 1, at 4.0-4.8 km										
Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	5.42E+08	4.65E+08	1.07E+09	1.21E+09	1.61E+09	1.81E+09	2.10E+09	1.14E-03	305
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	5.97E+08	5.29E+08	1.21E+09	1.47E+09	2.06E+09	2.15E+09	2.35E+09	1.14E-03	305
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	3.62E+06	3.31E+06	7.09E+06	8.24E+06	1.13E+07	1.28E+07	2.24E+07	3.71E-04	279
Total Center Ground Conc. (Bq/m2)	1.0000	1.73E+07	1.50E+07	3.31E+07	3.80E+07	5.15E+07	5.59E+07	1.05E+08	3.71E-04	279
Ground-Level Dilution, X/Q (s/m3)	1.0000	3.31E-06	2.55E-06	7.44E-06	1.01E-05	1.21E-05	1.30E-05	1.42E-05	2.26E-03	374
Cs-137 Adjusted Source, Q (Bq)	1.0000	1.98E+14	2.00E+14	2.06E+14	2.09E+14	2.15E+14	2.18E+14	2.24E+14	1.15E-03	849
Plume Sigma-y (m)	1.0000	5.58E+02	4.90E+02	1.06E+03	****	****	****	1.08E+03	9.12E-02	2
Plume Sigma-z (m)	1.0000	5.66E+02	9.80E+01	3.00E+03	3.01E+03	3.04E+03	3.06E+03	3.06E+03	3.42E-03	342
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.53E+05	1.09E+05	1.31E+05	1.43E+05	****	****	1.59E+05	2.02E-02	110
Source Term 1: Plume 1, at 4.8-5.6 km										
Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	4.44E+08	3.69E+08	9.45E+08	1.08E+09	1.34E+09	1.47E+09	1.80E+09	1.13E-03	696
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	4.84E+08	4.05E+08	1.01E+09	1.12E+09	1.43E+09	1.59E+09	1.99E+09	1.13E-03	696
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	2.86E+06	2.73E+06	5.67E+06	6.88E+06	8.76E+06	9.69E+06	1.40E+07	3.71E-04	279
Total Center Ground Conc. (Bq/m2)	1.0000	1.37E+07	1.14E+07	2.81E+07	3.18E+07	3.84E+07	4.17E+07	6.56E+07	3.71E-04	279
Ground-Level Dilution, X/Q (s/m3)	1.0000	2.76E-06	2.16E-06	6.24E-06	9.88E-06	1.15E-05	****	1.20E-05	6.27E-03	305
Cs-137 Adjusted Source, Q (Bq)	1.0000	1.94E+14	1.90E+14	2.06E+14	2.09E+14	2.15E+14	2.18E+14	2.24E+14	1.15E-03	849
Plume Sigma-y (m)	1.0000	6.29E+02	5.29E+02	****	****	****	****	1.25E+03	1.06E-01	2
Plume Sigma-z (m)	1.0000	6.79E+02	9.90E+01	3.03E+03	3.15E+03	3.46E+03	3.61E+03	3.78E+03	2.28E-03	344
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.53E+05	1.09E+05	1.32E+05	1.44E+05	****	****	1.60E+05	2.02E-02	110

Source Term 1: Plume 1, at 5.6-8.1 km

Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	3.14E+08	2.75E+08	6.72E+08	8.27E+08	1.08E+09	1.14E+09	1.28E+09	1.13E-03	592
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	3.37E+08	3.02E+08	7.36E+08	9.57E+08	1.14E+09	1.21E+09	1.39E+09	1.13E-03	592
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	1.92E+06	1.91E+06	3.65E+06	4.33E+06	5.48E+06	5.83E+06	8.24E+06	3.23E-04	90
Total Center Ground Conc. (Bq/m2)	1.0000	9.17E+06	9.13E+06	1.82E+07	2.14E+07	2.75E+07	3.04E+07	3.85E+07	3.23E-04	90
Ground-Level Dilution, X/Q (s/m3)	1.0000	2.01E-06	1.39E-06	4.64E-06	6.46E-06	7.94E-06	8.44E-06	9.04E-06	2.27E-03	305
Cs-137 Adjusted Source, Q (Bq)	1.0000	1.89E+14	1.69E+14	2.05E+14	2.08E+14	2.14E+14	2.17E+14	2.23E+14	1.15E-03	849
Plume Sigma-y (m)	1.0000	7.68E+02	6.23E+02	1.11E+03	1.30E+03	****	****	1.57E+03	2.09E-02	62
Plume Sigma-z (m)	1.0000	9.17E+02	1.90E+02	4.84E+03	5.07E+03	5.23E+03	5.30E+03	5.31E+03	4.56E-03	344
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.54E+05	1.08E+05	1.30E+05	1.40E+05	****	****	1.63E+05	1.28E-02	110

Source Term 1: Plume 1, at 8.1-11.3 km

Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	1.90E+08	1.48E+08	3.98E+08	5.00E+08	6.87E+08	7.41E+08	8.51E+08	1.14E-03	937
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	2.00E+08	1.62E+08	4.19E+08	5.20E+08	7.17E+08	7.72E+08	9.03E+08	1.14E-03	937
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	1.08E+06	1.02E+06	2.17E+06	2.63E+06	3.58E+06	3.98E+06	7.17E+06	1.14E-04	278
Total Center Ground Conc. (Bq/m2)	1.0000	5.19E+06	5.11E+06	1.02E+07	1.14E+07	1.47E+07	1.64E+07	3.35E+07	1.14E-04	278
Ground-Level Dilution, X/Q (s/m3)	1.0000	1.26E-06	1.00E-06	3.01E-06	3.81E-06	5.34E-06	5.66E-06	6.05E-06	2.28E-03	305
Cs-137 Adjusted Source, Q (Bq)	1.0000	1.82E+14	1.57E+14	2.05E+14	2.07E+14	2.14E+14	2.17E+14	2.23E+14	1.15E-03	849
Plume Sigma-y (m)	1.0000	1.00E+03	8.19E+02	1.97E+03	****	****	****	2.12E+03	8.48E-02	2
Plume Sigma-z (m)	1.0000	1.36E+03	2.05E+02	6.57E+03	7.24E+03	7.88E+03	8.17E+03	8.21E+03	4.55E-03	213
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.56E+05	1.08E+05	1.31E+05	1.42E+05	****	****	1.69E+05	1.14E-02	110

Source Term 1: Plume 1, at 11.3-16.1 km

Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	1.11E+08	8.60E+07	2.38E+08	2.94E+08	3.66E+08	4.01E+08	4.86E+08	1.14E-03	316
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	1.14E+08	8.70E+07	2.46E+08	3.03E+08	3.75E+08	4.11E+08	5.06E+08	1.14E-03	316
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	5.92E+05	5.20E+05	1.15E+06	1.36E+06	1.98E+06	2.18E+06	4.04E+06	1.14E-04	327
Total Center Ground Conc. (Bq/m2)	1.0000	2.83E+06	2.51E+06	5.70E+06	6.88E+06	8.80E+06	9.74E+06	1.88E+07	1.14E-04	327
Ground-Level Dilution, X/Q (s/m3)	1.0000	7.69E-07	5.21E-07	1.74E-06	2.35E-06	3.34E-06	3.60E-06	3.91E-06	2.28E-03	305
Cs-137 Adjusted Source, Q (Bq)	1.0000	1.75E+14	1.49E+14	2.04E+14	2.07E+14	2.13E+14	2.16E+14	2.22E+14	1.15E-03	849
Plume Sigma-y (m)	1.0000	1.34E+03	1.02E+03	2.47E+03	****	****	****	2.87E+03	7.41E-02	2
Plume Sigma-z (m)	1.0000	2.07E+03	3.03E+02	9.63E+03	1.07E+04	1.25E+04	****	1.28E+04	8.36E-03	50
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.59E+05	1.07E+05	1.26E+05	1.35E+05	1.59E+05	1.70E+05	1.77E+05	3.40E-03	314

Source Term 1: Plume 1, at 64.4-80.5 km

Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	5.65E+06	4.08E+06	1.11E+07	1.38E+07	2.18E+07	2.53E+07	3.08E+07	1.15E-03	410
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	5.66E+06	4.08E+06	1.11E+07	1.39E+07	2.21E+07	2.55E+07	3.09E+07	1.15E-03	410
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	3.11E+04	2.15E+04	6.29E+04	8.54E+04	1.45E+05	1.78E+05	3.62E+05	1.13E-03	120
Total Center Ground Conc. (Bq/m2)	1.0000	1.47E+05	1.04E+05	3.05E+05	4.01E+05	6.73E+05	8.65E+05	1.67E+06	1.13E-03	120
Ground-Level Dilution, X/Q (s/m3)	1.0000	4.55E-08	3.23E-08	9.84E-08	1.21E-07	1.95E-07	2.25E-07	3.10E-07	1.13E-03	516
Cs-137 Adjusted Source, Q (Bq)	1.0000	1.36E+14	1.11E+14	1.56E+14	1.81E+14	2.03E+14	2.05E+14	2.10E+14	1.14E-03	800
Plume Sigma-y (m)	1.0000	5.81E+03	5.21E+03	8.76E+03	1.02E+04	****	****	1.21E+04	1.20E-02	382
Plume Sigma-z (m)	1.0000	8.65E+03	1.90E+03	3.00E+04	3.25E+04	3.90E+04	4.22E+04	5.06E+04	1.13E-03	1
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	1.92E+05	1.53E+05	2.12E+05	2.20E+05	2.38E+05	2.47E+05	2.66E+05	1.13E-03	516

Source Term 1: Plume 1, at 113-161 km

Cs-137 Center Air Conc. (Bq-s/m3)	1.0000	2.09E+06	1.67E+06	3.66E+06	4.67E+06	6.87E+06	7.77E+06	1.00E+07	1.15E-03	721
Cs-137 Ground Air Conc. (Bq-s/m3)	1.0000	2.09E+06	1.67E+06	3.66E+06	4.67E+06	6.87E+06	7.77E+06	1.00E+07	1.15E-03	721
Cs-137 Center Ground Conc. (Bq/m2)	1.0000	1.16E+04	8.05E+03	2.26E+04	3.33E+04	5.95E+04	7.40E+04	1.11E+05	1.13E-03	74
Total Center Ground Conc. (Bq/m2)	1.0000	5.44E+04	3.75E+04	1.05E+05	1.51E+05	2.63E+05	3.16E+05	5.13E+05	1.13E-03	74
Ground-Level Dilution, X/Q (s/m3)	1.0000	1.96E-08	1.45E-08	3.58E-08	4.68E-08	7.33E-08	7.84E-08	9.08E-08	1.12E-03	549
Cs-137 Adjusted Source, Q (Bq)	1.0000	1.16E+14	1.03E+14	1.22E+14	1.31E+14	1.55E+14	1.67E+14	1.95E+14	1.14E-03	800
Plume Sigma-y (m)	1.0000	1.00E+04	9.17E+03	1.29E+04	1.46E+04	1.95E+04	****	2.10E+04	6.30E-03	382
Plume Sigma-z (m)	1.0000	1.03E+04	2.73E+03	3.17E+04	3.67E+04	5.02E+04	5.14E+04	5.40E+04	1.11E-03	139
Plume Height (m)	1.0000	5.00E+01	****	****	****	****	****	5.00E+01	1.00E+00	1
Plume Arrival Time (s)	1.0000	2.26E+05	2.09E+05	2.50E+05	2.69E+05	3.07E+05	3.16E+05	3.35E+05	1.14E-03	514

"ATMOS" DESCRIPTION = OCP3 low density no spray
"EARLY" DESCRIPTION = OCP3 low density no spray, EARLY input
"CHRONC" DESCRIPTION = OCP3 low density no spray

SOURCE TERM 1 OF 1:
OCP3 low density no spray

OVERALL RESULTS OBTAINED BY COMBINING 6 EMERGENCY RESPONSE COHORTS FROM "EARLY" WITH THE WEIGHTING FRACTIONS BELOW APPLIED TO THEM:

	FRACTION OF THE PEOPLE

COHORT 1 = Group 1	0.300
COHORT 2 = Group 2	0.417
COHORT 3 = Group 3	0.006
COHORT 4 = Group 4	0.100
COHORT 5 = Group 5	0.172
COHORT 6 = Group 6	0.005

AND THEN MERGING THE 6 RESULTS ABOVE WITH THE SINGLE SET OF RESULTS FROM "CHRONC" DESCRIBED BELOW:

COHORT 7 = OCP3 low density no spray

RESULTS WHICH ARE PRODUCED ONLY BY "EARLY" OR ONLY BY "CHRONC" ARE PRESENTED IN LATER SECTIONS.

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
HEALTH EFFECTS CASES											
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	1.0000	2.63E+01	2.41E+01	3.79E+01	4.31E+01	5.48E+01	5.92E+01	7.62E+01	1.12E-03	391
CAN INJ/TOTAL	0-32.2 km	1.0000	1.51E+02	1.24E+02	2.48E+02	2.90E+02	3.68E+02	4.06E+02	5.22E+02	1.12E-03	391
CAN INJ/TOTAL	0-48.3 km	1.0000	3.27E+02	2.98E+02	5.42E+02	6.14E+02	7.75E+02	8.40E+02	1.07E+03	1.12E-03	391
CAN INJ/TOTAL	0-64.4 km	1.0000	5.65E+02	4.86E+02	9.83E+02	1.17E+03	1.75E+03	2.08E+03	3.03E+03	1.15E-03	311
CAN INJ/TOTAL	0-80.5 km	1.0000	7.45E+02	6.56E+02	1.26E+03	1.55E+03	2.37E+03	2.76E+03	3.88E+03	1.15E-03	311
CAN INJ/TOTAL	0-161 km	1.0000	1.51E+03	1.15E+03	2.83E+03	3.60E+03	5.36E+03	5.81E+03	6.89E+03	1.14E-03	118
CAN INJ/TOTAL	0-322 km	1.0000	2.24E+03	1.53E+03	4.66E+03	6.06E+03	9.38E+03	1.10E+04	1.52E+04	1.15E-03	711
CAN INJ/TOTAL	0-805 km	1.0000	2.63E+03	2.04E+03	5.35E+03	6.67E+03	9.73E+03	1.13E+04	1.54E+04	1.15E-03	711
CAN INJ/TOTAL	0-1609 km	1.0000	2.70E+03	2.10E+03	5.43E+03	6.69E+03	9.73E+03	1.13E+04	1.54E+04	1.15E-03	711
CAN FAT/TOTAL	0-16.1 km	1.0000	1.18E+01	1.06E+01	1.63E+01	1.97E+01	2.46E+01	2.70E+01	3.33E+01	1.12E-03	391
CAN FAT/TOTAL	0-32.2 km	1.0000	6.66E+01	5.74E+01	1.09E+02	1.19E+02	1.49E+02	1.63E+02	2.28E+02	1.12E-03	391
CAN FAT/TOTAL	0-48.3 km	1.0000	1.44E+02	1.19E+02	2.34E+02	2.74E+02	3.51E+02	3.85E+02	4.68E+02	1.12E-03	391
CAN FAT/TOTAL	0-64.4 km	1.0000	2.48E+02	2.14E+02	4.28E+02	5.35E+02	7.99E+02	9.28E+02	1.33E+03	1.15E-03	311
CAN FAT/TOTAL	0-80.5 km	1.0000	3.26E+02	2.88E+02	5.68E+02	7.15E+02	1.05E+03	1.23E+03	1.70E+03	1.15E-03	311

CAN FAT/TOTAL	0-161 km	1.0000	6.59E+02	5.16E+02	1.25E+03	1.65E+03	2.30E+03	2.50E+03	3.02E+03	1.14E-03	118
CAN FAT/TOTAL	0-322 km	1.0000	9.77E+02	7.03E+02	2.01E+03	2.66E+03	4.25E+03	5.08E+03	6.67E+03	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	1.14E+03	8.78E+02	2.29E+03	2.88E+03	4.31E+03	5.08E+03	6.75E+03	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	1.18E+03	9.13E+02	2.31E+03	2.88E+03	4.31E+03	5.08E+03	6.75E+03	1.15E-03	711
CAN FAT/THYROID	0-16.1 km	1.0000	6.69E-02	6.19E-02	1.00E-01	1.12E-01	1.43E-01	1.59E-01	2.05E-01	1.12E-03	391
CAN FAT/THYROID	0-80.5 km	1.0000	2.02E+00	1.65E+00	3.48E+00	4.30E+00	6.10E+00	6.91E+00	1.06E+01	1.15E-03	311
CAN FAT/THYROID	0-161 km	1.0000	4.07E+00	3.15E+00	7.70E+00	9.92E+00	1.30E+01	1.46E+01	1.87E+01	1.14E-03	118
CAN FAT/THYROID	0-1609 km	1.0000	7.18E+00	5.46E+00	1.38E+01	1.79E+01	2.64E+01	3.05E+01	4.20E+01	1.15E-03	711
CAN FAT/BREAST	0-16.1 km	1.0000	9.82E-01	9.25E-01	1.34E+00	1.54E+00	2.08E+00	2.29E+00	2.81E+00	1.12E-03	391
CAN FAT/BREAST	0-80.5 km	1.0000	3.02E+01	2.56E+01	5.21E+01	6.63E+01	9.07E+01	1.04E+02	1.62E+02	1.15E-03	311
CAN FAT/BREAST	0-161 km	1.0000	6.05E+01	4.54E+01	1.15E+02	1.45E+02	2.18E+02	2.39E+02	2.90E+02	1.14E-03	118
CAN FAT/BREAST	0-1609 km	1.0000	1.04E+02	7.77E+01	2.10E+02	2.74E+02	4.27E+02	5.07E+02	6.53E+02	1.15E-03	711
CAN FAT/LUNG	0-16.1 km	1.0000	1.91E+00	1.72E+00	2.81E+00	3.19E+00	4.03E+00	4.45E+00	5.57E+00	1.12E-03	391
CAN FAT/LUNG	0-80.5 km	1.0000	5.79E+01	5.08E+01	1.00E+02	1.17E+02	1.68E+02	1.97E+02	3.07E+02	1.15E-03	311
CAN FAT/LUNG	0-161 km	1.0000	1.17E+02	8.89E+01	2.24E+02	2.87E+02	4.04E+02	4.64E+02	5.46E+02	1.14E-03	118
CAN FAT/LUNG	0-1609 km	1.0000	2.05E+02	1.46E+02	4.03E+02	5.19E+02	7.98E+02	9.75E+02	1.23E+03	1.15E-03	711
CAN FAT/LEUKEMIA	0-1609 km	1.0000	1.24E+02	9.87E+01	2.44E+02	3.08E+02	4.52E+02	5.27E+02	7.12E+02	1.15E-03	711
CAN FAT/BONE	0-1609 km	1.0000	3.02E+00	2.26E+00	5.98E+00	7.52E+00	1.12E+01	1.30E+01	1.78E+01	1.15E-03	711
CAN FAT/LIVER	0-1609 km	1.0000	3.03E+01	2.28E+01	5.98E+01	7.41E+01	1.10E+02	1.28E+02	1.76E+02	1.15E-03	711
CAN FAT/COLON	0-1609 km	1.0000	2.24E+02	1.71E+02	4.39E+02	5.45E+02	8.08E+02	9.76E+02	1.25E+03	1.15E-03	711
CAN FAT/RESIDUAL	0-1609 km	1.0000	4.78E+02	3.56E+02	9.65E+02	1.16E+03	1.72E+03	2.03E+03	2.67E+03	1.15E-03	711

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
EARLY FATALITY DISTANCE (km)											
ERL FAT/TOTAL RISK > 0.000		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
POPULATION EXCEEDING DOSE											
EARLY dose A-RED MARR > 2.32 Sv		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-LUNGS > 13.6 Sv		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-STOMACH > 6.50 Sv		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
POPULATION DOSE (Sv)											
L-ICRP60ED TOT LIF	0-16.1 km	1.0000	2.87E+02	2.76E+02	3.96E+02	4.51E+02	5.54E+02	5.93E+02	6.86E+02	1.13E-03	247
L-ICRP60ED TOT LIF	0-80.5 km	1.0000	6.27E+03	5.44E+03	1.05E+04	1.24E+04	1.84E+04	2.16E+04	3.26E+04	1.15E-03	311
L-ICRP60ED TOT LIF	0-161 km	1.0000	1.25E+04	9.65E+03	2.38E+04	3.09E+04	4.56E+04	5.13E+04	5.79E+04	1.14E-03	118
L-ICRP60ED TOT LIF	0-1609 km	1.0000	2.22E+04	1.65E+04	4.37E+04	5.47E+04	8.35E+04	1.01E+05	1.30E+05	1.15E-03	711

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
POPULATION WEIGHTED RISK											
CAN FAT/TOTAL	0-16.1 km	1.0000	1.72E-04	1.46E-04	2.53E-04	2.92E-04	3.60E-04	3.93E-04	4.73E-04	1.12E-03	391
CAN FAT/TOTAL	0-32.2 km	1.0000	1.17E-04	1.01E-04	2.06E-04	2.28E-04	2.87E-04	3.13E-04	3.67E-04	1.12E-03	391
CAN FAT/TOTAL	0-48.3 km	1.0000	8.57E-05	7.45E-05	1.31E-04	1.55E-04	2.09E-04	2.23E-04	2.54E-04	1.12E-03	391
CAN FAT/TOTAL	0-64.4 km	1.0000	6.04E-05	5.13E-05	1.07E-04	1.32E-04	2.09E-04	2.35E-04	3.56E-04	1.15E-03	311
CAN FAT/TOTAL	0-80.5 km	1.0000	5.02E-05	4.14E-05	9.00E-05	1.10E-04	1.65E-04	1.96E-04	2.84E-04	1.15E-03	311
CAN FAT/TOTAL	0-161 km	1.0000	2.84E-05	2.09E-05	5.82E-05	7.50E-05	1.09E-04	1.20E-04	1.47E-04	1.14E-03	118
CAN FAT/TOTAL	0-322 km	1.0000	1.61E-05	1.00E-05	3.64E-05	5.12E-05	8.26E-05	1.02E-04	1.32E-04	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	7.79E-06	5.09E-06	1.71E-05	2.29E-05	3.68E-05	4.49E-05	5.93E-05	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	4.45E-06	2.89E-06	9.67E-06	1.23E-05	2.06E-05	2.46E-05	3.35E-05	1.15E-03	711
CAN FAT/TOTAL	16.1-32.2 km	1.0000	1.11E-04	9.00E-05	2.07E-04	2.31E-04	2.98E-04	3.16E-04	3.56E-04	1.12E-03	391
CAN FAT/TOTAL	32.2-48.3 km	1.0000	7.02E-05	6.24E-05	1.18E-04	1.35E-04	1.87E-04	2.08E-04	2.49E-04	1.15E-03	311
CAN FAT/TOTAL	48.3-64.4 km	1.0000	4.35E-05	2.58E-05	1.03E-04	1.36E-04	2.34E-04	2.76E-04	4.42E-04	1.15E-03	311
CAN FAT/TOTAL	64.4-80.5 km	1.0000	3.25E-05	2.33E-05	7.17E-05	9.10E-05	1.25E-04	1.41E-04	1.81E-04	1.13E-03	714
CAN FAT/TOTAL	80.5-161 km	1.0000	1.95E-05	9.50E-06	5.02E-05	6.87E-05	1.13E-04	1.30E-04	1.75E-04	1.14E-03	118
CAN FAT/TOTAL	161-322 km	1.0000	8.24E-06	1.98E-06	2.45E-05	3.93E-05	7.55E-05	1.07E-04	1.52E-04	1.15E-03	711
CAN FAT/TOTAL	322-805 km	0.9920	1.12E-06	7.21E-07	2.80E-06	3.52E-06	5.26E-06	5.77E-06	8.83E-06	1.14E-03	400
CAN FAT/TOTAL	805-1609 km	0.9071	1.19E-07	6.56E-10	3.71E-07	7.50E-07	1.70E-06	2.07E-06	2.47E-06	1.14E-03	825

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)											
L-ICRP60ED	0-0.2 km	1.0000	6.00E-02	5.14E-02	5.56E-02	5.76E-02	6.24E-02	6.45E-02	6.94E-02	1.14E-03	315
L-ICRP60ED	0.2-0.5 km	1.0000	6.27E-02	5.14E-02	5.50E-02	5.66E-02	6.05E-02	6.23E-02	6.70E-02	8.56E-04	292
L-ICRP60ED	0.5-1.2 km	1.0000	6.14E-02	5.13E-02	5.50E-02	5.66E-02	6.06E-02	6.24E-02	6.65E-02	1.12E-03	759
L-ICRP60ED	1.2-1.6 km	1.0000	5.95E-02	5.12E-02	5.48E-02	5.65E-02	6.05E-02	6.23E-02	6.64E-02	1.13E-03	286
L-ICRP60ED	1.6-2.1 km	1.0000	5.77E-02	5.11E-02	5.48E-02	5.65E-02	6.05E-02	6.24E-02	6.65E-02	1.13E-03	696
L-ICRP60ED	2.1-3.2 km	1.0000	5.41E-02	5.07E-02	5.44E-02	5.61E-02	6.02E-02	6.20E-02	6.62E-02	1.14E-03	947
L-ICRP60ED	3.2-4.0 km	1.0000	5.05E-02	5.03E-02	5.41E-02	5.58E-02	6.00E-02	6.19E-02	6.62E-02	1.14E-03	166
L-ICRP60ED	4.0-4.8 km	1.0000	4.66E-02	4.58E-02	5.36E-02	5.54E-02	5.98E-02	6.18E-02	6.62E-02	1.14E-03	978
L-ICRP60ED	4.8-5.6 km	1.0000	4.32E-02	3.95E-02	5.26E-02	5.43E-02	5.84E-02	6.02E-02	6.62E-02	5.99E-04	683
L-ICRP60ED	5.6-8.1 km	1.0000	3.73E-02	3.35E-02	5.04E-02	5.25E-02	5.78E-02	6.02E-02	6.57E-02	1.13E-03	394
L-ICRP60ED	8.1-11.3 km	1.0000	3.14E-02	3.04E-02	3.84E-02	4.25E-02	5.09E-02	5.23E-02	5.98E-02	1.52E-04	326
L-ICRP60ED	11.3-16.1 km	1.0000	2.79E-02	2.51E-02	3.30E-02	3.50E-02	4.01E-02	4.25E-02	4.83E-02	1.12E-03	296
L-ICRP60ED	16.1-20.9 km	1.0000	2.87E-02	2.70E-02	3.37E-02	3.57E-02	4.09E-02	4.33E-02	4.91E-02	1.12E-03	296
L-ICRP60ED	20.9-25.8 km	1.0000	2.68E-02	2.40E-02	3.20E-02	3.36E-02	3.77E-02	3.96E-02	4.40E-02	1.12E-03	296
L-ICRP60ED	25.8-32.2 km	1.0000	2.50E-02	2.23E-02	3.03E-02	3.13E-02	3.38E-02	3.49E-02	3.75E-02	1.15E-03	695
L-ICRP60ED	32.2-40.2 km	1.0000	2.26E-02	2.12E-02	2.89E-02	3.05E-02	3.20E-02	3.27E-02	3.65E-02	1.52E-04	94
L-ICRP60ED	40.2-48.3 km	1.0000	1.98E-02	2.01E-02	2.59E-02	2.88E-02	3.21E-02	3.32E-02	3.58E-02	1.13E-03	359
L-ICRP60ED	48.3-64.4 km	1.0000	1.69E-02	1.52E-02	2.37E-02	2.58E-02	3.08E-02	3.23E-02	3.58E-02	1.13E-03	157
L-ICRP60ED	64.4-80.5 km	1.0000	1.32E-02	1.12E-02	2.19E-02	2.38E-02	2.88E-02	3.01E-02	3.07E-02	1.14E-03	328
L-ICRP60ED	80.5-113 km	1.0000	9.21E-03	7.17E-03	1.83E-02	2.10E-02	2.44E-02	2.61E-02	3.04E-02	1.13E-03	479
L-ICRP60ED	113-161 km	1.0000	5.99E-03	4.24E-03	1.17E-02	1.50E-02	2.12E-02	2.23E-02	2.91E-02	1.14E-04	88
L-ICRP60ED	161-241 km	1.0000	3.20E-03	2.34E-03	6.40E-03	8.53E-03	1.44E-02	1.78E-02	2.50E-02	1.14E-03	458
L-ICRP60ED	241-322 km	1.0000	1.75E-03	1.21E-03	3.53E-03	5.06E-03	7.95E-03	9.33E-03	1.48E-02	1.13E-03	317
L-ICRP60ED	322-563 km	1.0000	8.09E-04	6.30E-04	1.58E-03	2.10E-03	3.16E-03	3.66E-03	5.02E-03	1.13E-03	314
L-ICRP60ED	563-805 km	1.0000	5.13E-04	3.19E-04	1.14E-03	1.38E-03	2.08E-03	2.30E-03	2.87E-03	1.15E-03	508
L-ICRP60ED	805-1609 km	1.0000	1.06E-04	6.54E-05	2.69E-04	3.26E-04	4.39E-04	4.99E-04	7.61E-04	3.04E-04	633

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)											

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
GROUND CONC. (Bq/m2)											
AREA (ha) THAT EXCEEDS THRESHOLD											
Cs-137 Area exceeds	3.70E+04 Bq/m2	1.0000	1.69E+04	1.44E+04	2.54E+04	2.95E+04	3.49E+04	3.73E+04	4.32E+04	1.14E-03	936
AREA (ha) THAT EXCEEDS THRESHOLD											
Cs-137 Area exceeds	1.85E+05 Bq/m2	1.0000	1.27E+04	1.09E+04	1.79E+04	2.07E+04	2.42E+04	2.59E+04	3.45E+04	1.12E-03	391
AREA (ha) THAT EXCEEDS THRESHOLD											
Cs-137 Area exceeds	5.55E+05 Bq/m2	1.0000	9.48E+03	8.69E+03	1.25E+04	1.41E+04	1.83E+04	2.02E+04	2.40E+04	1.14E-04	78
AREA (ha) THAT EXCEEDS THRESHOLD											
Cs-137 Area exceeds	1.48E+06 Bq/m2	1.0000	6.18E+03	5.76E+03	1.03E+04	1.09E+04	1.24E+04	1.31E+04	1.79E+04	1.14E-04	80
AREA (ha) THAT EXCEEDS THRESHOLD											

Cs-137 Area exceeds 3.70E+04 Bq/m2 1.0000 2.16E+05 2.01E+05 3.21E+05 3.59E+05 4.66E+05 5.12E+05 5.85E+05 1.11E-03 392
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.85E+05 Bq/m2 1.0000 8.73E+04 7.89E+04 1.33E+05 1.56E+05 2.08E+05 2.20E+05 2.87E+05 1.52E-04 416
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 5.55E+05 Bq/m2 1.0000 3.23E+04 2.90E+04 5.92E+04 7.37E+04 1.02E+05 1.05E+05 1.26E+05 1.43E-04 10
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.48E+06 Bq/m2 1.0000 1.12E+04 1.02E+04 2.09E+04 2.55E+04 3.60E+04 4.07E+04 6.73E+04 1.43E-04 76
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 3.70E+04 Bq/m2 1.0000 4.78E+05 4.13E+05 7.56E+05 8.96E+05 1.14E+06 1.23E+06 1.45E+06 1.14E-03 720
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.85E+05 Bq/m2 1.0000 1.12E+05 9.15E+04 2.08E+05 2.64E+05 3.58E+05 3.95E+05 4.88E+05 1.13E-03 320
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 5.55E+05 Bq/m2 1.0000 3.40E+04 2.95E+04 6.34E+04 8.39E+04 1.22E+05 1.37E+05 1.77E+05 1.14E-03 295
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.48E+06 Bq/m2 1.0000 1.12E+04 1.02E+04 2.10E+04 2.57E+04 3.62E+04 4.08E+04 6.73E+04 1.43E-04 76
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 3.70E+04 Bq/m2 1.0000 6.99E+05 5.89E+05 1.20E+06 1.43E+06 2.10E+06 2.35E+06 3.12E+06 1.14E-03 70
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.85E+05 Bq/m2 1.0000 1.16E+05 9.30E+04 2.20E+05 2.80E+05 3.63E+05 3.99E+05 4.88E+05 1.13E-03 320
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 5.55E+05 Bq/m2 1.0000 3.40E+04 2.95E+04 6.34E+04 8.39E+04 1.22E+05 1.37E+05 1.77E+05 1.14E-03 295
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.48E+06 Bq/m2 1.0000 1.12E+04 1.02E+04 2.10E+04 2.57E+04 3.62E+04 4.08E+04 6.73E+04 1.43E-04 76

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 low density no spray
"EARLY" DESCRIPTION = OCP3 low density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 low density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 1 = Group 1

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
HEALTH EFFECTS CASES
ERL FAT/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
ERL FAT/TOTAL 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
ERL FAT/TOTAL 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-32.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-48.3 km 1.0000 3.49E+00 2.61E+00 7.10E+00 9.10E+00 1.24E+01 1.38E+01 1.74E+01 1.15E-03 324
CAN INJ/TOTAL 0-64.4 km 1.0000 7.03E+00 5.48E+00 1.32E+01 1.69E+01 3.02E+01 3.48E+01 6.14E+01 8.56E-04 303
CAN INJ/TOTAL 0-80.5 km 1.0000 9.39E+00 7.61E+00 1.79E+01 2.29E+01 3.51E+01 4.09E+01 6.90E+01 8.56E-04 303
CAN INJ/TOTAL 0-161 km 1.0000 1.84E+01 1.28E+01 3.74E+01 4.88E+01 7.38E+01 8.13E+01 1.08E+02 1.14E-03 548
CAN INJ/TOTAL 0-322 km 1.0000 2.72E+01 1.60E+01 6.34E+01 8.72E+01 1.38E+02 1.64E+02 2.08E+02 1.15E-03 711
CAN INJ/TOTAL 0-805 km 1.0000 2.97E+01 1.94E+01 7.11E+01 9.49E+01 1.42E+02 1.67E+02 2.11E+02 1.15E-03 711
CAN INJ/TOTAL 0-1609 km 1.0000 3.00E+01 1.99E+01 7.11E+01 9.49E+01 1.42E+02 1.67E+02 2.11E+02 1.15E-03 711
CAN FAT/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-32.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-48.3 km 1.0000 1.61E+00 1.21E+00 3.26E+00 4.09E+00 6.81E+00 7.27E+00 7.99E+00 1.15E-03 681
CAN FAT/TOTAL 0-64.4 km 1.0000 3.24E+00 2.52E+00 6.41E+00 8.05E+00 1.28E+01 1.55E+01 2.83E+01 8.56E-04 303
CAN FAT/TOTAL 0-80.5 km 1.0000 4.32E+00 3.44E+00 8.37E+00 1.03E+01 1.44E+01 1.66E+01 3.19E+01 8.56E-04 303
CAN FAT/TOTAL 0-161 km 1.0000 8.46E+00 6.21E+00 1.75E+01 2.27E+01 3.38E+01 3.83E+01 5.00E+01 1.14E-03 548
CAN FAT/TOTAL 0-322 km 1.0000 1.26E+01 7.67E+00 2.99E+01 3.87E+01 6.58E+01 7.56E+01 9.58E+01 1.15E-03 711
CAN FAT/TOTAL 0-805 km 1.0000 1.37E+01 8.79E+00 3.17E+01 4.21E+01 6.85E+01 7.71E+01 9.72E+01 1.15E-03 711
CAN FAT/TOTAL 0-1609 km 1.0000 1.38E+01 9.05E+00 3.18E+01 4.22E+01 6.85E+01 7.71E+01 9.72E+01 1.15E-03 711
CAN FAT/THYROID 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/THYROID 0-80.5 km 1.0000 2.53E-02 2.05E-02 4.90E-02 6.02E-02 9.42E-02 1.14E-01 1.86E-01 8.56E-04 303
CAN FAT/THYROID 0-161 km 1.0000 4.93E-02 3.57E-02 1.04E-01 1.26E-01 2.00E-01 2.25E-01 2.90E-01 1.14E-03 548
CAN FAT/THYROID 0-1609 km 1.0000 7.98E-02 5.26E-02 1.73E-01 2.36E-01 3.82E-01 4.55E-01 5.60E-01 1.15E-03 711
CAN FAT/BREAST 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/BREAST 0-80.5 km 1.0000 2.93E-01 2.33E-01 5.72E-01 7.01E-01 1.12E+00 1.32E+00 2.12E+00 8.56E-04 303
CAN FAT/BREAST 0-161 km 1.0000 5.72E-01 4.12E-01 1.17E+00 1.47E+00 2.28E+00 2.59E+00 3.64E+00 1.14E-03 118
CAN FAT/BREAST 0-1609 km 1.0000 9.26E-01 6.10E-01 2.15E+00 2.90E+00 4.36E+00 5.11E+00 6.57E+00 1.15E-03 711
CAN FAT/LUNG 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/LUNG 0-80.5 km 1.0000 8.36E-01 6.84E-01 1.54E+00 2.01E+00 3.03E+00 3.76E+00 6.14E+00 8.56E-04 303
CAN FAT/LUNG 0-161 km 1.0000 1.64E+00 1.18E+00 3.38E+00 4.25E+00 6.57E+00 7.54E+00 9.70E+00 1.14E-03 548
CAN FAT/LUNG 0-1609 km 1.0000 2.69E+00 1.67E+00 6.09E+00 8.23E+00 1.24E+01 1.42E+01 1.89E+01 1.15E-03 711
CAN FAT/LEUKEMIA 0-1609 km 1.0000 2.11E+00 1.32E+00 5.10E+00 6.67E+00 1.01E+01 1.13E+01 1.46E+01 1.15E-03 711
CAN FAT/BONE 0-1609 km 1.0000 6.14E-02 3.89E-02 1.34E-01 1.90E-01 2.95E-01 3.33E-01 4.27E-01 1.14E-03 548
CAN FAT/LIVER 0-1609 km 1.0000 3.01E-01 2.01E-01 6.88E-01 9.55E-01 1.42E+00 1.67E+00 2.12E+00 1.15E-03 711
CAN FAT/COLON 0-1609 km 1.0000 2.68E+00 1.66E+00 6.06E+00 8.21E+00 1.24E+01 1.42E+01 1.88E+01 1.15E-03 711
CAN FAT/RESIDUAL 0-1609 km 1.0000 5.00E+00 3.30E+00 1.11E+01 1.44E+01 2.35E+01 2.75E+01 3.52E+01 1.15E-03 711

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
EARLY FATALITY DISTANCE (km)
ERL FAT/TOTAL RISK > 0.000 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
POPULATION EXCEEDING DOSE
EARLY dose A-RED MARR > 2.32 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
EARLY dose A-LUNGS > 13.6 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
EARLY dose A-STOMACH > 6.50 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
POPULATION DOSE (Sv)
L-ICRP60ED TOT LIF 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED TOT LIF 0-80.5 km 1.0000 7.79E+01 6.28E+01 1.41E+02 1.80E+02 3.02E+02 3.48E+02 5.73E+02 8.56E-04 303
L-ICRP60ED TOT LIF 0-161 km 1.0000 1.52E+02 1.11E+02 3.19E+02 4.02E+02 6.14E+02 7.09E+02 8.94E+02 1.14E-03 548
L-ICRP60ED TOT LIF 0-1609 km 1.0000 2.48E+02 1.56E+02 5.67E+02 7.55E+02 1.15E+03 1.31E+03 1.75E+03 1.15E-03 711

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
POPULATION WEIGHTED RISK
CAN FAT/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-32.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-48.3 km 1.0000 1.11E-06 8.46E-07 2.23E-06 2.85E-06 4.50E-06 5.09E-06 5.53E-06 1.15E-03 681
CAN FAT/TOTAL 0-64.4 km 1.0000 8.99E-07 7.12E-07 1.71E-06 2.24E-06 3.51E-06 4.09E-06 7.87E-06 8.56E-04 303
CAN FAT/TOTAL 0-80.5 km 1.0000 7.62E-07 6.13E-07 1.38E-06 1.74E-06 2.82E-06 3.34E-06 5.62E-06 8.56E-04 303
CAN FAT/TOTAL 0-161 km 1.0000 4.32E-07 3.20E-07 9.18E-07 1.14E-06 1.71E-06 2.02E-06 2.55E-06 1.14E-03 548
CAN FAT/TOTAL 0-322 km 1.0000 2.51E-07 1.45E-07 5.87E-07 7.81E-07 1.21E-06 1.40E-06 1.91E-06 1.15E-03 711

CAN FAT/TOTAL	0-805 km	1.0000	1.22E-07	7.85E-08	2.83E-07	3.68E-07	5.89E-07	6.89E-07	8.62E-07	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	6.92E-08	4.46E-08	1.49E-07	2.08E-07	3.09E-07	3.58E-07	4.86E-07	1.15E-03	711
CAN FAT/TOTAL	16.1-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	32.2-48.3 km	1.0000	1.67E-06	1.24E-06	3.31E-06	4.16E-06	7.06E-06	7.43E-06	8.28E-06	1.15E-03	681
CAN FAT/TOTAL	48.3-64.4 km	1.0000	7.56E-07	4.60E-07	1.73E-06	2.38E-06	4.52E-06	5.59E-06	9.98E-06	8.56E-04	303
CAN FAT/TOTAL	64.4-80.5 km	1.0000	5.24E-07	3.78E-07	1.13E-06	1.42E-06	2.23E-06	2.56E-06	3.40E-06	1.14E-03	548
CAN FAT/TOTAL	80.5-161 km	1.0000	2.97E-07	1.39E-07	7.52E-07	1.04E-06	1.93E-06	2.20E-06	2.79E-06	1.14E-03	118
CAN FAT/TOTAL	161-322 km	1.0000	1.35E-07	3.37E-08	4.53E-07	6.92E-07	1.13E-06	1.36E-06	2.03E-06	1.15E-03	711
CAN FAT/TOTAL	322-805 km	0.9920	1.83E-08	1.21E-08	4.35E-08	5.64E-08	9.30E-08	1.07E-07	1.30E-07	1.15E-03	544
CAN FAT/TOTAL	805-1609 km	0.9071	1.53E-09	8.11E-12	4.92E-09	9.44E-09	2.11E-08	2.31E-08	2.81E-08	1.14E-03	825

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)											
L-ICRP60ED	0-0.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.2-0.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.5-1.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.2-1.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.6-2.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	2.1-3.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	3.2-4.0 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.0-4.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.8-5.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	5.6-8.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	8.1-11.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	11.3-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	16.1-20.9 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	20.9-25.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	25.8-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	32.2-40.2 km	1.0000	8.19E-04	6.14E-04	1.55E-03	2.01E-03	2.93E-03	3.48E-03	5.01E-03	1.15E-03	324
L-ICRP60ED	40.2-48.3 km	1.0000	5.60E-04	4.28E-04	1.10E-03	1.36E-03	2.09E-03	2.25E-03	2.65E-03	1.15E-03	129
L-ICRP60ED	48.3-64.4 km	1.0000	3.67E-04	2.93E-04	7.11E-04	9.13E-04	1.31E-03	1.50E-03	2.10E-03	1.14E-03	723
L-ICRP60ED	64.4-80.5 km	1.0000	2.38E-04	1.87E-04	4.53E-04	5.86E-04	8.89E-04	1.05E-03	1.59E-03	1.14E-03	463
L-ICRP60ED	80.5-113 km	1.0000	1.46E-04	1.10E-04	2.75E-04	3.47E-04	5.39E-04	6.09E-04	9.65E-04	1.12E-03	321
L-ICRP60ED	113-161 km	1.0000	8.81E-05	7.08E-05	1.60E-04	2.05E-04	2.73E-04	3.10E-04	4.27E-04	1.15E-03	585
L-ICRP60ED	161-241 km	1.0000	4.84E-05	3.86E-05	9.21E-05	1.10E-04	1.52E-04	1.75E-04	2.22E-04	3.04E-04	640
L-ICRP60ED	241-322 km	1.0000	2.86E-05	2.29E-05	5.44E-05	6.80E-05	1.07E-04	1.21E-04	1.97E-04	3.04E-04	640
L-ICRP60ED	322-563 km	1.0000	1.39E-05	1.14E-05	2.55E-05	3.14E-05	4.40E-05	5.07E-05	8.79E-05	3.04E-04	638
L-ICRP60ED	563-805 km	1.0000	6.94E-06	5.65E-06	1.30E-05	1.63E-05	2.69E-05	3.10E-05	4.39E-05	1.52E-04	507
L-ICRP60ED	805-1609 km	1.0000	1.43E-06	9.05E-07	3.54E-06	4.40E-06	5.94E-06	6.56E-06	1.31E-05	3.04E-04	633

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)											
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	1.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD											
A-THYROID Area exceeds	5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 low density no spray
"EARLY" DESCRIPTION = OCP3 low density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 low density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 2 = Group 2

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
HEALTH EFFECTS CASES											
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-32.2 km	1.0000	4.12E+00	2.99E+00	8.75E+00	1.12E+01	1.68E+01	2.00E+01	2.54E+01	1.13E-03	135
CAN INJ/TOTAL	0-48.3 km	1.0000	7.65E+00	5.84E+00	1.43E+01	1.92E+01	3.02E+01	3.28E+01	3.90E+01	1.13E-03	135
CAN INJ/TOTAL	0-64.4 km	1.0000	1.12E+01	8.89E+00	2.15E+01	2.68E+01	3.75E+01	4.25E+01	6.94E+01	8.56E-04	303
CAN INJ/TOTAL	0-80.5 km	1.0000	1.36E+01	1.09E+01	2.50E+01	3.09E+01	4.55E+01	5.25E+01	7.71E+01	8.56E-04	303
CAN INJ/TOTAL	0-161 km	1.0000	2.27E+01	1.81E+01	4.27E+01	5.39E+01	7.55E+01	8.26E+01	1.11E+02	1.14E-03	548
CAN INJ/TOTAL	0-322 km	1.0000	3.17E+01	2.18E+01	6.86E+01	9.18E+01	1.40E+02	1.66E+02	2.11E+02	1.15E-03	711
CAN INJ/TOTAL	0-805 km	1.0000	3.42E+01	2.43E+01	7.45E+01	9.72E+01	1.50E+02	1.81E+02	2.14E+02	1.15E-03	711
CAN INJ/TOTAL	0-1609 km	1.0000	3.45E+01	2.47E+01	7.45E+01	9.72E+01	1.50E+02	1.81E+02	2.14E+02	1.15E-03	711
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	1.0000	1.88E+00	1.30E+00	3.89E+00	5.28E+00	7.59E+00	8.28E+00	1.16E+01	1.13E-03	135
CAN FAT/TOTAL	0-48.3 km	1.0000	3.49E+00	2.67E+00	7.18E+00	8.70E+00	1.22E+01	1.37E+01	1.79E+01	1.13E-03	135
CAN FAT/TOTAL	0-64.4 km	1.0000	5.13E+00	3.99E+00	9.39E+00	1.13E+01	1.64E+01	1.93E+01	3.17E+01	8.56E-04	303
CAN FAT/TOTAL	0-80.5 km	1.0000	6.22E+00	5.15E+00	1.12E+01	1.34E+01	2.01E+01	2.25E+01	3.53E+01	8.56E-04	303
CAN FAT/TOTAL	0-161 km	1.0000	1.04E+01	8.42E+00	1.99E+01	2.43E+01	3.47E+01	3.90E+01	5.09E+01	1.14E-03	548
CAN FAT/TOTAL	0-322 km	1.0000	1.45E+01	1.00E+01	3.15E+01	4.13E+01	6.84E+01	7.69E+01	9.65E+01	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	1.56E+01	1.12E+01	3.31E+01	4.32E+01	6.86E+01	7.73E+01	9.79E+01	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	1.58E+01	1.14E+01	3.32E+01	4.32E+01	6.86E+01	7.73E+01	9.79E+01	1.15E-03	711
CAN FAT/THYROID	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-80.5 km	1.0000	5.14E-02	4.11E-02	9.27E-02	1.09E-01	1.48E-01	1.68E-01	2.95E-01	8.56E-04	303
CAN FAT/THYROID	0-161 km	1.0000	8.51E-02	6.83E-02	1.56E-01	2.03E-01	2.92E-01	3.32E-01	4.30E-01	1.14E-03	548
CAN FAT/THYROID	0-1609 km	1.0000	1.27E-01	9.12E-02	2.58E-01	3.37E-01	5.41E-01	6.39E-01	7.81E-01	1.15E-03	711
CAN FAT/BREAST	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-80.5 km	1.0000	4.19E-01	3.40E-01	7.71E-01	9.48E-01	1.29E+00	1.46E+00	2.34E+00	8.56E-04	303
CAN FAT/BREAST	0-161 km	1.0000	6.98E-01	5.63E-01	1.26E+00	1.56E+00	2.36E+00	2.73E+00	3.70E+00	1.14E-03	118
CAN FAT/BREAST	0-1609 km	1.0000	1.05E+00	7.74E-01	2.25E+00	2.94E+00	4.38E+00	5.13E+00	6.60E+00	1.15E-03	711
CAN FAT/LUNG	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-80.5 km	1.0000	1.20E+00	1.00E+00	2.21E+00	2.67E+00	3.73E+00	4.23E+00	6.78E+00	8.56E-04	303
CAN FAT/LUNG	0-161 km	1.0000	2.00E+00	1.54E+00	3.71E+00	4.68E+00	6.64E+00	7.56E+00	9.86E+00	1.14E-03	548
CAN FAT/LUNG	0-1609 km	1.0000	3.05E+00	2.21E+00	6.34E+00	8.35E+00	1.24E+01	1.42E+01	1.90E+01	1.15E-03	711
CAN FAT/LEUKEMIA	0-1609 km	1.0000	2.39E+00	1.65E+00	5.12E+00	6.73E+00	1.01E+01	1.14E+01	1.47E+01	1.15E-03	711
CAN FAT/BONE	0-1609 km	1.0000	6.98E-02	5.00E-02	1.39E-01	1.93E-01	3.02E-01	3.38E-01	4.30E-01	1.14E-03	548
CAN FAT/LIVER	0-1609 km	1.0000	3.42E-01	2.46E-01	7.42E-01	9.71E-01	1.42E+00	1.67E+00	2.14E+00	1.15E-03	711
CAN FAT/COLON	0-1609 km	1.0000	3.04E+00	2.19E+00	6.30E+00	8.33E+00	1.24E+01	1.42E+01	1.89E+01	1.15E-03	711
CAN FAT/RESIDUAL	0-1609 km	1.0000	5.67E+00	4.00E+00	1.15E+01	1.46E+01	2.35E+01	2.75E+01	3.54E+01	1.15E-03	711

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
EARLY FATALITY DISTANCE (km)											

ERL FAT/TOTAL RISK > 0.000	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION EXCEEDING DOSE

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
EARLY dose A-RED MARR > 2.32 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-LUNGS > 13.6 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-STOMACH > 6.50 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION DOSE (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED TOT LIF	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED TOT LIF	0-80.5 km	1.0000	1.14E+02	9.37E+01	2.11E+02	2.53E+02	3.63E+02	4.17E+02	6.47E+02	8.56E-04	303
L-ICRP60ED TOT LIF	0-161 km	1.0000	1.90E+02	1.44E+02	3.53E+02	4.39E+02	6.59E+02	7.46E+02	9.29E+02	1.14E-03	548
L-ICRP60ED TOT LIF	0-1609 km	1.0000	2.88E+02	2.09E+02	5.99E+02	7.70E+02	1.15E+03	1.33E+03	1.79E+03	1.15E-03	711

POPULATION WEIGHTED RISK

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	1.0000	3.93E-06	2.84E-06	8.03E-06	1.05E-05	1.42E-05	1.62E-05	2.43E-05	1.13E-03	135
CAN FAT/TOTAL	0-48.3 km	1.0000	2.42E-06	1.86E-06	4.90E-06	6.01E-06	9.20E-06	1.04E-05	1.24E-05	1.13E-03	135
CAN FAT/TOTAL	0-64.4 km	1.0000	1.42E-06	1.11E-06	2.63E-06	3.28E-06	5.08E-06	5.56E-06	8.82E-06	8.56E-04	303
CAN FAT/TOTAL	0-80.5 km	1.0000	1.10E-06	8.96E-07	2.04E-06	2.44E-06	3.56E-06	4.12E-06	6.23E-06	8.56E-04	303
CAN FAT/TOTAL	0-161 km	1.0000	5.29E-07	4.19E-07	1.01E-06	1.20E-06	1.80E-06	2.08E-06	2.60E-06	1.14E-03	548
CAN FAT/TOTAL	0-322 km	1.0000	2.89E-07	2.01E-07	6.09E-07	8.12E-07	1.24E-06	1.43E-06	1.93E-06	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	1.39E-07	1.00E-07	3.03E-07	3.77E-07	6.05E-07	7.14E-07	8.68E-07	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	7.88E-08	5.68E-08	1.58E-07	2.14E-07	3.24E-07	3.70E-07	4.90E-07	1.15E-03	711
CAN FAT/TOTAL	16.1-32.2 km	1.0000	4.34E-06	3.11E-06	9.58E-06	1.19E-05	1.84E-05	2.12E-05	2.68E-05	1.13E-03	135
CAN FAT/TOTAL	32.2-48.3 km	1.0000	1.67E-06	1.25E-06	3.33E-06	4.20E-06	7.06E-06	7.43E-06	8.30E-06	1.15E-03	681
CAN FAT/TOTAL	48.3-64.4 km	1.0000	7.58E-07	4.61E-07	1.73E-06	2.39E-06	4.54E-06	5.59E-06	1.00E-05	8.56E-04	303
CAN FAT/TOTAL	64.4-80.5 km	1.0000	5.25E-07	3.78E-07	1.14E-06	1.42E-06	2.23E-06	2.56E-06	3.41E-06	1.14E-03	548
CAN FAT/TOTAL	80.5-161 km	1.0000	2.98E-07	1.40E-07	7.57E-07	1.04E-06	1.93E-06	2.20E-06	2.79E-06	1.14E-03	118
CAN FAT/TOTAL	161-322 km	1.0000	1.35E-07	3.38E-08	4.53E-07	6.92E-07	1.15E-06	1.37E-06	2.03E-06	1.15E-03	711
CAN FAT/TOTAL	322-805 km	0.9920	1.83E-08	1.21E-08	4.35E-08	5.64E-08	9.30E-08	1.07E-07	1.30E-07	1.15E-03	544
CAN FAT/TOTAL	805-1609 km	0.9071	1.53E-09	8.11E-12	4.92E-09	9.44E-09	2.11E-08	2.31E-08	2.81E-08	1.14E-03	825

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.2-0.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.5-1.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.2-1.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.6-2.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	2.1-3.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	3.2-4.0 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.0-4.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.8-5.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	5.6-8.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	8.1-11.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	11.3-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	16.1-20.9 km	1.0000	2.84E-03	2.33E-03	5.83E-03	7.13E-03	8.18E-03	8.67E-03	9.85E-03	1.11E-03	312
L-ICRP60ED	20.9-25.8 km	1.0000	1.91E-03	1.42E-03	3.78E-03	4.84E-03	7.16E-03	7.95E-03	9.95E-03	1.13E-03	516
L-ICRP60ED	25.8-32.2 km	1.0000	1.27E-03	1.00E-03	2.57E-03	3.22E-03	4.73E-03	5.42E-03	7.50E-03	1.13E-03	517
L-ICRP60ED	32.2-40.2 km	1.0000	8.37E-04	6.34E-04	1.58E-03	2.04E-03	2.94E-03	3.48E-03	5.08E-03	1.15E-03	324
L-ICRP60ED	40.2-48.3 km	1.0000	5.72E-04	4.38E-04	1.12E-03	1.38E-03	2.09E-03	2.27E-03	2.71E-03	1.15E-03	129
L-ICRP60ED	48.3-64.4 km	1.0000	3.75E-04	3.01E-04	7.15E-04	9.38E-04	1.32E-03	1.51E-03	2.14E-03	1.14E-03	723
L-ICRP60ED	64.4-80.5 km	1.0000	2.43E-04	1.91E-04	4.62E-04	6.02E-04	8.96E-04	1.05E-03	1.61E-03	1.14E-03	463
L-ICRP60ED	80.5-113 km	1.0000	1.49E-04	1.13E-04	2.82E-04	3.55E-04	5.44E-04	6.12E-04	9.75E-04	1.12E-03	321
L-ICRP60ED	113-161 km	1.0000	8.99E-05	7.30E-05	1.64E-04	2.09E-04	2.97E-04	3.36E-04	4.36E-04	1.15E-03	585
L-ICRP60ED	161-241 km	1.0000	4.94E-05	3.91E-05	9.37E-05	1.11E-04	1.53E-04	1.75E-04	2.27E-04	3.04E-04	640
L-ICRP60ED	241-322 km	1.0000	2.92E-05	2.34E-05	5.51E-05	6.91E-05	1.07E-04	1.21E-04	2.02E-04	3.04E-04	640
L-ICRP60ED	322-563 km	1.0000	1.41E-05	1.16E-05	2.59E-05	3.17E-05	4.41E-05	5.07E-05	8.99E-05	3.04E-04	638
L-ICRP60ED	563-805 km	1.0000	7.05E-06	5.77E-06	1.31E-05	1.64E-05	2.69E-05	3.11E-05	4.47E-05	1.52E-04	507
L-ICRP60ED	805-1609 km	1.0000	1.45E-06	9.11E-07	3.58E-06	4.44E-06	5.99E-06	6.64E-06	1.33E-05	3.04E-04	632

DOSE FOUND AT ALL LOCATIONS (Sv)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
AREA (ha) THAT EXCEEDS THRESHOLD										
L-ICRP60ED Area exceeds 1.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD										
L-ICRP60ED Area exceeds 5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD										
A-THYROID Area exceeds 5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 low density no spray
"EARLY" DESCRIPTION = OCP3 low density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 low density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 3 = Group 3

HEALTH EFFECTS CASES

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-32.2 km	1.0000	2.22E+00	1.52E+00	4.93E+00	6.26E+00	9.76E+00	1.11E+01	1.41E+01	1.13E-03	135
CAN INJ/TOTAL	0-48.3 km	1.0000	4.11E+00	3.17E+00	8.04E+00	1.04E+01	1.54E+01	1.82E+01	2.18E+01	1.13E-03	135
CAN INJ/TOTAL	0-64.4 km	1.0000	6.02E+00	4.86E+00	1.13E+01	1.39E+01	2.12E+01	2.34E+01	3.80E+01	8.56E-04	303
CAN INJ/TOTAL	0-80.5 km	1.0000	7.30E+00	5.95E+00	1.31E+01	1.63E+01	2.33E+01	2.60E+01	4.24E+01	8.56E-04	303
CAN INJ/TOTAL	0-161 km	1.0000	1.22E+01	9.83E+00	2.31E+01	2.95E+01	4.02E+01	4.58E+01	6.28E+01	1.14E-03	548
CAN INJ/TOTAL	0-322 km	1.0000	1.72E+01	1.14E+01	3.59E+01	4.79E+01	7.72E+01	9.13E+01	1.16E+02	1.14E-03	576
CAN INJ/TOTAL	0-805 km	1.0000	1.86E+01	1.26E+01	3.80E+01	5.14E+01	8.28E+01	9.84E+01	1.17E+02	1.14E-03	548
CAN INJ/TOTAL	0-1609 km	1.0000	1.88E+01	1.28E+01	3.82E+01	5.14E+01	8.28E+01	9.84E+01	1.17E+02	1.14E-03	548
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	1.0000	1.04E+00	7.43E-01	2.16E+00	2.96E+00	4.69E+00	5.35E+00	6.63E+00	1.13E-03	135
CAN FAT/TOTAL	0-48.3 km	1.0000	1.92E+00	1.39E+00	3.68E+00	4.85E+00	7.32E+00	8.09E+00	1.02E+01	1.13E-03	135

CAN FAT/TOTAL	0-64.4 km	1.0000	2.82E+00	2.22E+00	5.44E+00	6.75E+00	1.01E+01	1.18E+01	1.78E+01	8.56E-04	303
CAN FAT/TOTAL	0-80.5 km	1.0000	3.41E+00	2.78E+00	6.20E+00	7.62E+00	1.17E+01	1.35E+01	1.99E+01	8.56E-04	303
CAN FAT/TOTAL	0-161 km	1.0000	5.71E+00	4.59E+00	1.08E+01	1.29E+01	1.94E+01	2.23E+01	2.94E+01	1.14E-03	548
CAN FAT/TOTAL	0-322 km	1.0000	8.05E+00	5.48E+00	1.68E+01	2.31E+01	3.78E+01	4.53E+01	5.42E+01	1.14E-03	576
CAN FAT/TOTAL	0-805 km	1.0000	8.71E+00	6.22E+00	1.87E+01	2.48E+01	3.88E+01	4.57E+01	5.50E+01	1.14E-03	548
CAN FAT/TOTAL	0-1609 km	1.0000	8.78E+00	6.30E+00	1.88E+01	2.48E+01	3.88E+01	4.57E+01	5.50E+01	1.14E-03	548
CAN FAT/THYROID	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-80.5 km	1.0000	1.96E-02	1.54E-02	3.60E-02	4.42E-02	5.96E-02	6.61E-02	1.14E-01	8.56E-04	303
CAN FAT/THYROID	0-161 km	1.0000	3.27E-02	2.63E-02	6.29E-02	7.79E-02	1.13E-01	1.28E-01	1.68E-01	1.14E-03	548
CAN FAT/THYROID	0-1609 km	1.0000	4.97E-02	3.53E-02	1.04E-01	1.29E-01	2.10E-01	2.65E-01	3.09E-01	1.14E-03	548
CAN FAT/BREAST	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-80.5 km	1.0000	2.00E-01	1.57E-01	3.62E-01	4.40E-01	6.43E-01	7.37E-01	1.15E+00	8.56E-04	303
CAN FAT/BREAST	0-161 km	1.0000	3.34E-01	2.69E-01	6.38E-01	7.93E-01	1.14E+00	1.29E+00	1.67E+00	1.14E-03	548
CAN FAT/BREAST	0-1609 km	1.0000	5.11E-01	3.63E-01	1.07E+00	1.32E+00	2.16E+00	2.67E+00	3.18E+00	1.15E-03	711
CAN FAT/LUNG	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-80.5 km	1.0000	6.71E-01	5.53E-01	1.22E+00	1.49E+00	2.23E+00	2.53E+00	3.88E+00	8.56E-04	303
CAN FAT/LUNG	0-161 km	1.0000	1.12E+00	9.09E-01	2.15E+00	2.61E+00	3.57E+00	3.97E+00	5.80E+00	1.14E-03	548
CAN FAT/LUNG	0-1609 km	1.0000	1.73E+00	1.21E+00	3.58E+00	4.84E+00	7.31E+00	8.95E+00	1.09E+01	1.14E-03	548
CAN FAT/LEUKEMIA	0-1609 km	1.0000	1.46E+00	1.03E+00	3.16E+00	4.11E+00	6.86E+00	7.61E+00	9.26E+00	1.14E-03	548
CAN FAT/BONE	0-1609 km	1.0000	4.37E-02	3.07E-02	9.33E-02	1.19E-01	1.96E-01	2.21E-01	2.78E-01	1.14E-03	548
CAN FAT/LIVER	0-1609 km	1.0000	1.80E-01	1.24E-01	3.68E-01	4.90E-01	7.61E-01	9.08E-01	1.12E+00	1.14E-03	576
CAN FAT/COLON	0-1609 km	1.0000	1.72E+00	1.20E+00	3.57E+00	4.84E+00	7.31E+00	8.95E+00	1.09E+01	1.14E-03	548
CAN FAT/RESIDUAL	0-1609 km	1.0000	3.08E+00	2.18E+00	6.63E+00	8.70E+00	1.27E+01	1.45E+01	1.92E+01	1.14E-03	548

EARLY FATALITY DISTANCE (km)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL RISK > 0.000	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION EXCEEDING DOSE

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
EARLY dose A-RED MARR > 2.32 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-LUNGS > 13.6 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-STOMACH > 6.50 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION DOSE (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED TOT LIF	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED TOT LIF	0-80.5 km	1.0000	6.04E+01	5.08E+01	1.10E+02	1.28E+02	1.83E+02	2.12E+02	3.51E+02	8.56E-04	303
L-ICRP60ED TOT LIF	0-161 km	1.0000	1.01E+02	8.10E+01	1.93E+02	2.34E+02	3.38E+02	3.83E+02	5.19E+02	1.14E-03	548
L-ICRP60ED TOT LIF	0-1609 km	1.0000	1.55E+02	1.09E+02	3.25E+02	4.33E+02	6.88E+02	7.71E+02	9.67E+02	1.14E-03	548

POPULATION WEIGHTED RISK

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	1.0000	2.17E-06	1.48E-06	4.77E-06	6.06E-06	9.74E-06	1.10E-05	1.38E-05	1.13E-03	135
CAN FAT/TOTAL	0-48.3 km	1.0000	1.33E-06	1.03E-06	2.67E-06	3.41E-06	5.16E-06	5.69E-06	7.08E-06	1.13E-03	135
CAN FAT/TOTAL	0-64.4 km	1.0000	7.83E-07	6.26E-07	1.44E-06	1.87E-06	2.50E-06	2.79E-06	4.95E-06	8.56E-04	303
CAN FAT/TOTAL	0-80.5 km	1.0000	6.02E-07	5.06E-07	1.08E-06	1.24E-06	1.71E-06	1.96E-06	3.50E-06	8.56E-04	303
CAN FAT/TOTAL	0-161 km	1.0000	2.91E-07	2.31E-07	5.61E-07	7.10E-07	1.02E-06	1.15E-06	1.50E-06	1.14E-03	548
CAN FAT/TOTAL	0-322 km	1.0000	1.61E-07	1.08E-07	3.47E-07	4.64E-07	7.10E-07	8.86E-07	1.08E-06	1.14E-03	576
CAN FAT/TOTAL	0-805 km	1.0000	7.72E-08	5.45E-08	1.57E-07	2.16E-07	3.31E-07	3.75E-07	4.88E-07	1.14E-03	548
CAN FAT/TOTAL	0-1609 km	1.0000	4.39E-08	3.13E-08	9.46E-08	1.20E-07	1.96E-07	2.20E-07	2.75E-07	1.14E-03	548
CAN FAT/TOTAL	16.1-32.2 km	1.0000	2.40E-06	1.68E-06	5.18E-06	6.80E-06	1.02E-05	1.16E-05	1.53E-05	1.13E-03	135
CAN FAT/TOTAL	32.2-48.3 km	1.0000	9.15E-07	7.12E-07	1.86E-06	2.36E-06	3.41E-06	3.80E-06	4.80E-06	1.15E-03	681
CAN FAT/TOTAL	48.3-64.4 km	1.0000	4.16E-07	2.49E-07	1.02E-06	1.33E-06	2.28E-06	2.72E-06	5.66E-06	8.56E-04	303
CAN FAT/TOTAL	64.4-80.5 km	1.0000	2.87E-07	2.12E-07	6.15E-07	7.85E-07	1.20E-06	1.40E-06	1.94E-06	1.14E-03	548
CAN FAT/TOTAL	80.5-161 km	1.0000	1.65E-07	7.75E-08	3.98E-07	5.81E-07	1.04E-06	1.17E-06	1.52E-06	1.14E-03	548
CAN FAT/TOTAL	161-322 km	1.0000	7.70E-08	1.88E-08	2.51E-07	3.77E-07	7.23E-07	8.42E-07	1.10E-06	1.15E-03	711
CAN FAT/TOTAL	322-805 km	0.9920	1.05E-08	7.49E-09	2.47E-08	3.20E-08	4.88E-08	5.79E-08	7.87E-08	1.15E-03	544
CAN FAT/TOTAL	805-1609 km	0.9071	8.20E-10	4.46E-12	2.63E-09	5.41E-09	1.09E-08	1.21E-08	1.52E-08	1.14E-03	725

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.2-0.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.5-1.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.2-1.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.6-2.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	2.1-3.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	3.2-4.0 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.0-4.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.8-5.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	5.6-8.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	8.1-11.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	11.3-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	16.1-20.9 km	1.0000	1.55E-03	1.16E-03	3.10E-03	3.94E-03	5.84E-03	6.56E-03	7.49E-03	1.14E-03	305
L-ICRP60ED	20.9-25.8 km	1.0000	1.02E-03	8.14E-04	2.07E-03	2.68E-03	3.97E-03	4.61E-03	5.65E-03	1.13E-03	516
L-ICRP60ED	25.8-32.2 km	1.0000	6.82E-04	5.23E-04	1.34E-03	1.72E-03	2.47E-03	2.80E-03	4.29E-03	1.13E-03	517
L-ICRP60ED	32.2-40.2 km	1.0000	4.46E-04	3.32E-04	8.71E-04	1.07E-03	1.70E-03	2.05E-03	2.70E-03	1.13E-03	517
L-ICRP60ED	40.2-48.3 km	1.0000	3.03E-04	2.30E-04	6.09E-04	7.68E-04	1.10E-03	1.21E-03	1.48E-03	1.15E-03	129
L-ICRP60ED	48.3-64.4 km	1.0000	1.98E-04	1.44E-04	3.87E-04	5.04E-04	7.70E-04	9.12E-04	1.20E-03	1.14E-03	723
L-ICRP60ED	64.4-80.5 km	1.0000	1.27E-04	1.01E-04	2.46E-04	3.16E-04	5.02E-04	5.41E-04	6.34E-04	1.14E-03	463
L-ICRP60ED	80.5-113 km	1.0000	7.80E-05	6.20E-05	1.42E-04	1.89E-04	2.53E-04	2.82E-04	4.00E-04	1.12E-03	321
L-ICRP60ED	113-161 km	1.0000	4.76E-05	3.83E-05	8.59E-05	1.04E-04	1.38E-04	1.55E-04	2.28E-04	1.15E-03	585
L-ICRP60ED	161-241 km	1.0000	2.65E-05	2.18E-05	5.03E-05	5.94E-05	8.62E-05	1.00E-04	1.25E-04	3.04E-04	640
L-ICRP60ED	241-322 km	1.0000	1.61E-05	1.25E-05	3.04E-05	3.69E-05	6.29E-05	7.35E-05	1.16E-04	3.04E-04	640
L-ICRP60ED	322-563 km	1.0000	7.86E-06	6.66E-06	1.36E-05	1.70E-05	2.44E-05	2.77E-05	5.23E-05	3.04E-04	638
L-ICRP60ED	563-805 km	1.0000	3.69E-06	3.10E-06	7.12E-06	8.75E-06	1.52E-05	1.96E-05	2.61E-05	1.52E-04	507
L-ICRP60ED	805-1609 km	1.0000	7.61E-07	4.78E-07	1.90E-06	2.34E-06	3.35E-06	3.77E-06	7.42E-06	6.09E-04	633

DOSE FOUND AT ALL LOCATIONS (Sv)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
AREA (ha) THAT EXCEEDS THRESHOLD										
L-ICRP60ED Area exceeds 1.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD										
L-ICRP60ED Area exceeds 5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD										
A-THYROID Area exceeds 5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 low density no spray

"EARLY" DESCRIPTION = OCP3 low density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 low density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 4 = Group 4

HEALTH EFFECTS CASES

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-32.2 km	1.0000	4.12E+00	2.99E+00	8.75E+00	1.12E+01	1.68E+01	2.00E+01	2.54E+01	1.13E-03	135
CAN INJ/TOTAL	0-48.3 km	1.0000	7.65E+00	5.84E+00	1.43E+01	1.92E+01	3.02E+01	3.28E+01	3.90E+01	1.13E-03	135
CAN INJ/TOTAL	0-64.4 km	1.0000	1.12E+01	8.89E+00	2.15E+01	2.68E+01	3.75E+01	4.25E+01	6.94E+01	8.56E-04	303
CAN INJ/TOTAL	0-80.5 km	1.0000	1.36E+01	1.09E+01	2.50E+01	3.09E+01	4.55E+01	5.25E+01	7.71E+01	8.56E-04	303
CAN INJ/TOTAL	0-161 km	1.0000	2.27E+01	1.81E+01	4.27E+01	5.39E+01	7.55E+01	8.26E+01	1.11E+02	1.14E-03	548
CAN INJ/TOTAL	0-322 km	1.0000	3.17E+01	2.18E+01	6.86E+01	9.18E+01	1.40E+02	1.66E+02	2.11E+02	1.15E-03	711
CAN INJ/TOTAL	0-805 km	1.0000	3.42E+01	2.43E+01	7.45E+01	9.72E+01	1.50E+02	1.81E+02	2.14E+02	1.15E-03	711
CAN INJ/TOTAL	0-1609 km	1.0000	3.45E+01	2.47E+01	7.45E+01	9.72E+01	1.50E+02	1.81E+02	2.14E+02	1.15E-03	711
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	1.0000	1.88E+00	1.30E+00	3.89E+00	5.28E+00	7.59E+00	8.28E+00	1.16E+01	1.13E-03	135
CAN FAT/TOTAL	0-48.3 km	1.0000	3.49E+00	2.67E+00	7.18E+00	8.70E+00	1.22E+01	1.37E+01	1.79E+01	1.13E-03	135
CAN FAT/TOTAL	0-64.4 km	1.0000	5.13E+00	3.99E+00	9.39E+00	1.13E+01	1.64E+01	1.93E+01	3.17E+01	8.56E-04	303
CAN FAT/TOTAL	0-80.5 km	1.0000	6.22E+00	5.15E+00	1.12E+01	1.34E+01	2.01E+01	2.25E+01	3.53E+01	8.56E-04	303
CAN FAT/TOTAL	0-161 km	1.0000	1.04E+01	8.42E+00	1.99E+01	2.43E+01	3.47E+01	3.90E+01	5.09E+01	1.14E-03	548
CAN FAT/TOTAL	0-322 km	1.0000	1.45E+01	1.00E+01	3.15E+01	4.13E+01	6.84E+01	7.69E+01	9.65E+01	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	1.56E+01	1.12E+01	3.31E+01	4.32E+01	6.86E+01	7.73E+01	9.79E+01	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	1.58E+01	1.14E+01	3.32E+01	4.32E+01	6.86E+01	7.73E+01	9.79E+01	1.15E-03	711
CAN FAT/THYROID	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-80.5 km	1.0000	5.14E-02	4.11E-02	9.27E-02	1.09E-01	1.48E-01	1.68E-01	2.95E-01	8.56E-04	303
CAN FAT/THYROID	0-161 km	1.0000	8.51E-02	6.83E-02	1.56E-01	2.03E-01	2.92E-01	3.32E-01	4.30E-01	1.14E-03	548
CAN FAT/THYROID	0-1609 km	1.0000	1.27E-01	9.12E-02	2.58E-01	3.37E-01	5.41E-01	6.39E-01	7.81E-01	1.15E-03	711
CAN FAT/BREAST	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-80.5 km	1.0000	4.19E-01	3.40E-01	7.71E-01	9.48E-01	1.29E+00	1.46E+00	2.34E+00	8.56E-04	303
CAN FAT/BREAST	0-161 km	1.0000	6.98E-01	5.63E-01	1.26E+00	1.56E+00	2.36E+00	2.73E+00	3.70E+00	1.14E-03	118
CAN FAT/BREAST	0-1609 km	1.0000	1.05E+00	7.74E-01	2.25E+00	2.94E+00	4.38E+00	5.13E+00	6.60E+00	1.15E-03	711
CAN FAT/LUNG	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-80.5 km	1.0000	1.20E+00	1.00E+00	2.21E+00	2.67E+00	3.73E+00	4.23E+00	6.78E+00	8.56E-04	303
CAN FAT/LUNG	0-161 km	1.0000	2.00E+00	1.54E+00	3.71E+00	4.68E+00	6.64E+00	7.56E+00	9.86E+00	1.14E-03	548
CAN FAT/LUNG	0-1609 km	1.0000	3.05E+00	2.21E+00	6.34E+00	8.35E+00	1.24E+01	1.42E+01	1.90E+01	1.15E-03	711
CAN FAT/LEUKEMIA	0-1609 km	1.0000	2.39E+00	1.65E+00	5.12E+00	6.73E+00	1.01E+01	1.14E+01	1.47E+01	1.15E-03	711
CAN FAT/BONE	0-1609 km	1.0000	6.98E-02	5.00E-02	1.39E-01	1.93E-01	3.02E-01	3.38E-01	4.30E-01	1.14E-03	548
CAN FAT/LIVER	0-1609 km	1.0000	3.42E-01	2.46E-01	7.42E-01	9.71E-01	1.42E+00	1.67E+00	2.14E+00	1.15E-03	711
CAN FAT/COLON	0-1609 km	1.0000	3.04E+00	2.19E+00	6.30E+00	8.33E+00	1.24E+01	1.42E+01	1.89E+01	1.15E-03	711
CAN FAT/RESIDUAL	0-1609 km	1.0000	5.67E+00	4.00E+00	1.15E+01	1.46E+01	2.35E+01	2.75E+01	3.54E+01	1.15E-03	711

EARLY FATALITY DISTANCE (km)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL RISK > 0.000	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION EXCEEDING DOSE

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
EARLY dose A-RED MARR > 2.32 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-LUNGS > 13.6 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-STOMACH > 6.50 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION DOSE (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED TOT LIF	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED TOT LIF	0-80.5 km	1.0000	1.14E+02	9.37E+01	2.11E+02	2.53E+02	3.63E+02	4.17E+02	6.47E+02	8.56E-04	303
L-ICRP60ED TOT LIF	0-161 km	1.0000	1.90E+02	1.44E+02	3.53E+02	4.39E+02	6.59E+02	7.46E+02	9.29E+02	1.14E-03	548
L-ICRP60ED TOT LIF	0-1609 km	1.0000	2.88E+02	2.09E+02	5.99E+02	7.70E+02	1.15E+03	1.33E+03	1.79E+03	1.15E-03	711

POPULATION WEIGHTED RISK

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	1.0000	3.93E-06	2.84E-06	8.03E-06	1.05E-05	1.42E-05	1.62E-05	2.43E-05	1.13E-03	135
CAN FAT/TOTAL	0-48.3 km	1.0000	2.42E-06	1.86E-06	4.90E-06	6.01E-06	9.20E-06	1.04E-05	1.24E-05	1.13E-03	135
CAN FAT/TOTAL	0-64.4 km	1.0000	1.42E-06	1.11E-06	2.63E-06	3.28E-06	5.08E-06	5.56E-06	8.82E-06	8.56E-04	303
CAN FAT/TOTAL	0-80.5 km	1.0000	1.10E-06	8.96E-07	2.04E-06	2.44E-06	3.56E-06	4.12E-06	6.23E-06	8.56E-04	303
CAN FAT/TOTAL	0-161 km	1.0000	5.29E-07	4.19E-07	1.01E-06	1.20E-06	1.80E-06	2.08E-06	2.60E-06	1.14E-03	548
CAN FAT/TOTAL	0-322 km	1.0000	2.89E-07	2.01E-07	6.09E-07	8.12E-07	1.24E-06	1.43E-06	1.93E-06	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	1.39E-07	1.00E-07	3.03E-07	3.77E-07	6.05E-07	7.14E-07	8.68E-07	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	7.88E-08	5.68E-08	1.58E-07	2.14E-07	3.24E-07	3.70E-07	4.90E-07	1.15E-03	711
CAN FAT/TOTAL	16.1-32.2 km	1.0000	4.34E-06	3.11E-06	9.58E-06	1.19E-05	1.84E-05	2.12E-05	2.68E-05	1.13E-03	135
CAN FAT/TOTAL	32.2-48.3 km	1.0000	1.67E-06	1.25E-06	3.33E-06	4.20E-06	7.06E-06	7.43E-06	8.30E-06	1.15E-03	681
CAN FAT/TOTAL	48.3-64.4 km	1.0000	7.58E-07	4.61E-07	1.73E-06	2.39E-06	4.54E-06	5.59E-06	1.00E-05	8.56E-04	303
CAN FAT/TOTAL	64.4-80.5 km	1.0000	5.25E-07	3.78E-07	1.14E-06	1.42E-06	2.23E-06	2.56E-06	3.41E-06	1.14E-03	548
CAN FAT/TOTAL	80.5-161 km	1.0000	2.98E-07	1.40E-07	7.57E-07	1.04E-06	1.93E-06	2.20E-06	2.79E-06	1.14E-03	118
CAN FAT/TOTAL	161-322 km	1.0000	1.35E-07	3.38E-08	4.53E-07	6.92E-07	1.15E-06	1.37E-06	2.03E-06	1.15E-03	711
CAN FAT/TOTAL	322-805 km	0.9920	1.83E-08	1.21E-08	4.35E-08	5.64E-08	9.30E-08	1.07E-07	1.30E-07	1.15E-03	544
CAN FAT/TOTAL	805-1609 km	0.9071	1.53E-09	8.11E-12	4.92E-09	9.44E-09	2.11E-08	2.31E-08	2.81E-08	1.14E-03	825

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.2-0.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.5-1.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.2-1.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.6-2.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	2.1-3.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	3.2-4.0 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.0-4.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.8-5.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	5.6-8.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	8.1-11.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	11.3-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	16.1-20.9 km	1.0000	2.84E-03	2.33E-03	5.83E-03	7.13E-03	8.18E-03	8.67E-03	9.85E-03	1.11E-03	312
L-ICRP60ED	20.9-25.8 km	1.0000	1.91E-03	1.42E-03	3.78E-03	4.84E-03	7.16E-03	7.95E-03	9.95E-03	1.13E-03	516

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 L-ICRP60ED             25.8-32.2 km  1.0000  1.27E-03  1.00E-03  2.57E-03  3.22E-03  4.73E-03  5.42E-03  7.50E-03  1.13E-03  517
 L-ICRP60ED             32.2-40.2 km  1.0000  8.37E-04  6.34E-04  1.58E-03  2.04E-03  2.94E-03  3.48E-03  5.08E-03  1.15E-03  324
 L-ICRP60ED             40.2-48.3 km  1.0000  5.72E-04  4.38E-04  1.12E-03  1.38E-03  2.09E-03  2.27E-03  2.71E-03  1.15E-03  129
 L-ICRP60ED             48.3-64.4 km  1.0000  3.75E-04  3.01E-04  7.15E-04  9.38E-04  1.32E-03  1.51E-03  2.14E-03  1.14E-03  723
 L-ICRP60ED             64.4-80.5 km  1.0000  2.43E-04  1.91E-04  4.62E-04  6.02E-04  8.96E-04  1.05E-03  1.61E-03  1.14E-03  463
 L-ICRP60ED              80.5-113 km  1.0000  1.49E-04  1.13E-04  2.82E-04  3.55E-04  5.44E-04  6.12E-04  9.75E-04  1.12E-03  321
 L-ICRP60ED               113-161 km  1.0000  8.99E-05  7.30E-05  1.64E-04  2.09E-04  2.97E-04  3.36E-04  4.36E-04  1.15E-03  585
 L-ICRP60ED               161-241 km  1.0000  4.94E-05  3.91E-05  9.37E-05  1.11E-04  1.53E-04  1.75E-04  2.27E-04  3.04E-04  640
 L-ICRP60ED               241-322 km  1.0000  2.92E-05  2.34E-05  5.51E-05  6.91E-05  1.07E-04  1.21E-04  2.02E-04  3.04E-04  640
 L-ICRP60ED               322-563 km  1.0000  1.41E-05  1.16E-05  2.59E-05  3.17E-05  4.41E-05  5.07E-05  8.99E-05  3.04E-04  638
 L-ICRP60ED               563-805 km  1.0000  7.05E-06  5.77E-06  1.31E-05  1.64E-05  2.69E-05  3.11E-05  4.47E-05  1.52E-04  507
 L-ICRP60ED              805-1609 km  1.0000  1.45E-06  9.11E-07  3.58E-06  4.44E-06  5.99E-06  6.64E-06  1.33E-05  3.04E-04  632

                              PROB                QUANTILES                 PEAK     PEAK  PEAK
                            NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   1.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 AREA (ha) THAT EXCEEDS THRESHOLD
 A-THYROID  Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

 **** Indicates that the value is outside resolution of the analysis.
 Optionally increase number of trials for better resolution.

  "ATMOS"  DESCRIPTION = OCP3 low density no spray
  "EARLY"  DESCRIPTION = OCP3 low density no spray, EARLY input

SOURCE TERM  1 OF  1:
   OCP3 low density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

   COHORT 5 = Group 5

                              PROB                QUANTILES                 PEAK     PEAK  PEAK
                            NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
HEALTH EFFECTS CASES
 ERL FAT/TOTAL              0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL              0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL              0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-32.2 km  1.0000  4.12E+00  2.99E+00  8.75E+00  1.12E+01  1.68E+01  2.00E+01  2.54E+01  1.13E-03  135
 CAN INJ/TOTAL              0-48.3 km  1.0000  7.65E+00  5.84E+00  1.43E+01  1.92E+01  3.02E+01  3.28E+01  3.90E+01  1.13E-03  135
 CAN INJ/TOTAL              0-64.4 km  1.0000  1.12E+01  8.89E+00  2.15E+01  2.68E+01  3.75E+01  4.25E+01  6.94E+01  8.56E-04  303
 CAN INJ/TOTAL              0-80.5 km  1.0000  1.36E+01  1.09E+01  2.50E+01  3.09E+01  4.55E+01  5.25E+01  7.71E+01  8.56E-04  303
 CAN INJ/TOTAL               0-161 km  1.0000  2.27E+01  1.81E+01  4.27E+01  5.39E+01  7.55E+01  8.26E+01  1.11E+02  1.14E-03  548
 CAN INJ/TOTAL               0-322 km  1.0000  3.17E+01  2.18E+01  6.86E+01  9.18E+01  1.40E+02  1.66E+02  2.11E+02  1.15E-03  711
 CAN INJ/TOTAL               0-805 km  1.0000  3.42E+01  2.43E+01  7.45E+01  9.72E+01  1.50E+02  1.81E+02  2.14E+02  1.15E-03  711
 CAN INJ/TOTAL              0-1609 km  1.0000  3.45E+01  2.47E+01  7.45E+01  9.72E+01  1.50E+02  1.81E+02  2.14E+02  1.15E-03  711
 CAN FAT/TOTAL              0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-32.2 km  1.0000  1.88E+00  1.30E+00  3.89E+00  5.28E+00  7.59E+00  8.28E+00  1.16E+01  1.13E-03  135
 CAN FAT/TOTAL              0-48.3 km  1.0000  3.49E+00  2.67E+00  7.18E+00  8.70E+00  1.22E+01  1.37E+01  1.79E+01  1.13E-03  135
 CAN FAT/TOTAL              0-64.4 km  1.0000  5.13E+00  3.99E+00  9.39E+00  1.13E+01  1.64E+01  1.93E+01  3.17E+01  8.56E-04  303
 CAN FAT/TOTAL              0-80.5 km  1.0000  6.22E+00  5.15E+00  1.12E+01  1.34E+01  2.01E+01  2.25E+01  3.53E+01  8.56E-04  303
 CAN FAT/TOTAL               0-161 km  1.0000  1.04E+01  8.42E+00  1.99E+01  2.43E+01  3.47E+01  3.90E+01  5.09E+01  1.14E-03  548
 CAN FAT/TOTAL               0-322 km  1.0000  1.45E+01  1.00E+01  3.15E+01  4.13E+01  6.84E+01  7.69E+01  9.65E+01  1.15E-03  711
 CAN FAT/TOTAL               0-805 km  1.0000  1.56E+01  1.12E+01  3.31E+01  4.32E+01  6.86E+01  7.73E+01  9.79E+01  1.15E-03  711
 CAN FAT/TOTAL              0-1609 km  1.0000  1.58E+01  1.14E+01  3.32E+01  4.32E+01  6.86E+01  7.73E+01  9.79E+01  1.15E-03  711
 CAN FAT/THYROID            0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID            0-80.5 km  1.0000  5.14E-02  4.11E-02  9.27E-02  1.09E-01  1.48E-01  1.68E-01  2.95E-01  8.56E-04  303
 CAN FAT/THYROID             0-161 km  1.0000  8.51E-02  6.83E-02  1.56E-01  2.03E-01  2.92E-01  3.32E-01  4.30E-01  1.14E-03  548
 CAN FAT/THYROID            0-1609 km  1.0000  1.27E-01  9.12E-02  2.58E-01  3.37E-01  5.41E-01  6.39E-01  7.81E-01  1.15E-03  711
 CAN FAT/BREAST             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST             0-80.5 km  1.0000  4.19E-01  3.40E-01  7.71E-01  9.48E-01  1.29E+00  1.46E+00  2.34E+00  8.56E-04  303
 CAN FAT/BREAST              0-161 km  1.0000  6.98E-01  5.63E-01  1.26E+00  1.56E+00  2.36E+00  2.73E+00  3.70E+00  1.14E-03  118
 CAN FAT/BREAST             0-1609 km  1.0000  1.05E+00  7.74E-01  2.25E+00  2.94E+00  4.38E+00  5.13E+00  6.60E+00  1.15E-03  711
 CAN FAT/LUNG               0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG               0-80.5 km  1.0000  1.20E+00  1.00E+00  2.21E+00  2.67E+00  3.73E+00  4.23E+00  6.78E+00  8.56E-04  303
 CAN FAT/LUNG                0-161 km  1.0000  2.00E+00  1.54E+00  3.71E+00  4.68E+00  6.64E+00  7.56E+00  9.86E+00  1.14E-03  548
 CAN FAT/LUNG               0-1609 km  1.0000  3.05E+00  2.21E+00  6.34E+00  8.35E+00  1.24E+01  1.42E+01  1.90E+01  1.15E-03  711
 CAN FAT/LEUKEMIA           0-1609 km  1.0000  2.39E+00  1.65E+00  5.12E+00  6.73E+00  1.01E+01  1.14E+01  1.47E+01  1.15E-03  711
 CAN FAT/BONE               0-1609 km  1.0000  6.98E-02  5.00E-02  1.39E-01  1.93E-01  3.02E-01  3.38E-01  4.30E-01  1.14E-03  548
 CAN FAT/LIVER              0-1609 km  1.0000  3.42E-01  2.46E-01  7.42E-01  9.71E-01  1.42E+00  1.67E+00  2.14E+00  1.15E-03  711
 CAN FAT/COLON              0-1609 km  1.0000  3.04E+00  2.19E+00  6.30E+00  8.33E+00  1.24E+01  1.42E+01  1.89E+01  1.15E-03  711
 CAN FAT/RESIDUAL           0-1609 km  1.0000  5.67E+00  4.00E+00  1.15E+01  1.46E+01  2.35E+01  2.75E+01  3.54E+01  1.15E-03  711

                              PROB                QUANTILES                 PEAK     PEAK  PEAK
                            NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
EARLY FATALITY DISTANCE (km)
 ERL FAT/TOTAL RISK > 0.000            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                              PROB                QUANTILES                 PEAK     PEAK  PEAK
                            NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
POPULATION EXCEEDING DOSE
 EARLY dose A-RED MARR > 2.32 Sv       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-LUNGS > 13.6 Sv          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-STOMACH > 6.50 Sv        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                              PROB                QUANTILES                 PEAK     PEAK  PEAK
                            NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
POPULATION DOSE (Sv)
 L-ICRP60ED  TOT LIF        0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF        0-80.5 km  1.0000  1.14E+02  9.37E+01  2.11E+02  2.53E+02  3.63E+02  4.17E+02  6.47E+02  8.56E-04  303
 L-ICRP60ED  TOT LIF         0-161 km  1.0000  1.90E+02  1.44E+02  3.53E+02  4.39E+02  6.59E+02  7.46E+02  9.29E+02  1.14E-03  548
 L-ICRP60ED  TOT LIF        0-1609 km  1.0000  2.88E+02  2.09E+02  5.99E+02  7.70E+02  1.15E+03  1.33E+03  1.79E+03  1.15E-03  711

                              PROB                QUANTILES                 PEAK     PEAK  PEAK
                            NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
POPULATION WEIGHTED RISK
 CAN FAT/TOTAL              0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-32.2 km  1.0000  3.93E-06  2.84E-06  8.03E-06  1.05E-05  1.42E-05  1.62E-05  2.43E-05  1.13E-03  135
 CAN FAT/TOTAL              0-48.3 km  1.0000  2.42E-06  1.86E-06  4.90E-06  6.01E-06  9.20E-06  1.04E-05  1.24E-05  1.13E-03  135
 CAN FAT/TOTAL              0-64.4 km  1.0000  1.42E-06  1.11E-06  2.63E-06  3.28E-06  5.08E-06  5.56E-06  8.82E-06  8.56E-04  303
 CAN FAT/TOTAL              0-80.5 km  1.0000  1.10E-06  8.96E-07  2.04E-06  2.44E-06  3.56E-06  4.12E-06  6.23E-06  8.56E-04  303
 CAN FAT/TOTAL               0-161 km  1.0000  5.29E-07  4.19E-07  1.01E-06  1.20E-06  1.80E-06  2.08E-06  2.60E-06  1.14E-03  548
 CAN FAT/TOTAL               0-322 km  1.0000  2.89E-07  2.01E-07  6.09E-07  8.12E-07  1.24E-06  1.43E-06  1.93E-06  1.15E-03  711
 CAN FAT/TOTAL               0-805 km  1.0000  1.39E-07  1.00E-07  3.03E-07  3.77E-07  6.05E-07  7.14E-07  8.68E-07  1.15E-03  711
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CAN FAT/TOTAL	0-1609 km	1.0000	7.88E-08	5.68E-08	1.58E-07	2.14E-07	3.24E-07	3.70E-07	4.90E-07	1.15E-03	711
CAN FAT/TOTAL	16.1-32.2 km	1.0000	4.34E-06	3.11E-06	9.58E-06	1.19E-05	1.84E-05	2.12E-05	2.68E-05	1.13E-03	135
CAN FAT/TOTAL	32.2-48.3 km	1.0000	1.67E-06	1.25E-06	3.33E-06	4.20E-06	7.06E-06	7.43E-06	8.30E-06	1.15E-03	681
CAN FAT/TOTAL	48.3-64.4 km	1.0000	7.58E-07	4.61E-07	1.73E-06	2.39E-06	4.54E-06	5.59E-06	1.00E-05	8.56E-04	303
CAN FAT/TOTAL	64.4-80.5 km	1.0000	5.25E-07	3.78E-07	1.14E-06	1.42E-06	2.23E-06	2.56E-06	3.41E-06	1.14E-03	548
CAN FAT/TOTAL	80.5-161 km	1.0000	2.98E-07	1.40E-07	7.57E-07	1.04E-06	1.93E-06	2.20E-06	2.79E-06	1.14E-03	118
CAN FAT/TOTAL	161-322 km	1.0000	1.35E-07	3.38E-08	4.53E-07	6.92E-07	1.15E-06	1.37E-06	2.03E-06	1.15E-03	711
CAN FAT/TOTAL	322-805 km	0.9920	1.83E-08	1.21E-08	4.35E-08	5.64E-08	9.30E-08	1.07E-07	1.30E-07	1.15E-03	544
CAN FAT/TOTAL	805-1609 km	0.9071	1.53E-09	8.11E-12	4.92E-09	9.44E-09	2.11E-08	2.31E-08	2.81E-08	1.14E-03	825

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)											
L-ICRP60ED	0-0.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.2-0.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.5-1.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.2-1.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.6-2.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	2.1-3.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	3.2-4.0 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.0-4.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.8-5.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	5.6-8.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	8.1-11.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	11.3-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	16.1-20.9 km	1.0000	2.84E-03	2.33E-03	5.83E-03	7.13E-03	8.18E-03	8.67E-03	9.85E-03	1.11E-03	312
L-ICRP60ED	20.9-25.8 km	1.0000	1.91E-03	1.42E-03	3.78E-03	4.84E-03	7.16E-03	7.95E-03	9.95E-03	1.13E-03	516
L-ICRP60ED	25.8-32.2 km	1.0000	1.27E-03	1.00E-03	2.57E-03	3.22E-03	4.73E-03	5.42E-03	7.50E-03	1.13E-03	517
L-ICRP60ED	32.2-40.2 km	1.0000	8.37E-04	6.34E-04	1.58E-03	2.04E-03	2.94E-03	3.48E-03	5.08E-03	1.15E-03	324
L-ICRP60ED	40.2-48.3 km	1.0000	5.72E-04	4.38E-04	1.12E-03	1.38E-03	2.09E-03	2.27E-03	2.71E-03	1.15E-03	129
L-ICRP60ED	48.3-64.4 km	1.0000	3.75E-04	3.01E-04	7.15E-04	9.38E-04	1.32E-03	1.51E-03	2.14E-03	1.14E-03	723
L-ICRP60ED	64.4-80.5 km	1.0000	2.43E-04	1.91E-04	4.62E-04	6.02E-04	8.96E-04	1.05E-03	1.61E-03	1.14E-03	463
L-ICRP60ED	80.5-113 km	1.0000	1.49E-04	1.13E-04	2.82E-04	3.55E-04	5.44E-04	6.12E-04	9.75E-04	1.12E-03	321
L-ICRP60ED	113-161 km	1.0000	8.99E-05	7.30E-05	1.64E-04	2.09E-04	2.97E-04	3.36E-04	4.36E-04	1.15E-03	585
L-ICRP60ED	161-241 km	1.0000	4.94E-05	3.91E-05	9.37E-05	1.11E-04	1.53E-04	1.75E-04	2.27E-04	3.04E-04	640
L-ICRP60ED	241-322 km	1.0000	2.92E-05	2.34E-05	5.51E-05	6.91E-05	1.07E-04	1.21E-04	2.02E-04	3.04E-04	640
L-ICRP60ED	322-563 km	1.0000	1.41E-05	1.16E-05	2.59E-05	3.17E-05	4.41E-05	5.07E-05	8.99E-05	3.04E-04	638
L-ICRP60ED	563-805 km	1.0000	7.05E-06	5.77E-06	1.31E-05	1.64E-05	2.69E-05	3.11E-05	4.47E-05	1.52E-04	507
L-ICRP60ED	805-1609 km	1.0000	1.45E-06	9.11E-07	3.58E-06	4.44E-06	5.99E-06	6.64E-06	1.33E-05	3.04E-04	632

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)											
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	1.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD											
A-THYROID Area exceeds	5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 low density no spray
"EARLY" DESCRIPTION = OCP3 low density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 low density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 6 = Group 6

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
HEALTH EFFECTS CASES											
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	1.0000	2.08E+00	1.83E+00	3.66E+00	4.44E+00	5.96E+00	6.61E+00	7.93E+00	1.14E-03	463
CAN INJ/TOTAL	0-32.2 km	1.0000	6.20E+00	5.21E+00	1.11E+01	1.34E+01	2.02E+01	2.24E+01	2.80E+01	1.13E-03	135
CAN INJ/TOTAL	0-48.3 km	1.0000	9.73E+00	8.13E+00	1.79E+01	2.21E+01	3.11E+01	3.41E+01	4.16E+01	1.13E-03	135
CAN INJ/TOTAL	0-64.4 km	1.0000	1.33E+01	1.09E+01	2.36E+01	2.95E+01	3.86E+01	4.31E+01	7.08E+01	8.56E-04	303
CAN INJ/TOTAL	0-80.5 km	1.0000	1.57E+01	1.26E+01	2.66E+01	3.20E+01	4.58E+01	5.25E+01	7.86E+01	8.56E-04	303
CAN INJ/TOTAL	0-161 km	1.0000	2.48E+01	2.08E+01	4.51E+01	5.57E+01	7.86E+01	8.80E+01	1.12E+02	1.14E-03	548
CAN INJ/TOTAL	0-322 km	1.0000	3.37E+01	2.43E+01	7.18E+01	9.27E+01	1.40E+02	1.66E+02	2.13E+02	1.15E-03	711
CAN INJ/TOTAL	0-805 km	1.0000	3.62E+01	2.67E+01	7.54E+01	9.89E+01	1.51E+02	1.81E+02	2.16E+02	1.15E-03	711
CAN INJ/TOTAL	0-1609 km	1.0000	3.65E+01	2.71E+01	7.55E+01	9.89E+01	1.51E+02	1.81E+02	2.16E+02	1.15E-03	711
CAN FAT/TOTAL	0-16.1 km	1.0000	9.58E-01	8.47E-01	1.74E+00	2.14E+00	2.95E+00	3.19E+00	3.70E+00	1.14E-03	463
CAN FAT/TOTAL	0-32.2 km	1.0000	2.84E+00	2.33E+00	5.34E+00	6.59E+00	9.38E+00	1.05E+01	1.28E+01	1.13E-03	135
CAN FAT/TOTAL	0-48.3 km	1.0000	4.45E+00	3.66E+00	8.11E+00	1.02E+01	1.33E+01	1.49E+01	1.91E+01	1.13E-03	135
CAN FAT/TOTAL	0-64.4 km	1.0000	6.09E+00	5.24E+00	1.08E+01	1.28E+01	1.89E+01	2.17E+01	3.24E+01	8.56E-04	303
CAN FAT/TOTAL	0-80.5 km	1.0000	7.17E+00	6.16E+00	1.22E+01	1.46E+01	2.11E+01	2.33E+01	3.60E+01	8.56E-04	303
CAN FAT/TOTAL	0-161 km	1.0000	1.13E+01	9.49E+00	2.09E+01	2.53E+01	3.52E+01	3.94E+01	5.15E+01	1.14E-03	548
CAN FAT/TOTAL	0-322 km	1.0000	1.54E+01	1.11E+01	3.23E+01	4.21E+01	6.85E+01	7.71E+01	9.72E+01	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	1.66E+01	1.21E+01	3.40E+01	4.50E+01	6.89E+01	7.75E+01	9.86E+01	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	1.67E+01	1.22E+01	3.41E+01	4.50E+01	6.89E+01	7.75E+01	9.86E+01	1.15E-03	711
CAN FAT/THYROID	0-16.1 km	1.0000	8.48E-03	7.44E-03	1.52E-02	1.95E-02	2.85E-02	3.13E-02	3.58E-02	1.14E-03	463
CAN FAT/THYROID	0-80.5 km	1.0000	5.98E-02	5.21E-02	1.04E-01	1.19E-01	1.61E-01	1.83E-01	3.01E-01	8.56E-04	303
CAN FAT/THYROID	0-161 km	1.0000	9.35E-02	7.84E-02	1.68E-01	2.13E-01	3.05E-01	3.42E-01	4.35E-01	1.14E-03	548
CAN FAT/THYROID	0-1609 km	1.0000	1.36E-01	1.03E-01	2.71E-01	3.49E-01	5.55E-01	6.45E-01	7.88E-01	1.15E-03	711
CAN FAT/BREAST	0-16.1 km	1.0000	5.90E-02	5.38E-02	1.02E-01	1.12E-01	1.40E-01	1.54E-01	1.88E-01	1.14E-03	463
CAN FAT/BREAST	0-80.5 km	1.0000	4.78E-01	4.03E-01	8.38E-01	1.01E+00	1.42E+00	1.65E+00	2.39E+00	8.56E-04	303
CAN FAT/BREAST	0-161 km	1.0000	7.57E-01	6.41E-01	1.32E+00	1.66E+00	2.42E+00	2.75E+00	3.77E+00	1.14E-03	118
CAN FAT/BREAST	0-1609 km	1.0000	1.11E+00	8.29E-01	2.27E+00	2.99E+00	4.40E+00	5.13E+00	6.64E+00	1.15E-03	711
CAN FAT/LUNG	0-16.1 km	1.0000	1.86E-01	1.51E-01	3.40E-01	4.13E-01	5.53E-01	5.96E-01	7.27E-01	1.14E-03	463
CAN FAT/LUNG	0-80.5 km	1.0000	1.38E+00	1.15E+00	2.41E+00	2.91E+00	3.84E+00	4.30E+00	6.91E+00	8.56E-04	303
CAN FAT/LUNG	0-161 km	1.0000	2.19E+00	1.78E+00	3.94E+00	5.02E+00	7.00E+00	7.84E+00	9.99E+00	1.14E-03	548
CAN FAT/LUNG	0-1609 km	1.0000	3.24E+00	2.41E+00	6.63E+00	8.69E+00	1.26E+01	1.44E+01	1.91E+01	1.15E-03	711
CAN FAT/LEUKEMIA	0-1609 km	1.0000	2.55E+00	1.88E+00	5.24E+00	6.80E+00	1.01E+01	1.14E+01	1.48E+01	1.15E-03	711
CAN FAT/BONE	0-1609 km	1.0000	7.46E-02	5.57E-02	1.44E-01	1.97E-01	3.02E-01	3.39E-01	4.33E-01	1.14E-03	548
CAN FAT/LIVER	0-1609 km	1.0000	3.62E-01	2.70E-01	7.49E-01	9.80E-01	1.43E+00	1.67E+00	2.15E+00	1.15E-03	711
CAN FAT/COLON	0-1609 km	1.0000	3.23E+00	2.40E+00	6.56E+00	8.72E+00	1.27E+01	1.44E+01	1.91E+01	1.15E-03	711
CAN FAT/RESIDUAL	0-1609 km	1.0000	6.01E+00	4.42E+00	1.16E+01	1.47E+01	2.35E+01	2.75E+01	3.56E+01	1.15E-03	711

		PROB		QUANTILES					PEAK	PEAK	PEAK
		NON-ZERO	MEAN	50TH	90TH	95TH	99TH	99.5TH	CONSEQ	PROB	TRIAL
EARLY FATALITY DISTANCE (km)											
ERL FAT/TOTAL RISK > 0.000		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION EXCEEDING DOSE

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
EARLY dose A-RED MARR	> 2.32 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-LUNGS	> 13.6 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-STOMACH	> 6.50 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION DOSE (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED TOT LIF	0-16.1 km	1.0000	1.75E+01	1.40E+01	3.11E+01	3.69E+01	5.22E+01	5.65E+01	6.66E+01	1.14E-03	463
L-ICRP60ED TOT LIF	0-80.5 km	1.0000	1.32E+02	1.10E+02	2.31E+02	2.78E+02	3.78E+02	4.27E+02	6.59E+02	8.56E-04	303
L-ICRP60ED TOT LIF	0-161 km	1.0000	2.07E+02	1.65E+02	3.69E+02	4.58E+02	6.62E+02	7.49E+02	9.41E+02	1.14E-03	548
L-ICRP60ED TOT LIF	0-1609 km	1.0000	3.05E+02	2.29E+02	6.11E+02	7.83E+02	1.18E+03	1.35E+03	1.80E+03	1.15E-03	711

POPULATION WEIGHTED RISK

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-16.1 km	1.0000	2.10E-05	1.85E-05	3.72E-05	4.55E-05	6.48E-05	7.18E-05	8.12E-05	1.14E-03	463
CAN FAT/TOTAL	0-32.2 km	1.0000	5.93E-06	5.09E-06	1.10E-05	1.32E-05	2.02E-05	2.21E-05	2.68E-05	1.13E-03	135
CAN FAT/TOTAL	0-48.3 km	1.0000	3.08E-06	2.55E-06	5.62E-06	7.12E-06	1.02E-05	1.11E-05	1.32E-05	1.13E-03	135
CAN FAT/TOTAL	0-64.4 km	1.0000	1.69E-06	1.34E-06	3.05E-06	3.70E-06	5.31E-06	5.74E-06	9.00E-06	8.56E-04	303
CAN FAT/TOTAL	0-80.5 km	1.0000	1.27E-06	1.07E-06	2.22E-06	2.63E-06	3.69E-06	4.20E-06	6.34E-06	8.56E-04	303
CAN FAT/TOTAL	0-161 km	1.0000	5.77E-07	4.92E-07	1.06E-06	1.26E-06	1.87E-06	2.13E-06	2.63E-06	1.14E-03	548
CAN FAT/TOTAL	0-322 km	1.0000	3.08E-07	2.23E-07	6.30E-07	8.39E-07	1.26E-06	1.45E-06	1.94E-06	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	1.47E-07	1.09E-07	3.10E-07	3.82E-07	6.25E-07	7.29E-07	8.75E-07	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	8.36E-08	6.26E-08	1.62E-07	2.17E-07	3.28E-07	3.74E-07	4.93E-07	1.15E-03	711
CAN FAT/TOTAL	16.1-32.2 km	1.0000	4.34E-06	3.11E-06	9.58E-06	1.19E-05	1.84E-05	2.12E-05	2.68E-05	1.13E-03	135
CAN FAT/TOTAL	32.2-48.3 km	1.0000	1.67E-06	1.25E-06	3.33E-06	4.20E-06	7.06E-06	7.43E-06	8.30E-06	1.15E-03	681
CAN FAT/TOTAL	48.3-64.4 km	1.0000	7.58E-07	4.61E-07	1.73E-06	2.39E-06	4.54E-06	5.59E-06	1.00E-05	8.56E-04	303
CAN FAT/TOTAL	64.4-80.5 km	1.0000	5.25E-07	3.78E-07	1.14E-06	1.42E-06	2.23E-06	2.56E-06	3.41E-06	1.14E-03	548
CAN FAT/TOTAL	80.5-161 km	1.0000	2.98E-07	1.40E-07	7.57E-07	1.04E-06	1.93E-06	2.20E-06	2.79E-06	1.14E-03	118
CAN FAT/TOTAL	161-322 km	1.0000	1.35E-07	3.38E-08	4.53E-07	6.92E-07	1.15E-06	1.37E-06	2.03E-06	1.15E-03	711
CAN FAT/TOTAL	322-805 km	0.9920	1.83E-08	1.21E-08	4.35E-08	5.64E-08	9.30E-08	1.07E-07	1.30E-07	1.15E-03	544
CAN FAT/TOTAL	805-1609 km	0.9071	1.53E-09	8.11E-12	4.92E-09	9.44E-09	2.11E-08	2.31E-08	2.81E-08	1.14E-03	825

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	1.0000	2.68E-01	2.34E-01	3.97E-01	4.82E-01	6.38E-01	7.05E-01	7.72E-01	1.14E-03	390
L-ICRP60ED	0.2-0.5 km	1.0000	1.09E-01	1.03E-01	1.43E-01	1.65E-01	2.13E-01	2.28E-01	2.62E-01	1.15E-03	266
L-ICRP60ED	0.5-1.2 km	1.0000	5.98E-02	5.59E-02	9.83E-02	1.07E-01	1.26E-01	1.35E-01	1.57E-01	1.13E-03	524
L-ICRP60ED	1.2-1.6 km	1.0000	4.29E-02	3.66E-02	7.62E-02	9.46E-02	1.11E-01	1.17E-01	1.31E-01	1.13E-03	592
L-ICRP60ED	1.6-2.1 km	1.0000	3.56E-02	3.06E-02	6.72E-02	8.94E-02	1.09E-01	1.14E-01	1.26E-01	1.13E-03	439
L-ICRP60ED	2.1-3.2 km	1.0000	2.70E-02	2.22E-02	5.15E-02	7.04E-02	8.45E-02	9.14E-02	1.04E-01	1.13E-03	374
L-ICRP60ED	3.2-4.0 km	1.0000	2.01E-02	1.64E-02	3.79E-02	5.21E-02	7.34E-02	7.68E-02	8.48E-02	1.13E-03	374
L-ICRP60ED	4.0-4.8 km	1.0000	1.62E-02	1.18E-02	2.85E-02	3.88E-02	5.58E-02	5.99E-02	6.96E-02	1.13E-03	374
L-ICRP60ED	4.8-5.6 km	1.0000	1.34E-02	1.04E-02	2.40E-02	2.97E-02	5.02E-02	5.22E-02	5.68E-02	1.13E-03	374
L-ICRP60ED	5.6-8.1 km	1.0000	9.70E-03	8.28E-03	1.67E-02	2.20E-02	3.31E-02	3.60E-02	3.97E-02	2.27E-03	463
L-ICRP60ED	8.1-11.3 km	1.0000	6.74E-03	7.07E-03	1.04E-02	1.38E-02	2.14E-02	2.30E-02	2.66E-02	1.14E-03	305
L-ICRP60ED	11.3-16.1 km	1.0000	4.51E-03	3.96E-03	8.03E-03	8.91E-03	1.13E-02	1.25E-02	1.56E-02	1.14E-03	305
L-ICRP60ED	16.1-20.9 km	1.0000	2.84E-03	2.33E-03	5.83E-03	7.13E-03	8.18E-03	8.67E-03	9.85E-03	1.11E-03	312
L-ICRP60ED	20.9-25.8 km	1.0000	1.91E-03	1.42E-03	3.78E-03	4.84E-03	7.16E-03	7.95E-03	9.95E-03	1.13E-03	516
L-ICRP60ED	25.8-32.2 km	1.0000	1.27E-03	1.00E-03	2.57E-03	3.22E-03	4.73E-03	5.42E-03	7.50E-03	1.13E-03	517
L-ICRP60ED	32.2-40.2 km	1.0000	8.37E-04	6.34E-04	1.58E-03	2.04E-03	2.94E-03	3.48E-03	5.08E-03	1.15E-03	324
L-ICRP60ED	40.2-48.3 km	1.0000	5.72E-04	4.38E-04	1.12E-03	1.38E-03	2.09E-03	2.27E-03	2.71E-03	1.15E-03	129
L-ICRP60ED	48.3-64.4 km	1.0000	3.75E-04	3.01E-04	7.15E-04	9.38E-04	1.32E-03	1.51E-03	2.14E-03	1.14E-03	723
L-ICRP60ED	64.4-80.5 km	1.0000	2.43E-04	1.91E-04	4.62E-04	6.02E-04	8.96E-04	1.05E-03	1.61E-03	1.14E-03	463
L-ICRP60ED	80.5-113 km	1.0000	1.49E-04	1.13E-04	2.82E-04	3.55E-04	5.44E-04	6.12E-04	9.75E-04	1.12E-03	321
L-ICRP60ED	113-161 km	1.0000	8.99E-05	7.30E-05	1.64E-04	2.09E-04	2.97E-04	3.36E-04	4.36E-04	1.15E-03	585
L-ICRP60ED	161-241 km	1.0000	4.94E-05	3.91E-05	9.37E-05	1.11E-04	1.53E-04	1.75E-04	2.27E-04	3.04E-04	640
L-ICRP60ED	241-322 km	1.0000	2.92E-05	2.34E-05	5.51E-05	6.91E-05	1.07E-04	1.21E-04	2.02E-04	3.04E-04	640
L-ICRP60ED	322-563 km	1.0000	1.41E-05	1.16E-05	2.59E-05	3.17E-05	4.41E-05	5.07E-05	8.99E-05	3.04E-04	638
L-ICRP60ED	563-805 km	1.0000	7.05E-06	5.77E-06	1.31E-05	1.64E-05	2.69E-05	3.11E-05	4.47E-05	1.52E-04	507
L-ICRP60ED	805-1609 km	1.0000	1.45E-06	9.11E-07	3.58E-06	4.44E-06	5.99E-06	6.64E-06	1.33E-05	3.04E-04	632

DOSE FOUND AT ALL LOCATIONS (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	1.00E-02 Sv	1.0000	4.05E+02	2.33E+02	1.00E+03	1.30E+03	2.14E+03	2.38E+03	3.10E+03	1.14E-03	514
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	5.00E-02 Sv	0.9037	1.68E+01	6.07E+00	3.78E+01	8.10E+01	1.69E+02	2.08E+02	2.71E+02	1.14E-03	463
AREA (ha) THAT EXCEEDS THRESHOLD											
A-THYROID Area exceeds	5.00E-02 Sv	0.2703	1.86E-01	0.00E+00	3.81E-01	4.93E-01	3.23E+00	3.90E+00	5.95E+00	1.13E-03	182

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 low density no spray
"EARLY" DESCRIPTION = OCP3 low density no spray, EARLY input
"CHRONC" DESCRIPTION = OCP3 low density no spray

SOURCE TERM 1 OF 1:
OCP3 low density no spray

RESULTS FROM THE "CHRONC" MODULE ALONE

COHORT 7 = OCP3 low density no spray

HEALTH EFFECTS CASES

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN INJ/TOTAL	0-16.1 km	1.0000	2.63E+01	2.41E+01	3.79E+01	4.31E+01	5.48E+01	5.92E+01	7.61E+01	1.12E-03	391
CAN INJ/TOTAL	0-32.2 km	1.0000	1.48E+02	1.22E+02	2.44E+02	2.83E+02	3.64E+02	4.03E+02	5.14E+02	1.12E-03	391
CAN INJ/TOTAL	0-48.3 km	1.0000	3.21E+02	2.93E+02	5.35E+02	6.01E+02	7.63E+02	8.31E+02	1.06E+03	1.12E-03	391
CAN INJ/TOTAL	0-64.4 km	1.0000	5.55E+02	4.68E+02	9.69E+02	1.16E+03	1.74E+03	2.07E+03	2.98E+03	1.15E-03	311
CAN INJ/TOTAL	0-80.5 km	1.0000	7.33E+02	6.39E+02	1.24E+03	1.53E+03	2.36E+03	2.75E+03	3.82E+03	1.15E-03	311
CAN INJ/TOTAL	0-161 km	1.0000	1.49E+03	1.14E+03	2.81E+03	3.54E+03	5.28E+03	5.72E+03	6.80E+03	1.14E-03	118
CAN INJ/TOTAL	0-322 km	1.0000	2.21E+03	1.51E+03	4.66E+03	5.98E+03	9.35E+03	1.10E+04	1.50E+04	1.15E-03	711
CAN INJ/TOTAL	0-805 km	1.0000	2.60E+03	2.01E+03	5.25E+03	6.55E+03	9.72E+03	1.12E+04	1.52E+04	1.15E-03	711
CAN INJ/TOTAL	0-1609 km	1.0000	2.67E+03	2.07E+03	5.33E+03	6.57E+03	9.72E+03	1.12E+04	1.52E+04	1.15E-03	711
CAN FAT/TOTAL	0-16.1 km	1.0000	1.18E+01	1.06E+01	1.63E+01	1.97E+01	2.46E+01	2.70E+01	3.33E+01	1.12E-03	391
CAN FAT/TOTAL	0-32.2 km	1.0000	6.52E+01	5.68E+01	1.07E+02	1.18E+02	1.48E+02	1.63E+02	2.24E+02	1.12E-03	391
CAN FAT/TOTAL	0-48.3 km	1.0000	1.41E+02	1.17E+02	2.28E+02	2.62E+02	3.41E+02	3.75E+02	4.62E+02	1.12E-03	391
CAN FAT/TOTAL	0-64.4 km	1.0000	2.43E+02	2.10E+02	4.24E+02	5.32E+02	7.73E+02	8.76E+02	1.31E+03	1.15E-03	311
CAN FAT/TOTAL	0-80.5 km	1.0000	3.21E+02	2.81E+02	5.56E+02	7.06E+02	9.62E+02	1.14E+03	1.67E+03	1.15E-03	311
CAN FAT/TOTAL	0-161 km	1.0000	6.50E+02	5.09E+02	1.22E+03	1.59E+03	2.27E+03	2.47E+03	2.98E+03	1.14E-03	118

CAN FAT/TOTAL	0-322 km	1.0000	9.63E+02	6.86E+02	1.99E+03	2.61E+03	4.23E+03	5.07E+03	6.58E+03	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	1.13E+03	8.69E+02	2.25E+03	2.85E+03	4.30E+03	5.08E+03	6.65E+03	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	1.16E+03	9.02E+02	2.28E+03	2.85E+03	4.30E+03	5.08E+03	6.65E+03	1.15E-03	711
CAN FAT/THYROID	0-16.1 km	1.0000	6.69E-02	6.19E-02	1.00E-01	1.12E-01	1.43E-01	1.59E-01	2.05E-01	1.12E-03	391
CAN FAT/THYROID	0-80.5 km	1.0000	1.97E+00	1.57E+00	3.39E+00	4.19E+00	6.06E+00	6.91E+00	1.04E+01	1.15E-03	311
CAN FAT/THYROID	0-161 km	1.0000	3.99E+00	3.10E+00	7.60E+00	9.75E+00	1.29E+01	1.45E+01	1.85E+01	1.14E-03	118
CAN FAT/THYROID	0-1609 km	1.0000	7.07E+00	5.37E+00	1.35E+01	1.74E+01	2.62E+01	3.05E+01	4.13E+01	1.15E-03	711
CAN FAT/BREAST	0-16.1 km	1.0000	9.82E-01	9.25E-01	1.34E+00	1.54E+00	2.08E+00	2.29E+00	2.81E+00	1.12E-03	391
CAN FAT/BREAST	0-80.5 km	1.0000	2.98E+01	2.53E+01	5.21E+01	6.63E+01	9.07E+01	1.04E+02	1.60E+02	1.15E-03	311
CAN FAT/BREAST	0-161 km	1.0000	5.98E+01	4.48E+01	1.14E+02	1.44E+02	2.16E+02	2.36E+02	2.86E+02	1.14E-03	118
CAN FAT/BREAST	0-1609 km	1.0000	1.03E+02	7.70E+01	2.10E+02	2.72E+02	4.26E+02	5.07E+02	6.46E+02	1.15E-03	711
CAN FAT/LUNG	0-16.1 km	1.0000	1.91E+00	1.71E+00	2.81E+00	3.19E+00	4.03E+00	4.45E+00	5.57E+00	1.12E-03	391
CAN FAT/LUNG	0-80.5 km	1.0000	5.68E+01	4.99E+01	9.82E+01	1.16E+02	1.68E+02	1.97E+02	3.01E+02	1.15E-03	311
CAN FAT/LUNG	0-161 km	1.0000	1.15E+02	8.77E+01	2.21E+02	2.86E+02	4.04E+02	4.64E+02	5.37E+02	1.14E-03	118
CAN FAT/LUNG	0-1609 km	1.0000	2.02E+02	1.43E+02	3.94E+02	5.08E+02	7.86E+02	9.73E+02	1.21E+03	1.15E-03	711
CAN FAT/LEUKEMIA	0-1609 km	1.0000	1.22E+02	9.63E+01	2.39E+02	3.03E+02	4.36E+02	5.09E+02	6.97E+02	1.15E-03	711
CAN FAT/BONE	0-1609 km	1.0000	2.95E+00	2.23E+00	5.86E+00	7.36E+00	1.09E+01	1.26E+01	1.73E+01	1.15E-03	711
CAN FAT/LIVER	0-1609 km	1.0000	3.00E+01	2.25E+01	5.96E+01	7.41E+01	1.10E+02	1.27E+02	1.74E+02	1.15E-03	711
CAN FAT/COLON	0-1609 km	1.0000	2.21E+02	1.70E+02	4.34E+02	5.40E+02	7.98E+02	9.75E+02	1.23E+03	1.15E-03	711
CAN FAT/RESIDUAL	0-1609 km	1.0000	4.73E+02	3.54E+02	9.54E+02	1.15E+03	1.72E+03	2.03E+03	2.64E+03	1.15E-03	711

POPULATION DOSE (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED TOT LIF	0-16.1 km	1.0000	2.87E+02	2.76E+02	3.96E+02	4.51E+02	5.54E+02	5.93E+02	6.86E+02	1.13E-03	247
L-ICRP60ED TOT LIF	0-80.5 km	1.0000	6.17E+03	5.36E+03	1.04E+04	1.22E+04	1.77E+04	2.08E+04	3.21E+04	1.15E-03	311
L-ICRP60ED TOT LIF	0-161 km	1.0000	1.24E+04	9.55E+03	2.35E+04	3.03E+04	4.24E+04	4.91E+04	5.71E+04	1.14E-03	118
L-ICRP60ED TOT LIF	0-1609 km	1.0000	2.19E+04	1.62E+04	4.30E+04	5.42E+04	8.34E+04	1.01E+05	1.28E+05	1.15E-03	711

POPULATION WEIGHTED RISK

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-16.1 km	1.0000	1.72E-04	1.46E-04	2.53E-04	2.92E-04	3.60E-04	3.93E-04	4.72E-04	1.12E-03	391
CAN FAT/TOTAL	0-32.2 km	1.0000	1.14E-04	1.00E-04	2.04E-04	2.24E-04	2.79E-04	3.05E-04	3.60E-04	1.12E-03	391
CAN FAT/TOTAL	0-48.3 km	1.0000	8.36E-05	7.29E-05	1.26E-04	1.47E-04	2.03E-04	2.17E-04	2.50E-04	1.12E-03	391
CAN FAT/TOTAL	0-64.4 km	1.0000	5.92E-05	5.05E-05	1.05E-04	1.29E-04	2.06E-04	2.32E-04	3.49E-04	1.15E-03	311
CAN FAT/TOTAL	0-80.5 km	1.0000	4.92E-05	4.04E-05	8.75E-05	1.08E-04	1.64E-04	1.96E-04	2.79E-04	1.15E-03	311
CAN FAT/TOTAL	0-161 km	1.0000	2.79E-05	2.05E-05	5.78E-05	7.41E-05	1.08E-04	1.19E-04	1.44E-04	1.14E-03	118
CAN FAT/TOTAL	0-322 km	1.0000	1.59E-05	9.75E-06	3.56E-05	5.09E-05	8.25E-05	1.01E-04	1.30E-04	1.15E-03	711
CAN FAT/TOTAL	0-805 km	1.0000	7.66E-06	4.97E-06	1.68E-05	2.26E-05	3.51E-05	4.15E-05	5.85E-05	1.15E-03	711
CAN FAT/TOTAL	0-1609 km	1.0000	4.37E-06	2.84E-06	9.55E-06	1.21E-05	2.00E-05	2.42E-05	3.30E-05	1.15E-03	711
CAN FAT/TOTAL	16.1-32.2 km	1.0000	1.08E-04	8.81E-05	2.01E-04	2.23E-04	2.84E-04	3.06E-04	3.51E-04	5.99E-04	382
CAN FAT/TOTAL	32.2-48.3 km	1.0000	6.85E-05	6.08E-05	1.16E-04	1.34E-04	1.86E-04	2.08E-04	2.45E-04	1.15E-03	311
CAN FAT/TOTAL	48.3-64.4 km	1.0000	4.27E-05	2.55E-05	1.01E-04	1.32E-04	2.30E-04	2.75E-04	4.34E-04	1.15E-03	311
CAN FAT/TOTAL	64.4-80.5 km	1.0000	3.20E-05	2.31E-05	6.98E-05	8.74E-05	1.23E-04	1.39E-04	1.79E-04	1.13E-03	714
CAN FAT/TOTAL	80.5-161 km	1.0000	1.92E-05	9.42E-06	4.87E-05	6.86E-05	1.11E-04	1.28E-04	1.72E-04	1.14E-03	118
CAN FAT/TOTAL	161-322 km	1.0000	8.11E-06	1.95E-06	2.39E-05	3.90E-05	7.55E-05	1.07E-04	1.50E-04	1.15E-03	711
CAN FAT/TOTAL	322-805 km	0.9920	1.10E-06	7.13E-07	2.75E-06	3.49E-06	5.26E-06	5.77E-06	8.73E-06	1.14E-03	400
CAN FAT/TOTAL	805-1609 km	0.9071	1.17E-07	6.54E-10	3.68E-07	7.47E-07	1.70E-06	2.07E-06	2.44E-06	1.14E-03	825

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	0.9977	5.87E-02	5.11E-02	5.47E-02	5.63E-02	6.03E-02	6.21E-02	6.60E-02	1.15E-03	703
L-ICRP60ED	0.2-0.5 km	1.0000	6.21E-02	5.14E-02	5.49E-02	5.65E-02	6.04E-02	6.21E-02	6.60E-02	1.14E-03	424
L-ICRP60ED	0.5-1.2 km	1.0000	6.11E-02	5.13E-02	5.49E-02	5.64E-02	6.03E-02	6.21E-02	6.60E-02	1.14E-03	371
L-ICRP60ED	1.2-1.6 km	1.0000	5.93E-02	5.12E-02	5.48E-02	5.64E-02	6.03E-02	6.21E-02	6.60E-02	1.13E-03	815
L-ICRP60ED	1.6-2.1 km	1.0000	5.76E-02	5.10E-02	5.46E-02	5.63E-02	6.02E-02	6.20E-02	6.60E-02	1.15E-03	227
L-ICRP60ED	2.1-3.2 km	1.0000	5.39E-02	5.07E-02	5.44E-02	5.60E-02	6.01E-02	6.19E-02	6.60E-02	1.14E-03	974
L-ICRP60ED	3.2-4.0 km	1.0000	5.04E-02	5.03E-02	5.40E-02	5.57E-02	5.99E-02	6.18E-02	6.60E-02	1.14E-03	926
L-ICRP60ED	4.0-4.8 km	1.0000	4.65E-02	4.57E-02	5.36E-02	5.54E-02	5.97E-02	6.16E-02	6.60E-02	1.14E-03	978
L-ICRP60ED	4.8-5.6 km	1.0000	4.31E-02	3.95E-02	5.26E-02	5.42E-02	5.82E-02	6.01E-02	6.60E-02	5.99E-04	683
L-ICRP60ED	5.6-8.1 km	1.0000	3.72E-02	3.35E-02	5.04E-02	5.25E-02	5.77E-02	6.01E-02	6.56E-02	1.13E-03	394
L-ICRP60ED	8.1-11.3 km	1.0000	3.13E-02	3.04E-02	3.84E-02	4.25E-02	5.09E-02	5.23E-02	5.98E-02	1.52E-04	326
L-ICRP60ED	11.3-16.1 km	1.0000	2.79E-02	2.51E-02	3.29E-02	3.49E-02	4.01E-02	4.25E-02	4.82E-02	1.12E-03	296
L-ICRP60ED	16.1-20.9 km	1.0000	2.67E-02	2.38E-02	3.19E-02	3.37E-02	3.83E-02	4.04E-02	4.55E-02	1.12E-03	296
L-ICRP60ED	20.9-25.8 km	1.0000	2.55E-02	2.23E-02	3.02E-02	3.12E-02	3.37E-02	3.48E-02	3.73E-02	1.12E-03	296
L-ICRP60ED	25.8-32.2 km	1.0000	2.41E-02	2.15E-02	2.72E-02	3.00E-02	3.21E-02	3.30E-02	3.51E-02	1.15E-03	695
L-ICRP60ED	32.2-40.2 km	1.0000	2.18E-02	2.06E-02	2.45E-02	2.63E-02	3.04E-02	3.10E-02	3.46E-02	1.52E-04	94
L-ICRP60ED	40.2-48.3 km	1.0000	1.93E-02	1.93E-02	2.30E-02	2.45E-02	2.82E-02	3.00E-02	3.43E-02	1.13E-03	359
L-ICRP60ED	48.3-64.4 km	1.0000	1.65E-02	1.47E-02	2.24E-02	2.38E-02	2.73E-02	2.90E-02	3.44E-02	1.13E-03	157
L-ICRP60ED	64.4-80.5 km	1.0000	1.30E-02	1.10E-02	2.11E-02	2.23E-02	2.52E-02	2.66E-02	2.99E-02	1.14E-03	328
L-ICRP60ED	80.5-113 km	1.0000	9.06E-03	7.13E-03	1.83E-02	2.10E-02	2.44E-02	2.60E-02	2.99E-02	1.13E-03	479
L-ICRP60ED	113-161 km	1.0000	5.90E-03	4.18E-03	1.16E-02	1.49E-02	2.11E-02	2.22E-02	2.88E-02	1.14E-04	88
L-ICRP60ED	161-241 km	1.0000	3.15E-03	2.29E-03	6.35E-03	8.40E-03	1.35E-02	1.62E-02	2.48E-02	1.14E-03	458
L-ICRP60ED	241-322 km	1.0000	1.72E-03	1.20E-03	3.51E-03	5.02E-03	7.57E-03	8.62E-03	1.47E-02	1.13E-03	317
L-ICRP60ED	322-563 km	1.0000	7.95E-04	6.18E-04	1.56E-03	2.09E-03	3.16E-03	3.65E-03	4.96E-03	1.13E-03	314
L-ICRP60ED	563-805 km	1.0000	5.06E-04	3.16E-04	1.13E-03	1.36E-03	2.04E-03	2.27E-03	2.83E-03	1.15E-03	508
L-ICRP60ED	805-1609 km	1.0000	1.04E-04	6.41E-05	2.65E-04	3.21E-04	4.23E-04	4.77E-04	7.48E-04	3.04E-04	633

L-ICRP60ED POP. DOSE (Sv) 0-16.1 km

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
TOTAL LONG-TERM PATHWAYS DOSE	1.0000	2.87E+02	2.76E+02	3.96E+02	4.51E+02	5.54E+02	5.93E+02	6.86E+02	1.13E-03	247
LONG-TERM DIRECT EXPOSURE PATHWAYS	1.0000	1.51E+02	1.25E+02	2.22E+02	2.51E+02	3.22E+02	3.49E+02	4.16E+02	1.12E-03	391
TOTAL INGESTION PATHWAYS DOSE	1.0000	1.27E+02	1.10E+02	2.02E+02	2.21E+02	2.71E+02	2.96E+02	3.41E+02	1.14E-03	297
LONG-TERM GROUNDSHINE DOSE	1.0000	1.51E+02	1.24E+02	2.22E+02	2.50E+02	3.21E+02	3.48E+02	4.14E+02	1.12E-03	391
LONG-TERM RESUSPENSION DOSE	1.0000	6.70E-01	5.88E-01	1.07E+00	1.24E+00	1.77E+00	2.04E+00	2.55E+00	1.13E-03	247
WATER INGESTION DOSE	1.0000	7.45E+01	6.99E+01	1.16E+02	1.30E+02	1.66E+02	1.85E+02	2.43E+02	1.14E-04	278
POP.-DEPENDENT DECONTAMINATION DOSE	1.0000	7.90E+00	7.11E+00	1.46E+01	1.84E+01	2.48E+01	2.77E+01	4.32E+01	5.99E-04	587
FARM-DEPENDENT DECONTAMINATION DOSE	1.0000	7.76E-01	6.59E-01	1.38E+00	1.64E+00	2.14E+00	2.26E+00	3.15E+00	1.52E-04	326
INGESTION OF GRAINS	1.0000	1.74E+00	1.15E+00	3.79E+00	4.87E+00	7.03E+00	7.78E+00	9.64E+00	1.14E-03	133
INGESTION OF LEAF VEG	1.0000	7.17E+00	6.78E+00	1.14E+01	1.25E+01	1.57E+01	1.73E+01	2.31E+01	1.14E-03	398
INGESTION OF ROOT CROPS	1.0000	4.99E+00	4.68E+00	8.71E+00	1.01E+01	1.12E+01	1.18E+01	1.51E+01	1.14E-04	148
INGESTION OF FRUITS	1.0000	4.45E+00	3.61E+00	8.94E+00	1.06E+01	1.36E+01	1.51E+01	1.90E+01	1.14E-03	133
INGESTION OF LEGUMES	1.0000	7.74E+00	7.42E+00	1.14E+01	1.23E+01	1.48E+01	1.60E+01	2.17E+01	1.14E-04	80
INGESTION OF BEEF	1.0000	1.08E+01	8.12E+00	2.28E+01	2.91E+01	3.84E+01	4.30E+01	5.94E+01	8.56E-04	954
INGESTION OF MILK	1.0000	1.11E+01	7.79E+00	2.25E+01	3.10E+01	5.06E+01	7.35E+01	1.31E+02	1.14E-03	375
INGESTION OF POULTRY	1.0000	3.44E+00	2.07E+00	7.86E+00	1.03E+01	1.50E+01	1.76E+01	2.31E+01	1.14E-03	398
INGESTION OF OTHER MEAT CROPS	1.0000	6.24E-01	4.40E-01	1.25E+00	1.66E+00	2.43E+00	2.73E+00	3.52E+00	1.14E-03	398

L-ICRP60ED POP. DOSE (Sv) 0-80.5 km

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
TOTAL LONG-TERM PATHWAYS DOSE	1.0000	6.17E+03	5.36E+03	1.04E+04	1.22E+04	1.77E+04	2.08E+04	3.21E+04	1.15E-03	311
LONG-TERM DIRECT EXPOSURE PATHWAYS	1.0000	5.39E+03	4.51E+03	9.79E+03	1.17E+04	1.75E+04	2.08E+04	3.06E+04	1.15E-03	311
TOTAL INGESTION PATHWAYS DOSE	1.0000	7.54E+02	5.56E+02	1.44E+03	1.86E+03	2.67E+03	3.06E+03	3.82E+03	1.14E-03	388
LONG-TERM GROUNDSHINE DOSE	1.0000	5.35E+03	4.45E+03	9.64E+03	1.15E+04	1.67E+04	1.97E+04	3.03E+04	1.15E-03	311
LONG-TERM RESUSPENSION DOSE	1.0000	3.73E+01	3.12E+01	6.81E+01	8.83E+01	1.29E+02	1.49E+02	2.20E+02	1.15E-03	311

WATER INGESTION DOSE	1.0000	9.23E+01	8.64E+01	1.31E+02	1.50E+02	2.02E+02	2.11E+02	2.70E+02	1.14E-04	278
POP.-DEPENDENT DECONTAMINATION DOSE	1.0000	2.09E+01	1.44E+01	4.21E+01	5.59E+01	9.69E+01	1.20E+02	1.93E+02	1.09E-03	287
FARM-DEPENDENT DECONTAMINATION DOSE	1.0000	9.23E-01	7.67E-01	1.88E+00	2.21E+00	2.98E+00	3.33E+00	5.19E+00	3.71E-04	89
INGESTION OF GRAINS	1.0000	2.51E+01	1.34E+01	6.64E+01	8.04E+01	1.13E+02	1.27E+02	1.65E+02	1.14E-03	133
INGESTION OF LEAF VEG	1.0000	5.01E+01	2.99E+01	1.15E+02	1.54E+02	2.63E+02	3.17E+02	4.60E+02	1.14E-03	388
INGESTION OF ROOT CROPS	1.0000	3.87E+01	2.44E+01	9.14E+01	1.10E+02	1.50E+02	1.72E+02	2.19E+02	1.14E-03	133
INGESTION OF FRUITS	1.0000	5.11E+01	3.23E+01	1.18E+02	1.45E+02	2.16E+02	2.40E+02	3.18E+02	1.14E-03	133
INGESTION OF LEGUMES	1.0000	4.03E+01	2.95E+01	8.41E+01	1.02E+02	1.33E+02	1.50E+02	1.91E+02	1.14E-03	133
INGESTION OF BEEF	1.0000	1.99E+02	1.50E+02	3.92E+02	5.00E+02	6.70E+02	7.60E+02	1.05E+03	1.14E-03	133
INGESTION OF MILK	1.0000	1.81E+02	1.23E+02	3.74E+02	4.91E+02	8.85E+02	1.09E+03	1.65E+03	1.14E-03	375
INGESTION OF POULTRY	1.0000	6.57E+01	3.72E+01	1.41E+02	1.99E+02	3.42E+02	4.08E+02	5.89E+02	1.14E-03	388
INGESTION OF OTHER MEAT CROPS	1.0000	1.01E+01	6.09E+00	2.22E+01	2.95E+01	5.35E+01	6.55E+01	8.89E+01	1.14E-03	388

L-ICRP60ED POP. DOSE (Sv) 0-161 km	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
TOTAL LONG-TERM PATHWAYS DOSE	1.0000	1.24E+04	9.55E+03	2.35E+04	3.03E+04	4.24E+04	4.91E+04	5.71E+04	1.14E-03	118
LONG-TERM DIRECT EXPOSURE PATHWAYS	1.0000	1.06E+04	7.66E+03	2.16E+04	2.81E+04	4.02E+04	4.63E+04	5.46E+04	1.14E-03	118
TOTAL INGESTION PATHWAYS DOSE	1.0000	1.78E+03	1.21E+03	3.65E+03	4.49E+03	6.47E+03	7.32E+03	9.07E+03	1.14E-03	388
LONG-TERM GROUNDSHINE DOSE	1.0000	1.05E+04	7.62E+03	2.14E+04	2.80E+04	4.02E+04	4.63E+04	5.42E+04	1.14E-03	118
LONG-TERM RESUSPENSION DOSE	1.0000	7.59E+01	5.32E+01	1.54E+02	2.08E+02	3.11E+02	3.37E+02	4.00E+02	1.14E-03	718
WATER INGESTION DOSE	1.0000	1.06E+02	1.01E+02	1.43E+02	1.65E+02	2.10E+02	2.19E+02	2.81E+02	1.14E-04	278
POP.-DEPENDENT DECONTAMINATION DOSE	1.0000	2.17E+01	1.48E+01	4.43E+01	5.99E+01	1.09E+02	1.45E+02	2.38E+02	1.14E-03	118
FARM-DEPENDENT DECONTAMINATION DOSE	1.0000	9.26E-01	7.70E-01	1.88E+00	2.21E+00	2.98E+00	3.33E+00	5.19E+00	3.71E-04	89
INGESTION OF GRAINS	1.0000	6.81E+01	5.31E+01	1.65E+02	2.08E+02	2.73E+02	3.08E+02	4.16E+02	1.14E-03	140
INGESTION OF LEAF VEG	1.0000	1.25E+02	7.57E+01	3.12E+02	4.12E+02	7.32E+02	8.94E+02	1.26E+03	1.14E-03	388
INGESTION OF ROOT CROPS	1.0000	9.69E+01	7.51E+01	2.27E+02	2.76E+02	3.64E+02	4.03E+02	5.52E+02	1.14E-03	140
INGESTION OF FRUITS	1.0000	1.35E+02	1.03E+02	3.30E+02	3.91E+02	5.51E+02	6.16E+02	8.03E+02	1.14E-03	140
INGESTION OF LEGUMES	1.0000	9.19E+01	7.27E+01	2.09E+02	2.41E+02	3.30E+02	3.71E+02	4.79E+02	1.14E-03	140
INGESTION OF BEEF	1.0000	5.23E+02	4.14E+02	1.07E+03	1.23E+03	1.71E+03	1.97E+03	2.66E+03	1.14E-03	140
INGESTION OF MILK	1.0000	4.24E+02	3.09E+02	8.76E+02	1.09E+03	1.69E+03	2.04E+03	3.66E+03	1.15E-03	380
INGESTION OF POULTRY	1.0000	1.79E+02	1.03E+02	4.21E+02	5.66E+02	9.59E+02	1.15E+03	1.65E+03	1.14E-03	388
INGESTION OF OTHER MEAT CROPS	1.0000	2.69E+01	1.55E+01	6.38E+01	8.33E+01	1.41E+02	1.76E+02	2.48E+02	1.14E-03	388

L-ICRP60ED POP. DOSE (Sv) 0-1609 km	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
TOTAL LONG-TERM PATHWAYS DOSE	1.0000	2.19E+04	1.62E+04	4.30E+04	5.42E+04	8.34E+04	1.01E+05	1.28E+05	1.15E-03	711
LONG-TERM DIRECT EXPOSURE PATHWAYS	1.0000	1.68E+04	1.09E+04	3.70E+04	5.11E+04	7.98E+04	9.75E+04	1.27E+05	1.15E-03	711
TOTAL INGESTION PATHWAYS DOSE	1.0000	5.07E+03	2.80E+03	1.18E+04	1.52E+04	2.39E+04	2.77E+04	3.40E+04	1.13E-03	396
LONG-TERM GROUNDSHINE DOSE	1.0000	1.67E+04	1.09E+04	3.67E+04	5.09E+04	7.98E+04	9.75E+04	1.26E+05	1.15E-03	711
LONG-TERM RESUSPENSION DOSE	1.0000	1.22E+02	7.96E+01	2.76E+02	3.59E+02	5.90E+02	7.11E+02	9.38E+02	1.15E-03	711
WATER INGESTION DOSE	1.0000	3.26E+02	2.89E+02	5.22E+02	5.46E+02	6.07E+02	6.35E+02	7.03E+02	1.13E-03	793
POP.-DEPENDENT DECONTAMINATION DOSE	1.0000	2.17E+01	1.48E+01	4.43E+01	5.99E+01	1.09E+02	1.45E+02	2.38E+02	1.14E-03	118
FARM-DEPENDENT DECONTAMINATION DOSE	1.0000	9.26E-01	7.70E-01	1.88E+00	2.21E+00	2.98E+00	3.33E+00	5.19E+00	3.71E-04	89
INGESTION OF GRAINS	1.0000	2.06E+02	1.03E+02	5.79E+02	7.47E+02	1.12E+03	1.24E+03	1.56E+03	1.14E-03	133
INGESTION OF LEAF VEG	1.0000	3.68E+02	1.25E+02	9.47E+02	1.32E+03	2.76E+03	3.41E+03	5.33E+03	1.13E-03	396
INGESTION OF ROOT CROPS	1.0000	2.82E+02	1.29E+02	7.78E+02	9.82E+02	1.34E+03	1.52E+03	2.06E+03	1.14E-03	133
INGESTION OF FRUITS	1.0000	4.05E+02	2.09E+02	1.12E+03	1.39E+03	2.15E+03	2.39E+03	3.00E+03	1.14E-03	133
INGESTION OF LEGUMES	1.0000	2.53E+02	1.18E+02	7.01E+02	8.45E+02	1.20E+03	1.36E+03	1.77E+03	1.14E-03	133
INGESTION OF BEEF	1.0000	1.51E+03	8.98E+02	3.65E+03	4.97E+03	7.24E+03	8.02E+03	1.09E+04	1.14E-03	984
INGESTION OF MILK	1.0000	1.10E+03	6.45E+02	2.74E+03	3.47E+03	5.20E+03	5.72E+03	7.38E+03	1.14E-03	984
INGESTION OF POULTRY	1.0000	5.40E+02	2.30E+02	1.32E+03	1.86E+03	3.69E+03	4.57E+03	7.10E+03	1.13E-03	396
INGESTION OF OTHER MEAT CROPS	1.0000	8.01E+01	3.50E+01	2.01E+02	2.74E+02	5.71E+02	7.28E+02	1.06E+03	1.13E-03	396

ECONOMIC COST MEASURES ($) 0-1609 km	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
TOTAL ECONOMIC COSTS	1.0000	5.37E+09	3.06E+09	1.17E+10	1.54E+10	3.55E+10	4.68E+10	1.18E+11	1.14E-03	118
POP.-DEPENDENT COSTS	1.0000	5.14E+09	2.84E+09	1.13E+10	1.51E+10	3.55E+10	4.68E+10	1.17E+11	1.14E-03	118
FARM-DEPENDENT COSTS	1.0000	2.27E+08	1.70E+08	4.51E+08	5.91E+08	8.95E+08	1.05E+09	1.57E+09	1.14E-03	2
POP.-DEPENDENT DECONTAMINATION COST	1.0000	5.54E+08	3.22E+08	1.20E+09	1.60E+09	3.55E+09	4.68E+09	1.18E+10	1.14E-03	118
FARM-DEPENDENT DECONTAMINATION COST	1.0000	2.67E+07	2.25E+07	5.06E+07	6.21E+07	8.98E+07	1.02E+08	1.18E+08	1.14E-03	98
POP.-DEPENDENT INTERDICTION COST	1.0000	4.58E+09	2.55E+09	1.06E+10	1.41E+10	2.92E+10	3.76E+10	1.05E+11	1.14E-03	118
FARM-DEPENDENT INTERDICTION COST	1.0000	9.07E+07	6.29E+07	1.85E+08	2.68E+08	4.00E+08	4.62E+08	1.03E+09	1.14E-03	2
POP.-DEPENDENT CONDEMNATION COST	0.5055	8.22E+06	2.09E+05	2.58E+07	4.13E+07	6.76E+07	9.49E+07	1.25E+08	1.14E-03	463
FARM-DEPENDENT CONDEMNATION COST	0.9977	1.61E+06	8.07E+05	4.29E+06	5.81E+06	8.40E+06	9.50E+06	1.18E+07	1.14E-03	546
EMERGENCY PHASE COST	0.8925	7.37E+05	4.35E+05	1.86E+06	2.58E+06	3.62E+06	4.01E+06	5.06E+06	1.14E-03	514
INTERMEDIATE PHASE COST	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
MILK DISPOSAL COST	1.0000	6.04E+06	3.42E+06	1.42E+07	1.96E+07	3.13E+07	3.29E+07	4.22E+07	1.52E-04	87
CROP DISPOSAL COST	1.0000	1.02E+08	7.38E+07	2.16E+08	2.75E+08	4.14E+08	4.88E+08	7.68E+08	1.13E-03	379

ECONOMIC COST MEASURES ($) 0-16.1 km	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
TOTAL ECONOMIC COSTS	1.0000	5.08E+08	4.62E+08	7.65E+08	8.53E+08	1.06E+09	1.13E+09	1.30E+09	1.13E-03	586
POP.-DEPENDENT COSTS	1.0000	4.62E+08	4.11E+08	7.19E+08	7.97E+08	1.01E+09	1.06E+09	1.20E+09	1.13E-03	586
FARM-DEPENDENT COSTS	1.0000	4.56E+07	4.10E+07	7.63E+07	8.49E+07	1.08E+08	1.19E+08	1.47E+08	1.11E-03	392
POP.-DEPENDENT DECONTAMINATION COST	1.0000	6.67E+07	6.23E+07	1.04E+08	1.12E+08	1.33E+08	1.44E+08	2.15E+08	1.14E-04	79
FARM-DEPENDENT DECONTAMINATION COST	1.0000	9.66E+06	8.97E+06	1.47E+07	1.76E+07	2.13E+07	2.22E+07	2.75E+07	1.14E-04	79
POP.-DEPENDENT INTERDICTION COST	1.0000	3.86E+08	3.44E+08	5.83E+08	6.64E+08	8.01E+08	8.60E+08	1.02E+09	1.13E-03	586
FARM-DEPENDENT INTERDICTION COST	1.0000	1.45E+07	1.22E+07	2.28E+07	2.58E+07	3.27E+07	3.53E+07	4.17E+07	1.12E-03	391
POP.-DEPENDENT CONDEMNATION COST	0.5055	8.22E+06	2.09E+05	2.58E+07	4.13E+07	6.76E+07	9.49E+07	1.25E+08	1.14E-03	463
FARM-DEPENDENT CONDEMNATION COST	0.9977	1.61E+06	8.07E+05	4.29E+06	5.81E+06	8.40E+06	9.50E+06	1.18E+07	1.14E-03	546
EMERGENCY PHASE COST	0.8925	7.22E+05	4.35E+05	1.84E+06	2.46E+06	3.48E+06	3.82E+06	4.67E+06	1.14E-03	459
INTERMEDIATE PHASE COST	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
MILK DISPOSAL COST	1.0000	1.69E+06	1.36E+06	3.31E+06	3.75E+06	4.99E+06	5.52E+06	6.86E+06	1.11E-03	392
CROP DISPOSAL COST	1.0000	1.82E+07	1.56E+07	3.28E+07	3.66E+07	4.73E+07	5.52E+07	7.39E+07	1.11E-03	392

ECONOMIC COST MEASURES ($) 16.1-32.2 km	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
TOTAL ECONOMIC COSTS	0.9886	1.81E+09	1.22E+09	4.08E+09	5.36E+09	7.43E+09	7.91E+09	1.01E+10	3.71E-04	89
POP.-DEPENDENT COSTS	0.9684	1.74E+09	1.17E+09	3.97E+09	5.24E+09	7.32E+09	7.83E+09	9.87E+09	4.85E-04	89
FARM-DEPENDENT COSTS	0.9886	6.22E+07	5.06E+07	1.15E+08	1.32E+08	1.83E+08	2.10E+08	2.85E+08	1.11E-03	392
POP.-DEPENDENT DECONTAMINATION COST	0.9684	1.89E+08	1.23E+08	4.31E+08	5.74E+08	8.12E+08	8.95E+08	1.66E+09	3.71E-04	89
FARM-DEPENDENT DECONTAMINATION COST	0.9684	8.70E+06	7.44E+06	1.68E+07	2.08E+07	2.69E+07	3.01E+07	5.21E+07	1.43E-04	77
POP.-DEPENDENT INTERDICTION COST	0.9684	1.55E+09	1.08E+09	3.55E+09	4.64E+09	6.32E+09	7.04E+09	8.21E+09	4.85E-04	89
FARM-DEPENDENT INTERDICTION COST	0.9886	2.20E+07	1.93E+07	3.93E+07	4.84E+07	6.18E+07	6.82E+07	8.34E+07	1.14E-04	78
POP.-DEPENDENT CONDEMNATION COST	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
FARM-DEPENDENT CONDEMNATION COST	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EMERGENCY PHASE COST	0.0176	1.52E+04	0.00E+00	0.00E+00	0.00E+00	5.87E+05	1.08E+06	1.81E+06	1.14E-03	264
INTERMEDIATE PHASE COST	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
MILK DISPOSAL COST	0.9875	2.13E+06	8.34E+05	5.91E+06	7.28E+06	1.01E+07	1.06E+07	1.42E+07	1.14E-04	78
CROP DISPOSAL COST	0.9886	2.94E+07	2.12E+07	6.45E+07	7.62E+07	1.04E+08	1.25E+08	1.83E+08	1.11E-03	392

ECONOMIC COST MEASURES ($) 32.2-48.3 km	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL

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 TOTAL ECONOMIC COSTS              0.8293  1.37E+09  5.05E+08  3.98E+09  5.28E+09  7.27E+09  8.13E+09  1.29E+10  1.43E-04  144
  POP.-DEPENDENT COSTS            0.6521  1.33E+09  4.71E+08  3.88E+09  5.21E+09  7.17E+09  8.06E+09  1.26E+10  1.43E-04  144
  FARM-DEPENDENT COSTS             0.8293  4.16E+07  3.34E+07  8.95E+07  1.08E+08  1.45E+08  1.65E+08  2.41E+08  1.43E-04   77
  POP.-DEPENDENT DECONTAMINATION COST  0.6521  1.36E+08  4.72E+07  3.92E+08  5.29E+08  7.57E+08  8.35E+08  1.42E+09  1.14E-03  143
  FARM-DEPENDENT DECONTAMINATION COST  0.6521  4.30E+06  2.85E+06  1.08E+07  1.31E+07  2.04E+07  2.35E+07  3.62E+07  1.43E-04   76
  POP.-DEPENDENT INTERDICTION COST    0.6521  1.19E+09  4.07E+08  3.49E+09  4.72E+09  6.81E+09  7.81E+09  1.14E+10  1.43E-04  144
  FARM-DEPENDENT INTERDICTION COST    0.8293  1.66E+07  1.29E+07  3.46E+07  4.23E+07  6.16E+07  7.09E+07  9.58E+07  1.14E-03   50
  POP.-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  EMERGENCY PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  INTERMEDIATE PHASE COST           0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  MILK DISPOSAL COST             0.7923  1.10E+06  3.57E+05  3.38E+06  4.50E+06  7.27E+06  8.27E+06  1.11E+07  1.43E-04   77
  CROP DISPOSAL COST             0.8293  1.96E+07  1.36E+07  4.53E+07  5.69E+07  7.86E+07  8.65E+07  1.30E+08  1.11E-03  392

                        PROB                  QUANTILES                    PEAK     PEAK  PEAK
                      NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
ECONOMIC COST MEASURES ($)  48.3-64.4 km
 TOTAL ECONOMIC COSTS              0.5507  8.04E+08  1.34E+07  2.25E+09  4.37E+09  1.30E+10  1.71E+10  3.73E+10  8.56E-04  303
  POP.-DEPENDENT COSTS            0.2875  7.82E+08  0.00E+00  2.20E+09  4.37E+09  1.30E+10  1.71E+10  3.73E+10  8.56E-04  303
  FARM-DEPENDENT COSTS             0.5507  2.23E+07  1.12E+07  6.05E+07  7.85E+07  1.23E+08  1.44E+08  2.26E+08  1.14E-03    2
  POP.-DEPENDENT DECONTAMINATION COST  0.2875  7.88E+07  0.00E+00  2.20E+08  4.37E+08  1.30E+09  1.71E+09  3.76E+09  8.56E-04  303
  FARM-DEPENDENT DECONTAMINATION COST  0.2875  1.90E+06  0.00E+00  7.03E+06  9.74E+06  1.84E+07  2.35E+07  3.43E+07  1.14E-03    2
  POP.-DEPENDENT INTERDICTION COST    0.2875  7.03E+08  0.00E+00  1.97E+09  3.64E+09  1.11E+10  1.34E+10  3.36E+10  8.56E-04  303
  FARM-DEPENDENT INTERDICTION COST    0.5507  1.03E+07  4.39E+06  2.96E+07  3.61E+07  5.42E+07  6.14E+07  1.23E+08  1.14E-03    2
  POP.-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  EMERGENCY PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  INTERMEDIATE PHASE COST           0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  MILK DISPOSAL COST             0.4750  3.06E+05  0.00E+00  9.18E+05  1.53E+06  2.76E+06  3.53E+06  7.58E+06  1.14E-03    2
  CROP DISPOSAL COST             0.5507  9.77E+06  4.11E+06  2.66E+07  3.59E+07  5.75E+07  6.52E+07  7.84E+07  1.14E-03  387

                        PROB                  QUANTILES                    PEAK     PEAK  PEAK
                      NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
ECONOMIC COST MEASURES ($)  64.4-80.5 km
 TOTAL ECONOMIC COSTS              0.3696  3.73E+08  0.00E+00  4.52E+08  3.00E+09  6.93E+09  8.35E+09  1.62E+10  1.14E-04  417
  POP.-DEPENDENT COSTS            0.1227  3.57E+08  0.00E+00  4.52E+08  2.92E+09  6.93E+09  8.35E+09  1.61E+10  1.14E-04  417
  FARM-DEPENDENT COSTS             0.3696  1.59E+07  0.00E+00  5.29E+07  7.76E+07  1.22E+08  1.43E+08  2.02E+08  1.14E-03  119
  POP.-DEPENDENT DECONTAMINATION COST  0.1227  3.61E+07  0.00E+00  4.52E+07  2.99E+08  6.93E+08  8.35E+08  1.62E+09  1.14E-04  417
  FARM-DEPENDENT DECONTAMINATION COST  0.1227  1.03E+06  0.00E+00  3.47E+06  6.49E+06  1.72E+07  2.07E+07  3.53E+07  1.14E-04  417
  POP.-DEPENDENT INTERDICTION COST    0.1227  3.21E+08  0.00E+00  3.83E+08  2.57E+09  6.34E+09  8.15E+09  1.45E+10  1.14E-04  417
  FARM-DEPENDENT INTERDICTION COST    0.3696  7.43E+06  0.00E+00  2.46E+07  3.53E+07  7.05E+07  7.87E+07  1.00E+08  1.11E-03   75
  POP.-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  EMERGENCY PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  INTERMEDIATE PHASE COST           0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  MILK DISPOSAL COST             0.2873  2.15E+05  0.00E+00  7.00E+05  1.15E+06  3.24E+06  3.97E+06  6.22E+06  1.14E-03    2
  CROP DISPOSAL COST             0.3696  7.23E+06  0.00E+00  2.46E+07  3.47E+07  6.69E+07  7.72E+07  9.82E+07  1.14E-03  119

                        PROB                  QUANTILES                    PEAK     PEAK  PEAK
                      NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
ECONOMIC COST MEASURES ($)   80.5-161 km
 TOTAL ECONOMIC COSTS              0.2185  5.07E+08  0.00E+00  1.47E+08  6.35E+08  1.13E+10  2.56E+10  1.16E+11  1.14E-03  118
  POP.-DEPENDENT COSTS            0.0469  4.73E+08  0.00E+00  0.00E+00  0.00E+00  1.13E+10  2.56E+10  1.16E+11  1.14E-03  118
  FARM-DEPENDENT COSTS             0.2185  3.41E+07  0.00E+00  1.12E+08  2.11E+08  5.25E+08  6.05E+08  9.03E+08  1.13E-03  379
  POP.-DEPENDENT DECONTAMINATION COST  0.0469  4.76E+07  0.00E+00  0.00E+00  0.00E+00  1.13E+09  2.56E+09  1.17E+10  1.14E-03  118
  FARM-DEPENDENT DECONTAMINATION COST  0.0469  1.15E+06  0.00E+00  0.00E+00  0.00E+00  3.55E+07  4.46E+07  6.63E+07  1.14E-04   88
  POP.-DEPENDENT INTERDICTION COST    0.0469  4.25E+08  0.00E+00  0.00E+00  0.00E+00  9.27E+09  2.34E+10  1.04E+11  1.14E-03  118
  FARM-DEPENDENT INTERDICTION COST    0.2185  1.65E+07  0.00E+00  5.43E+07  1.08E+08  2.40E+08  2.96E+08  4.11E+08  1.13E-03  837
  POP.-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  EMERGENCY PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  INTERMEDIATE PHASE COST           0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  MILK DISPOSAL COST             0.1543  5.45E+05  0.00E+00  8.12E+05  2.49E+06  1.32E+07  1.96E+07  2.88E+07  1.13E-03  837
  CROP DISPOSAL COST             0.2185  1.59E+07  0.00E+00  5.32E+07  9.91E+07  2.14E+08  2.52E+08  6.25E+08  1.13E-03  379

                        PROB                  QUANTILES                    PEAK     PEAK  PEAK
                      NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
ECONOMIC COST MEASURES ($)    161-322 km
 TOTAL ECONOMIC COSTS              0.0160  5.35E+06  0.00E+00  0.00E+00  0.00E+00  1.63E+08  4.10E+08  7.64E+08  1.14E-03  884
  POP.-DEPENDENT COSTS            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT COSTS             0.0160  5.35E+06  0.00E+00  0.00E+00  0.00E+00  1.63E+08  4.10E+08  7.64E+08  1.14E-03  884
  POP.-DEPENDENT DECONTAMINATION COST  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT DECONTAMINATION COST  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  POP.-DEPENDENT INTERDICTION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT INTERDICTION COST    0.0160  3.40E+06  0.00E+00  0.00E+00  0.00E+00  1.29E+08  2.24E+08  5.50E+08  2.27E-03    1
  POP.-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  EMERGENCY PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  INTERMEDIATE PHASE COST           0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  MILK DISPOSAL COST             0.0125  6.68E+04  0.00E+00  0.00E+00  0.00E+00  7.59E+05  4.07E+06  1.63E+07  2.29E-03  851
  CROP DISPOSAL COST             0.0160  1.88E+06  0.00E+00  0.00E+00  0.00E+00  3.41E+07  1.37E+08  4.92E+08  1.14E-03  884

                        PROB                  QUANTILES                    PEAK     PEAK  PEAK
                      NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
ECONOMIC COST MEASURES ($)    322-805 km
 TOTAL ECONOMIC COSTS              0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  POP.-DEPENDENT COSTS            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT COSTS             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  POP.-DEPENDENT DECONTAMINATION COST  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT DECONTAMINATION COST  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  POP.-DEPENDENT INTERDICTION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT INTERDICTION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  POP.-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  EMERGENCY PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  INTERMEDIATE PHASE COST           0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  MILK DISPOSAL COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  CROP DISPOSAL COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                        PROB                  QUANTILES                    PEAK     PEAK  PEAK
                      NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
ECONOMIC COST MEASURES ($)   805-1609 km
 TOTAL ECONOMIC COSTS              0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  POP.-DEPENDENT COSTS            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT COSTS             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  POP.-DEPENDENT DECONTAMINATION COST  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT DECONTAMINATION COST  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  POP.-DEPENDENT INTERDICTION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT INTERDICTION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
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POP.-DEPENDENT CONDEMNATION COST     0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
FARM-DEPENDENT CONDEMNATION COST     0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
EMERGENCY PHASE COST                 0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
INTERMEDIATE PHASE COST              0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
MILK DISPOSAL COST                   0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
CROP DISPOSAL COST                   0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB                 QUANTILES                PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
MAXIMUM LONG-TERM ACTION DISTANCE (km)
FARM-DEPENDENT DECONTAMINATION DIST.   1.0000  4.92E+01  3.89E+01  7.83E+01  9.80E+01  1.52E+02    ****    1.61E+02  8.09E-03   73
POP.-DEPENDENT DECONTAMINATION DIST.   1.0000  4.92E+01  3.89E+01  7.83E+01  9.80E+01  1.52E+02    ****    1.61E+02  8.09E-03   73
FARM-DEPENDENT INTERDICTION DIST.      1.0000  7.35E+01  5.76E+01  1.24E+02  1.48E+02  2.15E+02  2.39E+02  3.22E+02  1.13E-03  317
POP.-DEPENDENT INTERDICTION DIST.      1.0000  4.92E+01  3.89E+01  7.83E+01  9.80E+01  1.52E+02    ****    1.61E+02  8.09E-03   73
FARM-DEPENDENT CONDEMNATION DIST.      0.9977  3.13E+00  3.19E+00  5.60E+00  6.73E+00    ****      ****    8.05E+00  4.32E-02   56
POP.-DEPENDENT CONDEMNATION DIST.      0.5055  1.84E+00  1.06E+00  4.33E+00  5.24E+00    ****      ****    8.05E+00  2.18E-02   56
MILK DISPOSAL DIST.                    1.0000  6.59E+01  5.03E+01  1.13E+02  1.37E+02    ****      ****    2.41E+02  1.25E-02    1
CROP DISPOSAL DIST.                    1.0000  7.35E+01  5.76E+01  1.24E+02  1.48E+02  2.15E+02  2.39E+02  3.22E+02  1.13E-03  317

                          PROB                 QUANTILES                PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION     0-1609 km
FARM DECONTAMINATION (ha)               1.0000  1.71E+04  1.27E+04  3.27E+04  4.17E+04  6.08E+04  6.88E+04  8.49E+04  1.14E-03  295
POP. DECONTAMINATION (INDIVIDUALS)      1.0000  7.20E+04  4.04E+04  1.53E+05  2.17E+05  4.57E+05  6.45E+05  1.66E+06  1.14E-03  118
POP. DECONTAMINATION AREA (ha)          1.0000  2.21E+04  1.77E+04  4.21E+04  5.62E+04  8.57E+04  9.83E+04  1.21E+05  1.15E-03  602
FARM INTERDICTION (ha)                  1.0000  3.93E+04  2.65E+04  8.34E+04  1.13E+05  1.86E+05  2.32E+05  4.11E+05  1.14E-03    2
POP. INTERDICTION (INDIVIDUALS)         1.0000  7.20E+04  4.04E+04  1.53E+05  2.17E+05  4.57E+05  6.45E+05  1.66E+06  1.14E-03  118
POP. INTERDICTION AREA (ha)             1.0000  2.21E+04  1.77E+04  4.21E+04  5.62E+04  8.57E+04  9.83E+04  1.21E+05  1.15E-03  602
FARM CONDEMNATION (ha)                  0.9977  7.90E+01  4.54E+01  2.05E+02  2.55E+02  3.38E+02  3.64E+02  4.91E+02  3.23E-04   90
POP. CONDEMNATION (INDIVIDUALS)         0.5055  2.97E+01  1.15E+00  9.42E+01  1.38E+02  2.57E+02  3.10E+02  4.46E+02  1.14E-03  463
POP. CONDEMNATION AREA (ha)             0.9954  3.86E+01  1.61E+01  1.02E+02  1.38E+02  2.44E+02  2.93E+02  3.81E+02  1.13E-03  439
MILK DISPOSAL AREA (ha)                 1.0000  3.20E+04  2.14E+04  6.52E+04  9.64E+04  1.50E+05  1.81E+05  4.11E+05  1.14E-03    2
CROP DISPOSAL AREA (ha)                 1.0000  3.94E+04  2.67E+04  8.34E+04  1.13E+05  1.86E+05  2.32E+05  4.11E+05  1.14E-03    2

                          PROB                 QUANTILES                PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION     0-16.1 km
FARM DECONTAMINATION (ha)               1.0000  4.84E+03  4.46E+03  7.70E+03  8.64E+03  1.11E+04  1.22E+04  1.51E+04  1.11E-03  392
POP. DECONTAMINATION (INDIVIDUALS)      1.0000  5.90E+03  5.40E+03  8.95E+03  1.02E+04  1.23E+04  1.33E+04  1.57E+04  1.13E-03  586
POP. DECONTAMINATION AREA (ha)          1.0000  4.92E+03  4.43E+03  7.59E+03  8.44E+03  1.06E+04  1.14E+04  1.44E+04  5.99E-04  382
FARM INTERDICTION (ha)                  1.0000  5.77E+03  5.37E+03  9.64E+03  1.09E+04  1.41E+04  1.57E+04  1.98E+04  1.12E-03  391
POP. INTERDICTION (INDIVIDUALS)         1.0000  5.90E+03  5.40E+03  8.95E+03  1.02E+04  1.23E+04  1.33E+04  1.57E+04  1.13E-03  586
POP. INTERDICTION AREA (ha)             1.0000  4.92E+03  4.43E+03  7.59E+03  8.44E+03  1.06E+04  1.14E+04  1.44E+04  5.99E-04  382
FARM CONDEMNATION (ha)                  0.9977  7.90E+01  4.54E+01  2.05E+02  2.55E+02  3.38E+02  3.64E+02  4.91E+02  3.23E-04   90
POP. CONDEMNATION (INDIVIDUALS)         0.5055  2.97E+01  1.15E+00  9.42E+01  1.38E+02  2.57E+02  3.10E+02  4.46E+02  1.14E-03  463
POP. CONDEMNATION AREA (ha)             0.9954  3.86E+01  1.61E+01  1.02E+02  1.38E+02  2.44E+02  2.93E+02  3.81E+02  1.13E-03  439
MILK DISPOSAL AREA (ha)                 1.0000  5.62E+03  5.27E+03  9.12E+03  1.03E+04  1.21E+04  1.30E+04  1.51E+04  1.11E-03  392
CROP DISPOSAL AREA (ha)                 1.0000  5.85E+03  5.45E+03  9.72E+03  1.10E+04  1.41E+04  1.57E+04  1.98E+04  1.12E-03  391

                          PROB                 QUANTILES                PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION     16.1-32.2 km
FARM DECONTAMINATION (ha)               0.9684  5.96E+03  5.10E+03  1.10E+04  1.24E+04  1.65E+04  1.87E+04  2.90E+04  1.43E-04   77
POP. DECONTAMINATION (INDIVIDUALS)      0.9684  2.45E+04  1.60E+04  5.76E+04  7.36E+04  1.02E+05  1.07E+05  1.29E+05  3.71E-04   89
POP. DECONTAMINATION AREA (ha)          0.9684  7.29E+03  7.05E+03  1.17E+04  1.32E+04  1.73E+04  1.95E+04  2.97E+04  1.12E-03  464
FARM INTERDICTION (ha)                  0.9886  9.12E+03  7.99E+03  1.57E+04  1.98E+04  2.68E+04  3.04E+04  4.09E+04  1.11E-03  392
POP. INTERDICTION (INDIVIDUALS)         0.9684  2.45E+04  1.60E+04  5.76E+04  7.36E+04  1.02E+05  1.07E+05  1.29E+05  3.71E-04   89
POP. INTERDICTION AREA (ha)             0.9684  7.29E+03  7.05E+03  1.17E+04  1.32E+04  1.73E+04  1.95E+04  2.97E+04  1.12E-03  464
FARM CONDEMNATION (ha)                  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION (INDIVIDUALS)         0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION AREA (ha)             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
MILK DISPOSAL AREA (ha)                 0.9875  8.29E+03  7.26E+03  1.42E+04  1.75E+04  2.33E+04  2.54E+04  3.32E+04  1.14E-04   78
CROP DISPOSAL AREA (ha)                 0.9886  9.12E+03  7.99E+03  1.57E+04  1.98E+04  2.68E+04  3.04E+04  4.09E+04  1.11E-03  392

                          PROB                 QUANTILES                PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION     32.2-48.3 km
FARM DECONTAMINATION (ha)               0.6521  3.19E+03  2.08E+03  7.92E+03  1.02E+04  1.40E+04  1.60E+04  2.41E+04  1.43E-04  144
POP. DECONTAMINATION (INDIVIDUALS)      0.6521  1.87E+04  6.51E+03  5.61E+04  7.39E+04  1.02E+05  1.12E+05  1.79E+05  1.43E-04  144
POP. DECONTAMINATION AREA (ha)          0.6521  4.60E+03  3.40E+03  1.09E+04  1.28E+04  1.88E+04  2.10E+04  3.29E+04  1.14E-04   14
FARM INTERDICTION (ha)                  0.8293  7.15E+03  6.03E+03  1.46E+04  1.88E+04  2.81E+04  3.17E+04  3.81E+04  1.14E-03   50
POP. INTERDICTION (INDIVIDUALS)         0.6521  1.87E+04  6.51E+03  5.61E+04  7.39E+04  1.02E+05  1.12E+05  1.79E+05  1.43E-04  144
POP. INTERDICTION AREA (ha)             0.6521  4.60E+03  3.40E+03  1.09E+04  1.28E+04  1.88E+04  2.10E+04  3.29E+04  1.14E-04   14
FARM CONDEMNATION (ha)                  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION (INDIVIDUALS)         0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION AREA (ha)             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
MILK DISPOSAL AREA (ha)                 0.7923  5.93E+03  4.69E+03  1.27E+04  1.59E+04  2.42E+04  2.79E+04  3.81E+04  1.14E-03   50
CROP DISPOSAL AREA (ha)                 0.8293  7.15E+03  6.03E+03  1.46E+04  1.88E+04  2.81E+04  3.17E+04  3.81E+04  1.14E-03   50

                          PROB                 QUANTILES                PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION     48.3-64.4 km
FARM DECONTAMINATION (ha)               0.2875  1.43E+03  0.00E+00  5.03E+03  7.57E+03  1.31E+04  1.61E+04  2.58E+04  1.14E-03    2
POP. DECONTAMINATION (INDIVIDUALS)      0.2875  1.11E+04  0.00E+00  3.15E+04  6.12E+04  1.73E+05  2.32E+05  5.28E+05  8.56E-04  303
POP. DECONTAMINATION AREA (ha)          0.2875  2.46E+03  0.00E+00  8.69E+03  1.24E+04  2.11E+04  2.36E+04  3.54E+04  1.14E-03    2
FARM INTERDICTION (ha)                  0.5507  4.61E+03  2.11E+03  1.23E+04  1.62E+04  2.62E+04  3.14E+04  4.91E+04  1.14E-03    2
POP. INTERDICTION (INDIVIDUALS)         0.2875  1.11E+04  0.00E+00  3.15E+04  6.12E+04  1.73E+05  2.32E+05  5.28E+05  8.56E-04  303
POP. INTERDICTION AREA (ha)             0.2875  2.46E+03  0.00E+00  8.69E+03  1.24E+04  2.11E+04  2.36E+04  3.54E+04  1.14E-03    2
FARM CONDEMNATION (ha)                  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION (INDIVIDUALS)         0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION AREA (ha)             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
MILK DISPOSAL AREA (ha)                 0.4750  3.48E+03  0.00E+00  1.04E+04  1.33E+04  2.20E+04  2.56E+04  4.91E+04  1.14E-03    2
CROP DISPOSAL AREA (ha)                 0.5507  4.61E+03  2.11E+03  1.23E+04  1.62E+04  2.62E+04  3.14E+04  4.91E+04  1.14E-03    2

                          PROB                 QUANTILES                PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION     64.4-80.5 km
FARM DECONTAMINATION (ha)               0.1227  7.66E+02  0.00E+00  3.11E+03  4.68E+03  1.20E+04  1.42E+04  2.65E+04  1.14E-04  417
POP. DECONTAMINATION (INDIVIDUALS)      0.1227  5.06E+03  0.00E+00  6.22E+03  3.87E+04  9.89E+04  1.25E+05  2.28E+05  1.14E-04  417
POP. DECONTAMINATION AREA (ha)          0.1227  1.33E+03  0.00E+00  4.86E+03  9.19E+03  2.18E+04  2.43E+04  4.08E+04  1.14E-04  417
FARM INTERDICTION (ha)                  0.3696  3.39E+03  0.00E+00  1.14E+04  1.56E+04  2.93E+04  3.26E+04  3.98E+04  1.11E-03   75
POP. INTERDICTION (INDIVIDUALS)         0.1227  5.06E+03  0.00E+00  6.22E+03  3.87E+04  9.89E+04  1.25E+05  2.28E+05  1.14E-04  417
POP. INTERDICTION AREA (ha)             0.1227  1.33E+03  0.00E+00  4.86E+03  9.19E+03  2.18E+04  2.43E+04  4.08E+04  1.14E-04  417
FARM CONDEMNATION (ha)                  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION (INDIVIDUALS)         0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION AREA (ha)             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
MILK DISPOSAL AREA (ha)                 0.2873  2.43E+03  0.00E+00  8.77E+03  1.27E+04  2.59E+04  3.16E+04  3.98E+04  1.11E-03   75
CROP DISPOSAL AREA (ha)                 0.3696  3.39E+03  0.00E+00  1.14E+04  1.56E+04  2.93E+04  3.26E+04  3.98E+04  1.11E-03   75
```

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                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
AFFECTED AREA/POPULATION     80.5-161 km
 FARM DECONTAMINATION (ha)            0.0469  8.64E+02  0.00E+00  0.00E+00  0.00E+00  2.70E+04  3.17E+04  4.98E+04  1.14E-04   88
 POP. DECONTAMINATION (INDIVIDUALS)    0.0469  6.70E+03  0.00E+00  0.00E+00  0.00E+00  1.41E+05  3.66E+05  1.64E+06  1.14E-03  118
 POP. DECONTAMINATION AREA (ha)        0.0469  1.50E+03  0.00E+00  0.00E+00  0.00E+00  4.61E+04  5.49E+04  8.64E+04  1.14E-03  853
 FARM INTERDICTION (ha)              0.2185  7.76E+03  0.00E+00  2.55E+04  4.50E+04  1.04E+05  1.28E+05  2.01E+05  1.13E-03  379
 POP. INTERDICTION (INDIVIDUALS)       0.0469  6.70E+03  0.00E+00  0.00E+00  0.00E+00  1.41E+05  3.66E+05  1.64E+06  1.14E-03  118
 POP. INTERDICTION AREA (ha)          0.0469  1.50E+03  0.00E+00  0.00E+00  0.00E+00  4.61E+04  5.49E+04  8.64E+04  1.14E-03  853
 FARM CONDEMNATION (ha)              0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. CONDEMNATION (INDIVIDUALS)       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. CONDEMNATION AREA (ha)          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL AREA (ha)             0.1543  5.10E+03  0.00E+00  1.57E+04  3.37E+04  8.44E+04  1.06E+05  1.64E+05  1.13E-03  837
 CROP DISPOSAL AREA (ha)             0.2185  7.76E+03  0.00E+00  2.55E+04  4.50E+04  1.04E+05  1.28E+05  2.01E+05  1.13E-03  379

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
AFFECTED AREA/POPULATION     161-322 km
 FARM DECONTAMINATION (ha)            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. DECONTAMINATION (INDIVIDUALS)    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. DECONTAMINATION AREA (ha)        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 FARM INTERDICTION (ha)              0.0160  1.51E+03  0.00E+00  0.00E+00  0.00E+00  6.07E+04  1.22E+05  2.19E+05  2.27E-03    1
 POP. INTERDICTION (INDIVIDUALS)       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. INTERDICTION AREA (ha)          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 FARM CONDEMNATION (ha)              0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. CONDEMNATION (INDIVIDUALS)       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. CONDEMNATION AREA (ha)          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL AREA (ha)             0.0125  1.16E+03  0.00E+00  0.00E+00  0.00E+00  5.08E+04  1.10E+05  2.19E+05  2.27E-03    1
 CROP DISPOSAL AREA (ha)             0.0160  1.51E+03  0.00E+00  0.00E+00  0.00E+00  6.07E+04  1.22E+05  2.19E+05  2.27E-03    1

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
AFFECTED AREA/POPULATION     322-805 km
 FARM DECONTAMINATION (ha)            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. DECONTAMINATION (INDIVIDUALS)    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. DECONTAMINATION AREA (ha)        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 FARM INTERDICTION (ha)              0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. INTERDICTION (INDIVIDUALS)       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. INTERDICTION AREA (ha)          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 FARM CONDEMNATION (ha)              0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. CONDEMNATION (INDIVIDUALS)       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. CONDEMNATION AREA (ha)          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL AREA (ha)             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CROP DISPOSAL AREA (ha)             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
AFFECTED AREA/POPULATION     805-1609 km
 FARM DECONTAMINATION (ha)            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. DECONTAMINATION (INDIVIDUALS)    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. DECONTAMINATION AREA (ha)        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 FARM INTERDICTION (ha)              0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. INTERDICTION (INDIVIDUALS)       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. INTERDICTION AREA (ha)          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 FARM CONDEMNATION (ha)              0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. CONDEMNATION (INDIVIDUALS)       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 POP. CONDEMNATION AREA (ha)          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL AREA (ha)             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CROP DISPOSAL AREA (ha)             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
MAXIMUM ANNUAL FOOD DOSE (EFFECTIVE)
 PROJECTED FOR INDIVIDUAL 11.3-16.1 km  1.0000  1.22E-02  1.04E-02  2.36E-02  2.83E-02  3.12E-02  3.18E-02  3.31E-02  1.14E-03  140
 PROJECTED FOR INDIVIDUAL 25.8-32.2 km  1.0000  1.30E-02  1.11E-02  2.38E-02  2.76E-02  3.10E-02  3.16E-02  3.30E-02  1.13E-03  132
 PROJECTED FOR INDIVIDUAL 40.2-48.3 km  1.0000  1.31E-02  1.10E-02  2.39E-02  2.73E-02  3.09E-02  3.15E-02  3.28E-02  1.15E-03  152
 PROJECTED FOR INDIVIDUAL 48.3-64.4 km  1.0000  1.29E-02  1.08E-02  2.42E-02  2.81E-02  3.11E-02  3.17E-02  3.30E-02  1.14E-03  221
 PROJECTED FOR INDIVIDUAL 64.4-80.5 km  1.0000  1.15E-02  9.67E-03  2.24E-02  2.55E-02  3.04E-02  3.08E-02  3.31E-02  1.43E-04  159
 PROJECTED FOR INDIVIDUAL   113-161 km  1.0000  7.33E-03  4.56E-03  1.72E-02  2.11E-02  2.61E-02  2.85E-02  3.07E-02  1.14E-03  216
 PROJECTED FOR INDIVIDUAL   241-322 km  1.0000  2.67E-03  1.31E-03  6.91E-03  9.43E-03  1.57E-02  1.95E-02  2.97E-02  1.14E-03  318
 PROJECTED FOR INDIVIDUAL   563-805 km  1.0000  1.04E-03  2.79E-04  3.12E-03  3.85E-03  6.26E-03  7.45E-03  1.22E-02  1.13E-03  334
 PROJECTED FOR INDIVIDUAL  805-1609 km  1.0000  1.21E-04  3.49E-05  3.69E-04  5.60E-04  8.79E-04  1.03E-03  1.51E-03  1.12E-03  335

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
MAXIMUM ANNUAL FOOD DOSE (THYROID)
 PROJECTED FOR INDIVIDUAL 11.3-16.1 km  1.0000  1.35E-02  1.09E-02  2.90E-02  3.09E-02  3.36E-02  3.48E-02  3.75E-02  1.14E-03  170
 PROJECTED FOR INDIVIDUAL 25.8-32.2 km  1.0000  1.44E-02  1.19E-02  2.94E-02  3.10E-02  3.37E-02  3.49E-02  3.76E-02  1.13E-03  132
 PROJECTED FOR INDIVIDUAL 40.2-48.3 km  1.0000  1.45E-02  1.16E-02  3.00E-02  3.10E-02  3.34E-02  3.45E-02  3.69E-02  1.14E-03  213
 PROJECTED FOR INDIVIDUAL 48.3-64.4 km  1.0000  1.43E-02  1.15E-02  2.87E-02  3.06E-02  3.22E-02  3.30E-02  3.74E-02  1.14E-04  148
 PROJECTED FOR INDIVIDUAL 64.4-80.5 km  1.0000  1.27E-02  1.03E-02  2.59E-02  3.02E-02  3.29E-02  3.41E-02  3.68E-02  1.14E-03  141
 PROJECTED FOR INDIVIDUAL   113-161 km  1.0000  8.03E-03  5.14E-03  2.00E-02  2.31E-02  3.06E-02  3.16E-02  3.40E-02  1.14E-03  216
 PROJECTED FOR INDIVIDUAL   241-322 km  1.0000  2.89E-03  1.39E-03  7.29E-03  9.97E-03  1.67E-02  2.07E-02  3.14E-02  1.14E-03  318
 PROJECTED FOR INDIVIDUAL   563-805 km  1.0000  1.09E-03  2.94E-04  3.25E-03  4.05E-03  6.56E-03  7.62E-03  1.22E-02  1.13E-03  334
 PROJECTED FOR INDIVIDUAL  805-1609 km  1.0000  1.25E-04  3.53E-05  3.85E-04  5.77E-04  9.09E-04  1.07E-03  1.48E-03  1.12E-03  335

 **** Indicates that the value is outside resolution of the analysis.
 Optionally increase number of trials for better resolution.

Successful completion of MACCS2 was achieved!
This job required a total of    1793.750 CPU seconds

   Input processing required      0.406 CPU seconds
       Simulation required    1791.734 CPU seconds
  Output processing required      1.609 CPU seconds
```

MACCS2 11/13/2012 11:02:12 Version 3.7.0.0 : 11/9/12 110212.781
P1: ATMOS USER INPUT (UNIT 24) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Input\Atmos1.inp
P2: EARLY USER INPUT (UNIT 25) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Input\Early1.inp
P3: CHRONC USER INPUT (UNIT 26) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Input\Chronc1.inp
P4: METEOROLOGY DATA (UNIT 28) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Data\PB MACCS2 2006 Met Data 64WD.inp
P5: SITE DATA INPUT (UNIT 29) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Input\Sumpop_site.inp
P6: LIST OUTPUT (UNIT 06) = C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Output\Model1.out

USER INPUT IS READ FROM UNIT 24
RECORD IDENTIFIER FIELDS 11 CHARACTERS LONG ARE EXPECTED.
THE FIRST 499 COLUMNS OF EACH INPUT RECORD ARE PROCESSED.

```
RECORD
NUMBER                RECORD

   * File created using WinMACCS version 3.7.0 11/13/2012 10:58:48 AM
   *
   * MACCS2 Cyclical File: C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Data\LNT.txt
   *
   * Peach Bottom Revision 7 for Spent Fuel Pool Scoping Study
   *
   * The initial WinMACCS file for the seismic runs was created May 12, 2009 using the Jan 21 2009 file for the PB STSBO.
   *
   * Identifies this MACCS calculation
 1  RIATNAM1001    'OCP3 high density no spray'
   *
   * NUMRAD, Number of Radial Spatial Elements
 2  GENUMRAD001    26
   *
   * SPAEND, Spatial Endpoint Distances (km)
 3  GESPAEND001    0.16
 4  GESPAEND002    0.52
 5  GESPAEND003    1.21
 6  GESPAEND004    1.61
 7  GESPAEND005    2.13
 8  GESPAEND006    3.22
 9  GESPAEND007    4.02
10  GESPAEND008    4.83
11  GESPAEND009    5.63
12  GESPAEND010    8.05
13  GESPAEND011    11.27
14  GESPAEND012    16.09
15  GESPAEND013    20.92
16  GESPAEND014    25.75
17  GESPAEND015    32.19
18  GESPAEND016    40.23
19  GESPAEND017    48.28
20  GESPAEND018    64.37
21  GESPAEND019    80.47
22  GESPAEND020    112.65
23  GESPAEND021    160.93
24  GESPAEND022    241.14
25  GESPAEND023    321.87
26  GESPAEND024    563.27
27  GESPAEND025    804.67
28  GESPAEND026    1609.34
   *
   * Form 'Site File' Comment:
   * Updated to 2011 using Census Bureau data and CPI data.
   *
   * NUMCOR, Number of angular compass directions
29  GENUMCOR001    64
   *
   * Form 'Radionuclides' Comment:
   * From ORIGEN, and updated to correct for isotope-by-isotope release fractions (which cannot be done in the chemical group release fractions).
   *
   * NUMISO, Number of Nuclides
30  ISNUMISO001    69
   *
   * Form 'Chemical Names' Comment:
   * Group names are imported from MELMACCS.
   *
   * MAXGRP, Number of Element Groups
31  ISMAXGRP001    9
   *
   * Form 'Wet/Dry Depos Flags' Comment:
   * No change
   *
   * WETDEP, DRYDEP, Wet and Dry Deposition Flags for Each Nuclide Group
32  ISDEPFLA001    .FALSE. .FALSE.
33  ISDEPFLA002    .TRUE.  .TRUE.
34  ISDEPFLA003    .TRUE.  .TRUE.
35  ISDEPFLA004    .TRUE.  .TRUE.
36  ISDEPFLA005    .TRUE.  .TRUE.
37  ISDEPFLA006    .TRUE.  .TRUE.
38  ISDEPFLA007    .TRUE.  .TRUE.
39  ISDEPFLA008    .TRUE.  .TRUE.
40  ISDEPFLA009    .TRUE.  .TRUE.
   *
   * NUMSTB_ZERO = 0
41  ISNUMSTB001    0
   *
   * Form 'Pseudostable Radionuclides' Comment:
   * Come in thru MELMACCS.
   *
   * NUMSTB, Number of Pseudostable Radionuclides
42  ISNUMSTB001    16
********  RECORD NUMBER   42 REPLACES RECORD NUMBER   41  ********
   *
   * NAMSTB, List of Pseudostable Radionuclides
43  ISNAMSTB001    I-129
44  ISNAMSTB002    Xe-131m
45  ISNAMSTB003    Xe-133m
46  ISNAMSTB004    Cs-135
47  ISNAMSTB005    Sm-147
48  ISNAMSTB006    U-234
49  ISNAMSTB007    U-235
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50  ISNAMSTB008      U-236
51  ISNAMSTB009      U-237
52  ISNAMSTB010      Np-237
53  ISNAMSTB011      Rb-87
54  ISNAMSTB012      Zr-93
55  ISNAMSTB013      Nb-93m
56  ISNAMSTB014      Nb-95m
57  ISNAMSTB015      Tc-99
58  ISNAMSTB016      Pm-147
    *
    * NUCNAM, IGROUP, Chemical group associated with each nuclide
59  ISOTPGRP001      Kr-85   1
60  ISOTPGRP002      Kr-85m  1
61  ISOTPGRP003      Kr-87   1
62  ISOTPGRP004      Kr-88   1
63  ISOTPGRP005      Xe-133  1
64  ISOTPGRP006      Xe-135  1
65  ISOTPGRP007      Xe-135m 1
66  ISOTPGRP008      Cs-134  2
67  ISOTPGRP009      Cs-136  2
68  ISOTPGRP010      Cs-137  2
69  ISOTPGRP011      Rb-86   2
70  ISOTPGRP012      Rb-88   2
71  ISOTPGRP013      Ba-139  3
72  ISOTPGRP014      Ba-140  3
73  ISOTPGRP015      Sr-89   3
74  ISOTPGRP016      Sr-90   3
75  ISOTPGRP017      Sr-91   3
76  ISOTPGRP018      Sr-92   3
77  ISOTPGRP019      Ba-137m 3
78  ISOTPGRP020      I-131   4
79  ISOTPGRP021      I-132   4
80  ISOTPGRP022      I-133   4
81  ISOTPGRP023      I-134   4
82  ISOTPGRP024      I-135   4
83  ISOTPGRP025      Te-127  5
84  ISOTPGRP026      Te-127m 5
85  ISOTPGRP027      Te-129  5
86  ISOTPGRP028      Te-129m 5
87  ISOTPGRP029      Te-131m 5
88  ISOTPGRP030      Te-132  5
89  ISOTPGRP031      Te-131  5
90  ISOTPGRP032      Rh-105  6
91  ISOTPGRP033 Ru-103  6
92  ISOTPGRP034      Ru-105  6
93  ISOTPGRP035      Ru-106  6
94  ISOTPGRP036      Rh-103m 6
95  ISOTPGRP037     Rh-106  6
96  ISOTPGRP038      Nb-95   7
97  ISOTPGRP039      Co-58   7
98  ISOTPGRP040      Co-60   7
99  ISOTPGRP041      Mo-99   7
100  ISOTPGRP042      Tc-99m  7
101  ISOTPGRP043      Nb-97   7
102  ISOTPGRP044      Nb-97m  7
103  ISOTPGRP045      Ce-141  8
104  ISOTPGRP046      Ce-143  8
105  ISOTPGRP047      Ce-144  8
106  ISOTPGRP048      Np-239  8
107  ISOTPGRP049      Pu-238  8
108  ISOTPGRP050      Pu-239  8
109  ISOTPGRP051      Pu-240  8
110  ISOTPGRP052      Pu-241  8
111  ISOTPGRP053      Zr-95   8
112  ISOTPGRP054      Zr-97   8
113  ISOTPGRP055      Am-241  9
114  ISOTPGRP056      Cm-242  9
115  ISOTPGRP057      Cm-244  9
116  ISOTPGRP058      La-140  9
117  ISOTPGRP059      La-141  9
118  ISOTPGRP060      La-142  9
119  ISOTPGRP061      Nd-147  9
120  ISOTPGRP062      Pr-143  9
121  ISOTPGRP063      Y-90    9
122  ISOTPGRP064      Y-91    9
123  ISOTPGRP065      Y-92    9
124  ISOTPGRP066      Y-93    9
125  ISOTPGRP067      Y-91m   9
126  ISOTPGRP068      Pr-144  9
127  ISOTPGRP069      Pr-144m 9
    *
    * Form 'Wet Deposition' Comment:
    * Values from Nate et al's report, table 7, page 64 (April 2007).  Derived  assuming 1 micrometer particles.  Do not change.
    *
    * CWASH1, Washout Coefficient Number One, Linear Factor
128  WDCWASH1001      1.89E-05
    *
    * CWASH2, Washout Coefficient Number Two, Exponential Factor
129  WDCWASH2001      .664
    *
    * Form 'Dry Deposition' Comment:
    * Value Given by Nate.  MELMACCS cannot currently calculate a DDV based on a surface roughness greater than 20 cm
    *
    * NPSGRP, Number of Particle Size Groups
130  DDNPSGRP001      10
    *
    * VDEPOS, Dry Deposition Velocities for Each Particle Size Group (m/sec)
131  DDVDEPOS001      0.0011
132  DDVDEPOS002      0.001
133  DDVDEPOS003      0.0014
134  DDVDEPOS004      0.0023
135  DDVDEPOS005      0.0045
136  DDVDEPOS006      0.0092
137  DDVDEPOS007      0.0177
138  DDVDEPOS008      0.0291
139  DDVDEPOS009      0.0367
140  DDVDEPOS010      0.0367
    *
    * Form 'Dispersion Function' Comment:
    * From Nate's draft report (April 2007).
```

```
*
* CYSIGA, Dispersion function parameter
141  DPCYSIGA001     .7507
142  DPCYSIGA002     .7507
143  DPCYSIGA003     .4063
144  DPCYSIGA004     .2779
145  DPCYSIGA005     .2158
146  DPCYSIGA006     .2158
*
* CYSIGB, Dispersion function parameter
147  DPCYSIGB001     .866
148  DPCYSIGB002     .866
149  DPCYSIGB003     .865
150  DPCYSIGB004     .881
151  DPCYSIGB005     .866
152  DPCYSIGB006     .866
*
* CZSIGA, Dispersion function parameter
153  DPCZSIGA001     .0361
154  DPCZSIGA002     .0361
155  DPCZSIGA003     .2036
156  DPCZSIGA004     .2636
157  DPCZSIGA005     .2463
158  DPCZSIGA006     .2463
*
* CZSIGB, Dispersion function parameter
159  DPCZSIGB001     1.277
160  DPCZSIGB002     1.277
161  DPCZSIGB003     .859
162  DPCZSIGB004     .751
163  DPCZSIGB005     .619
164  DPCZSIGB006     .619
*
* Form 'Scaling Factors' Comment:
* ZSCALE correspond to a surface roughness of 60 cm.  The formula for calculating it is in the NUREG/CR-4691.
*
* YSCALE, linear scaling factor for sigma-y
165  DPYSCALE001     1.
*
* ZSCALE, linear scaling factor for sigma-z
166  DPZSCALE001     1.82
*
* DISPMD - dispersion long-range model
167  DPDISPMD001     LRDIST
*
* MNDMOD, plume meander model
168  PMMNDMOD001     NEW
*
* WINSP1, wind speed where the meander factor changes from constant to decreasing
169  PMWINSP1001     2.
*
* WINSP2, wind speed where the meander factor reaches one
170  PMWINSP2001     6.
*
* MNDIST, distance, for use in 1.145
171  PMMNDIST001     800.
*
* MNDFAC, plume meander stability class factors, for use in 1.145
172  PMMNDFAC001     1.
173  PMMNDFAC002     1.
174  PMMNDFAC003     1.
175  PMMNDFAC004     2.
176  PMMNDFAC005     3.
177  PMMNDFAC006     4.
*
* Form 'Plume Rise Scale Factor' Comment:
* Using standard modeling options.
*
* SCLCRW, scaling factor for entrainment of buoyant plume
178  PRSCLCRW001     1.
*
* SCLADP, scaling factor for the a–d stability plume rise formula
179  PRSCLADP001     1.
*
* SCLEFP, scaling factor for the e–f stability plume rise formula
180  PRSCLEFP001     1.
*
* Form 'Wake Effect Data' Comment:
* Data for Peach Bottom from NUREG-1150.
*
* BUILDH, building height (meters)
181  WEBUILDH001     50.
182  WEBUILDH002     50.
183  WEBUILDH003     50.
184  WEBUILDH004     50.
185  WEBUILDH005     50.
186  WEBUILDH006     50.
187  WEBUILDH007     50.
188  WEBUILDH008     50.
189  WEBUILDH009     50.
190  WEBUILDH010     50.
191  WEBUILDH011     50.
192  WEBUILDH012     50.
193  WEBUILDH013     50.
194  WEBUILDH014     50.
195  WEBUILDH015     50.
196  WEBUILDH016     50.
197  WEBUILDH017     50.
198  WEBUILDH018     50.
199  WEBUILDH019     50.
200  WEBUILDH020     50.
201  WEBUILDH021     50.
202  WEBUILDH022     50.
203  WEBUILDH023     50.
204  WEBUILDH024     50.
205  WEBUILDH025     50.
206  WEBUILDH026      50.
207  WEBUILDH027     50.
208  WEBUILDH028     50.
209  WEBUILDH029     50.
```

210 WEBUILDH030 50.
211 WEBUILDH031 50.
212 WEBUILDH032 50.
213 WEBUILDH033 50.
214 WEBUILDH034 50.
215 WEBUILDH035 50.
216 WEBUILDH036 50.
217 WEBUILDH037 50.
218 WEBUILDH038 50.
219 WEBUILDH039 50.
220 WEBUILDH040 50.
221 WEBUILDH041 50.
222 WEBUILDH042 50.
223 WEBUILDH043 50.
224 WEBUILDH044 50.
225 WEBUILDH045 50.
226 WEBUILDH046 50.
227 WEBUILDH047 50.
228 WEBUILDH048 50.
229 WEBUILDH049 50.
230 WEBUILDH050 50.
231 WEBUILDH051 50.
232 WEBUILDH052 50.
233 WEBUILDH053 50.
234 WEBUILDH054 50.
235 WEBUILDH055 50.
236 WEBUILDH056 50.
237 WEBUILDH057 50.
238 WEBUILDH058 50.
239 WEBUILDH059 50.
240 WEBUILDH060 50.
241 WEBUILDH061 50.
242 WEBUILDH062 50.
243 WEBUILDH063 50.
244 WEBUILDH064 50.
*
* SIGYINIT, initial value of sigma-y for each of the plumes (meters)
245 SIGYINIT001 11.6
246 SIGYINIT002 11.6
247 SIGYINIT003 11.6
248 SIGYINIT004 11.6
249 SIGYINIT005 11.6
250 SIGYINIT006 11.6
251 SIGYINIT007 11.6
252 SIGYINIT008 11.6
253 SIGYINIT009 11.6
254 SIGYINIT010 11.6
255 SIGYINIT011 11.6
256 SIGYINIT012 11.6
257 SIGYINIT013 11.6
258 SIGYINIT014 11.6
259 SIGYINIT015 11.6
260 SIGYINIT016 11.6
261 SIGYINIT017 11.6
262 SIGYINIT018 11.6
263 SIGYINIT019 11.6
264 SIGYINIT020 11.6
265 SIGYINIT021 11.6
266 SIGYINIT022 11.6
267 SIGYINIT023 11.6
268 SIGYINIT024 11.6
269 SIGYINIT025 11.6
270 SIGYINIT026 11.6
271 SIGYINIT027 11.6
272 SIGYINIT028 11.6
273 SIGYINIT029 11.6
274 SIGYINIT030 11.6
275 SIGYINIT031 11.6
276 SIGYINIT032 11.6
277 SIGYINIT033 11.6
278 SIGYINIT034 11.6
279 SIGYINIT035 11.6
280 SIGYINIT036 11.6
281 SIGYINIT037 11.6
282 SIGYINIT038 11.6
283 SIGYINIT039 11.6
284 SIGYINIT040 11.6
285 SIGYINIT041 11.6
286 SIGYINIT042 11.6
287 SIGYINIT043 11.6
288 SIGYINIT044 11.6
289 SIGYINIT045 11.6
290 SIGYINIT046 11.6
291 SIGYINIT047 11.6
292 SIGYINIT048 11.6
293 SIGYINIT049 11.6
294 SIGYINIT050 11.6
295 SIGYINIT051 11.6
296 SIGYINIT052 11.6
297 SIGYINIT053 11.6
298 SIGYINIT054 11.6
299 SIGYINIT055 11.6
300 SIGYINIT056 11.6
301 SIGYINIT057 11.6
302 SIGYINIT058 11.6
303 SIGYINIT059 11.6
304 SIGYINIT060 11.6
305 SIGYINIT061 11.6
306 SIGYINIT062 11.6
307 SIGYINIT063 11.6
308 SIGYINIT064 11.6
*
* SIGZINIT, initial value of sigma-z for each of the plumes (meters)
309 SIGZINIT001 23.3
310 SIGZINIT002 23.3
311 SIGZINIT003 23.3
312 SIGZINIT004 23.3
313 SIGZINIT005 23.3
314 SIGZINIT006 23.3
315 SIGZINIT007 23.3

```
316  SIGZINIT008      23.3
317  SIGZINIT009      23.3
318  SIGZINIT010      23.3
319  SIGZINIT011      23.3
320  SIGZINIT012      23.3
321  SIGZINIT013      23.3
322  SIGZINIT014      23.3
323  SIGZINIT015      23.3
324  SIGZINIT016      23.3
325  SIGZINIT017      23.3
326  SIGZINIT018      23.3
327  SIGZINIT019      23.3
328  SIGZINIT020      23.3
329  SIGZINIT021      23.3
330  SIGZINIT022      23.3
331  SIGZINIT023      23.3
332  SIGZINIT024      23.3
333  SIGZINIT025      23.3
334  SIGZINIT026      23.3
335  SIGZINIT027      23.3
336  SIGZINIT028      23.3
337  SIGZINIT029      23.3
338  SIGZINIT030      23.3
339  SIGZINIT031      23.3
340  SIGZINIT032      23.3
341  SIGZINIT033      23.3
342  SIGZINIT034      23.3
343  SIGZINIT035      23.3
344  SIGZINIT036      23.3
345  SIGZINIT037      23.3
346  SIGZINIT038      23.3
347  SIGZINIT039      23.3
348  SIGZINIT040      23.3
349  SIGZINIT041      23.3
350  SIGZINIT042      23.3
351  SIGZINIT043      23.3
352  SIGZINIT044      23.3
353  SIGZINIT045      23.3
354  SIGZINIT046      23.3
355  SIGZINIT047      23.3
356  SIGZINIT048      23.3
357  SIGZINIT049      23.3
358  SIGZINIT050      23.3
359  SIGZINIT051      23.3
360  SIGZINIT052      23.3
361  SIGZINIT053      23.3
362  SIGZINIT054      23.3
363  SIGZINIT055      23.3
364  SIGZINIT056      23.3
365  SIGZINIT057      23.3
366  SIGZINIT058      23.3
367  SIGZINIT059      23.3
368  SIGZINIT060      23.3
369  SIGZINIT061      23.3
370  SIGZINIT062      23.3
371  SIGZINIT063   23.3
372  SIGZINIT064      23.3
     *
     * ATNAM2, specific descriptive text describing this particular source term
373  RDATNAM2001      'OCP3 high density no spray'
     *
     * OALARM, time after accident initiation that off-site alarm is initiated (sec)
374  RDOALARM001      3600.
     *
     * Form 'Plume Parameters' Comment:
     * These values come from MELCOR PTF file.  Plume discretization is done by user.
     *
     * NUMREL, number of plumes
375  RDNUMREL001      64
     *
     * MAXRIS, selection of risk-dominant plume segment
376  RDMAXRIS001      14
     *
     * REFTIM, representative time point for dispersion and radioactive decay
377  RDREFTIM001      0.
378  RDREFTIM002      0.5
379  RDREFTIM003      0.5
380  RDREFTIM004      0.5
381  RDREFTIM005      0.5
382  RDREFTIM006      0.5
383  RDREFTIM007      0.5
384  RDREFTIM008      0.5
385  RDREFTIM009      0.5
386  RDREFTIM010      0.5
387  RDREFTIM011      0.5
388  RDREFTIM012      0.5
389  RDREFTIM013      0.5
390  RDREFTIM014      0.5
391  RDREFTIM015      0.5
392  RDREFTIM016      0.5
393  RDREFTIM017      0.5
394  RDREFTIM018      0.5
395  RDREFTIM019      0.5
396  RDREFTIM020      0.5
397  RDREFTIM021      0.5
398  RDREFTIM022      0.5
399  RDREFTIM023      0.5
400  RDREFTIM024      0.5
401  RDREFTIM025      0.5
402  RDREFTIM026      0.5
403  RDREFTIM027      0.5
404  RDREFTIM028      0.5
405  RDREFTIM029      0.5
406  RDREFTIM030      0.5
407  RDREFTIM031      0.5
408  RDREFTIM032      0.5
409  RDREFTIM033      0.5
410  RDREFTIM034      0.5
411  RDREFTIM035      0.5
412  RDREFTIM036      0.5
```

413 RDREFTIM037 0.5
414 RDREFTIM038 0.5
415 RDREFTIM039 0.5
416 RDREFTIM040 0.5
417 RDREFTIM041 0.5
418 RDREFTIM042 0.5
419 RDREFTIM043 0.5
420 RDREFTIM044 0.5
421 RDREFTIM045 0.5
422 RDREFTIM046 0.5
423 RDREFTIM047 0.5
424 RDREFTIM048 0.5
425 RDREFTIM049 0.5
426 RDREFTIM050 0.5
427 RDREFTIM051 0.5
428 RDREFTIM052 0.5
429 RDREFTIM053 0.5
430 RDREFTIM054 0.5
431 RDREFTIM055 0.5
432 RDREFTIM056 0.5
433 RDREFTIM057 0.5
434 RDREFTIM058 0.5
435 RDREFTIM059 0.5
436 RDREFTIM060 0.5
437 RDREFTIM061 0.5
438 RDREFTIM062 0.5
439 RDREFTIM063 0.5
440 RDREFTIM064 0.5
*
* PLM_DEN, plume rise model density
441 RDPLMMOD001 DENSITY
*
* Form 'Density and Flow' Comment:
* Come in thru MELMACCS.
*
* PLMFLO, Heat by Density
442 RDPLMFLA001 0.26934
443 RDPLMFLA002 0.27201
444 RDPLMFLA003 0.27467
445 RDPLMFLA004 0.27607
446 RDPLMFLA005 0.2761
447 RDPLMFLA006 0.276
448 RDPLMFLA007 0.26804
449 RDPLMFLA008 0.266
450 RDPLMFLA009 0.27937
451 RDPLMFLA010 0.28892
452 RDPLMFLA011 0.29341
453 RDPLMFLA012 0.29701
454 RDPLMFLA013 0.29941
455 RDPLMFLA014 45.064
456 RDPLMFLA015 90.862
457 RDPLMFLA016 92.481
458 RDPLMFLA017 94.051
459 RDPLMFLA018 95.884
460 RDPLMFLA019 97.289
461 RDPLMFLA020 98.627
462 RDPLMFLA021 98.143
463 RDPLMFLA022 90.445
464 RDPLMFLA023 82.861
465 RDPLMFLA024 80.667
466 RDPLMFLA025 78.449
467 RDPLMFLA026 76.815
468 RDPLMFLA027 76.664
469 RDPLMFLA028 76.743
470 RDPLMFLA029 76.368
471 RDPLMFLA030 77.172
472 RDPLMFLA031 77.974
473 RDPLMFLA032 77.411
474 RDPLMFLA033 76.839
475 RDPLMFLA034 76.701
476 RDPLMFLA035 76.561
477 RDPLMFLA036 76.303
478 RDPLMFLA037 76.318
479 RDPLMFLA038 76.353
480 RDPLMFLA039 76.373
481 RDPLMFLA040 76.734
482 RDPLMFLA041 76.992
483 RDPLMFLA042 77.125
484 RDPLMFLA043 77.385
485 RDPLMFLA044 77.651
486 RDPLMFLA045 78.071
487 RDPLMFLA046 78.381
488 RDPLMFLA047 78.453
489 RDPLMFLA048 78.613
490 RDPLMFLA049 78.754
491 RDPLMFLA050 78.792
492 RDPLMFLA051 78.735
493 RDPLMFLA052 78.735
494 RDPLMFLA053 78.736
495 RDPLMFLA054 78.697
496 RDPLMFLA055 78.808
497 RDPLMFLA056 79.027
498 RDPLMFLA057 79.514
499 RDPLMFLA058 80.406
500 RDPLMFLA059 80.858
501 RDPLMFLA060 80.593
502 RDPLMFLA061 80.335
503 RDPLMFLA062 80.175
504 RDPLMFLA063 80.055
505 RDPLMFLA064 79.96
*
* PLMDEN, Heat by Density
506 RDPLMDEN001 0.86021
507 RDPLMDEN002 0.84812
508 RDPLMDEN003 0.83644
509 RDPLMDEN004 0.8267
510 RDPLMDEN005 0.81841
511 RDPLMDEN006 0.81028
512 RDPLMDEN007 0.8031
513 RDPLMDEN008 0.79446

```
514  RDPLMDEN009      0.77807
515  RDPLMDEN010      0.75575
516  RDPLMDEN011      0.73136
517  RDPLMDEN012      0.70656
518  RDPLMDEN013      0.68146
519  RDPLMDEN014      0.79029
520  RDPLMDEN015      0.91733
521  RDPLMDEN016      0.91733
522  RDPLMDEN017  0.90857
523  RDPLMDEN018      0.90024
524  RDPLMDEN019      0.89041
525  RDPLMDEN020      0.88222
526  RDPLMDEN021      0.87581
527  RDPLMDEN022      0.91738
528  RDPLMDEN023      0.96826
529  RDPLMDEN024      0.97899
530  RDPLMDEN025      0.98934
531  RDPLMDEN026      0.99724
532  RDPLMDEN027      0.99861
533  RDPLMDEN028      0.99808
534  RDPLMDEN029      0.99979
535  RDPLMDEN030      0.99631
536  RDPLMDEN031      0.99307
537  RDPLMDEN032      0.99563
538  RDPLMDEN033      0.99769
539  RDPLMDEN034      0.99888
540  RDPLMDEN035      0.99952
541  RDPLMDEN036      1.
542  RDPLMDEN037      1.0005
543  RDPLMDEN038      1.0004
544  RDPLMDEN039      0.99985
545  RDPLMDEN040      0.99884
546  RDPLMDEN041      0.99771
547  RDPLMDEN042      0.99658
548  RDPLMDEN043      0.99541
549  RDPLMDEN044      0.99434
550  RDPLMDEN045      0.99298
551  RDPLMDEN046      0.99159
552  RDPLMDEN047      0.99086
553  RDPLMDEN048      0.9905
554  RDPLMDEN049      0.99022
555  RDPLMDEN050      0.99003
556  RDPLMDEN051      0.98994
557  RDPLMDEN052      0.98995
558  RDPLMDEN053      0.98994
559  RDPLMDEN054      0.98974
560  RDPLMDEN055      0.98942
561  RDPLMDEN056      0.9886
562  RDPLMDEN057      0.98646
563  RDPLMDEN058      0.98238
564  RDPLMDEN059      0.98012
565  RDPLMDEN060      0.9813
566  RDPLMDEN061      0.98255
567  RDPLMDEN062      0.9833
568  RDPLMDEN063      0.98386
569  RDPLMDEN064      0.9843
   *
   * BRGSMD, Briggs plume rise model
570  RDBRGSMD001      IMPROVED
   *
   * PLHITE, height of each plume segment at release (meters)
571  RDPLHITE001      50.
572  RDPLHITE002      50.
573  RDPLHITE003      50.
574  RDPLHITE004      50.
575  RDPLHITE005      50.
576  RDPLHITE006      50.
577  RDPLHITE007      50.
578  RDPLHITE008      50.
579  RDPLHITE009      50.
580  RDPLHITE010      50.
581  RDPLHITE011      50.
582  RDPLHITE012      50.
583  RDPLHITE013      50.
584  RDPLHITE014      50.
585  RDPLHITE015      50.
586  RDPLHITE016      50.
587  RDPLHITE017      50.
588  RDPLHITE018      50.
589  RDPLHITE019      50.
590  RDPLHITE020      50.
591  RDPLHITE021      50.
592  RDPLHITE022      50.
593  RDPLHITE023      50.
594  RDPLHITE024      50.
595  RDPLHITE025      50.
596  RDPLHITE026      50.
597  RDPLHITE027      50.
598  RDPLHITE028      50.
599  RDPLHITE029      50.
600  RDPLHITE030      50.
601  RDPLHITE031      50.
602  RDPLHITE032      50.
603  RDPLHITE033      50.
604  RDPLHITE034      50.
605  RDPLHITE035      50.
606  RDPLHITE036      50.
607  RDPLHITE037      50.
608  RDPLHITE038      50.
609  RDPLHITE039      50.
610  RDPLHITE040      50.
611  RDPLHITE041      50.
612  RDPLHITE042      50.
613  RDPLHITE043      50.
614  RDPLHITE044      50.
615  RDPLHITE045      50.
616  RDPLHITE046      50.
617  RDPLHITE047      50.
618  RDPLHITE048      50.
619  RDPLHITE049      50.
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620 RDPLHITE050 50.
621 RDPLHITE051 50.
622 RDPLHITE052 50.
623 RDPLHITE053 50.
624 RDPLHITE054 50.
625 RDPLHITE055 50.
626 RDPLHITE056 50.
627 RDPLHITE057 50.
628 RDPLHITE058 50.
629 RDPLHITE059 50.
630 RDPLHITE060 50.
631 RDPLHITE061 50.
632 RDPLHITE062 50.
633 RDPLHITE063 50.
634 RDPLHITE064 50.
*
* PLUDUR, duration of each plume segment (sec)
635 RDPLUDUR001 1440.
636 RDPLUDUR002 1800.
637 RDPLUDUR003 1800.
638 RDPLUDUR004 1800.
639 RDPLUDUR005 1800.
640 RDPLUDUR006 1800.
641 RDPLUDUR007 1800.
642 RDPLUDUR008 1800.
643 RDPLUDUR009 1800.
644 RDPLUDUR010 1800.
645 RDPLUDUR011 1800.
646 RDPLUDUR012 1800.
647 RDPLUDUR013 1800.
648 RDPLUDUR014 1800.
649 RDPLUDUR015 900.
650 RDPLUDUR016 900.
651 RDPLUDUR017 1800.
652 RDPLUDUR018 1800.
653 RDPLUDUR019 1800.
654 RDPLUDUR020 1800.
655 RDPLUDUR021 1800.
656 RDPLUDUR022 1800.
657 RDPLUDUR023 1800.
658 RDPLUDUR024 1800.
659 RDPLUDUR025 1800.
660 RDPLUDUR026 1800.
661 RDPLUDUR027 1800.
662 RDPLUDUR028 1800.
663 RDPLUDUR029 1800.
664 RDPLUDUR030 1800.
665 RDPLUDUR031 1800.
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670 RDPLUDUR036 1800.
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674 RDPLUDUR040 1800.
675 RDPLUDUR041 1800.
676 RDPLUDUR042 1800.
677 RDPLUDUR043 1800.
678 RDPLUDUR044 1800.
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680 RDPLUDUR046 1800.
681 RDPLUDUR047 1800.
682 RDPLUDUR048 1800.
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692 RDPLUDUR058 1800.
693 RDPLUDUR059 1800.
694 RDPLUDUR060 1800.
695 RDPLUDUR061 1800.
696 RDPLUDUR062 1800.
697 RDPLUDUR063 1800.
698 RDPLUDUR064 1800.
*
* PDELAY, time of release for each plume from xxxx (sec)
699 RDPDELAY001 1.46160E+05
700 RDPDELAY002 1.47600E+05
701 RDPDELAY003 1.49400E+05
702 RDPDELAY004 1.51200E+05
703 RDPDELAY005 1.53000E+05
704 RDPDELAY006 1.54800E+05
705 RDPDELAY007 1.56600E+05
706 RDPDELAY008 1.58400E+05
707 RDPDELAY009 1.60200E+05
708 RDPDELAY010 1.62000E+05
709 RDPDELAY011 1.63800E+05
710 RDPDELAY012 1.65600E+05
711 RDPDELAY013 1.67400E+05
712 RDPDELAY014 1.69200E+05
713 RDPDELAY015 1.71000E+05
714 RDPDELAY016 1.71900E+05
715 RDPDELAY017 1.72800E+05
716 RDPDELAY018 1.74600E+05
717 RDPDELAY019 1.76400E+05
718 RDPDELAY020 1.78200E+05
719 RDPDELAY021 1.80000E+05
720 RDPDELAY022 1.81800E+05
721 RDPDELAY023 1.83600E+05
722 RDPDELAY024 1.85400E+05
723 RDPDELAY025 1.87200E+05
724 RDPDELAY026 1.89000E+05
725 RDPDELAY027 1.90800E+05

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726  RDPDELAY028     1.92600E+05
727  RDPDELAY029     1.94400E+05
728  RDPDELAY030     1.96200E+05
729  RDPDELAY031     1.98000E+05
730  RDPDELAY032     1.99800E+05
731  RDPDELAY033     2.01600E+05
732  RDPDELAY034     2.03400E+05
733  RDPDELAY035     2.05200E+05
734  RDPDELAY036     2.07000E+05
735  RDPDELAY037     2.08800E+05
736  RDPDELAY038     2.10600E+05
737  RDPDELAY039     2.12400E+05
738  RDPDELAY040     2.14200E+05
739  RDPDELAY041     2.16000E+05
740  RDPDELAY042     2.17800E+05
741  RDPDELAY043     2.19600E+05
742  RDPDELAY044     2.21400E+05
743  RDPDELAY045     2.23200E+05
744  RDPDELAY046     2.25000E+05
745  RDPDELAY047     2.26800E+05
746  RDPDELAY048     2.28600E+05
747  RDPDELAY049     2.30400E+05
748  RDPDELAY050     2.32200E+05
749  RDPDELAY051     2.34000E+05
750  RDPDELAY052     2.35800E+05
751  RDPDELAY053     2.37600E+05
752  RDPDELAY054     2.39400E+05
753  RDPDELAY055     2.41200E+05
754  RDPDELAY056     2.43000E+05
755  RDPDELAY057     2.44800E+05
756  RDPDELAY058     2.46600E+05
757  RDPDELAY059     2.48400E+05
758  RDPDELAY060     2.50200E+05
759  RDPDELAY061     2.52000E+05
760  RDPDELAY062     2.53800E+05
761  RDPDELAY063     2.55600E+05
762  RDPDELAY064     2.57400E+05
     *
     * Form 'Particle Size Distribution' Comment:
     * Particle size distribution from MELMACCS.
     *
     * PSDIST, particle size distribution of each element group
763  RDPSDIST001     0.1     0.1     0.1     0.1     0.1     0.1     0.1     0.1     0.1     0.1
764  RDPSDIST002     0.024772        0.12138 0.26188 0.22205 0.11227 0.056887        0.028087        0.0099396       0.0026936       0.16004
765  RDPSDIST003     0.013704        0.066983        0.11612 0.11077 0.24282 0.26825 0.11485 0.025628        0.0026261       0.038253
766  RDPSDIST004     0.027282        0.13312 0.27279 0.21562 0.10619 0.045158        0.02129 0.0080353       0.0022414       0.16827
767  RDPSDIST005     0.027181        0.13224 0.27113 0.21463 0.10772 0.047025        0.021942        0.0081148       0.0022255       0.16779
768  RDPSDIST006     0.032448        0.15413 0.2882  0.20783 0.09458 0.031123        0.013884        0.0059096       0.0017136       0.17018
769  RDPSDIST007     0.025258        0.12406 0.27018 0.22952 0.10476 0.044551        0.023143        0.0090954       0.0027297       0.1667
770  RDPSDIST008     0.032214        0.15296 0.28591 0.21506 0.10068 0.033778        0.01521 0.0063688       0.0018231       0.156
771  RDPSDIST009     0.032168        0.15289 0.2863  0.21395 0.10008 0.03312 0.014638        0.0061232       0.0017501       0.15899
     *
     * CORINV, inventory of each radionuclide present in the facility at the time of accident initiation (becquerels)
772  RDCORINV001     Kr-85   1.37E+17
773  RDCORINV002     Kr-85m  0.
774  RDCORINV003     Kr-87   0.
775  RDCORINV004     Kr-88   0.
776  RDCORINV005     Xe-133  2.51E+16
777  RDCORINV006     Xe-135  4.2E-11
778  RDCORINV007     Xe-135m 0.
779  RDCORINV008     Cs-134  7.8E+17
780  RDCORINV009     Cs-136  1.82E+16
781  RDCORINV010     Cs-137  2.17E+18
782  RDCORINV011     Rb-86   1.01E+15
783  RDCORINV012     Rb-88   0.
784  RDCORINV013     Ba-139  0.
785  RDCORINV014     Ba-140  2.64E+17
786  RDCORINV015     Sr-89   5.5E+17
787  RDCORINV016     Sr-90   1.54E+18
788  RDCORINV017     Sr-91   9.93E-11
789  RDCORINV018     Sr-92   0.
790  RDCORINV019     Ba-137m 2.08E+18
791  RDCORINV020     I-131   5.27E+16
792  RDCORINV021     I-132   6.8E+14
793  RDCORINV022     I-133   3.54000E+05
794  RDCORINV023     I-134   0.
795  RDCORINV024     I-135   0.
796  RDCORINV025     Te-127  1.68E+16
797  RDCORINV026     Te-127m 1.7E+16
798  RDCORINV027     Te-129  2.01E+16
799  RDCORINV028     Te-129m 3.15E+16
800  RDCORINV029     Te-131m 3.05E+08
801  RDCORINV030     Te-132  6.61E+14
802  RDCORINV031     Te-131  6.88E+07
803  RDCORINV032     Rh-105  4.81E+10
804  RDCORINV033     Ru-103  1.15E+18
805  RDCORINV034     Ru-105  0.
806  RDCORINV035     Ru-106  1.17E+18
807  RDCORINV036     Rh-103m 1.14E+18
808  RDCORINV037     Rh-106  1.17E+18
809  RDCORINV038     Nb-95   1.72E+18
810  RDCORINV039     Co-58   2.05E+14
811  RDCORINV040     Co-60   3.E+15
812  RDCORINV041     Mo-99   1.96E+14
813  RDCORINV042     Tc-99m  1.91E+14
814  RDCORINV043     Nb-97   327.
815  RDCORINV044     Nb-97m  288.
816  RDCORINV045     Ce-141  9.44E+17
817  RDCORINV046     Ce-143  1.49E+10
818  RDCORINV047     Ce-144  1.8E+18
819  RDCORINV048     Np-239  1.13E+15
820  RDCORINV049     Pu-238  6.7E+16
821  RDCORINV050     Pu-239  5.29E+15
822  RDCORINV051     Pu-240  1.08E+16
823  RDCORINV052     Pu-241  1.51E+18
824  RDCORINV053     Zr-95   1.35E+18
825  RDCORINV054     Zr-97   322.
826  RDCORINV055     Am-241  2.86E+16
827  RDCORINV056     Cm-242  1.25E+17
828  RDCORINV057     Cm-244  6.6E+16
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829  RDCORINV058      La-140  3.38E+17
830  RDCORINV059      La-141  0.
831  RDCORINV060      La-142  0.
832  RDCORINV061      Nd-147  8.15E+16
833  RDCORINV062      Pr-143  3.11E+17
834  RDCORINV063      Y-90    1.4E+18
835  RDCORINV064      Y-91    8.89E+17
836  RDCORINV065      Y-92    0.
837  RDCORINV066      Y-93    5.89E-09
838  RDCORINV067      Y-91m   7.E-11
839  RDCORINV068      Pr-144  1.81E+18
840  RDCORINV069      Pr-144m 2.54E+16
     *
     * Form 'Inventory Scale Factor' Comment:
     * Set by MELMACCS.
     *
     * CORSCA, scaling factor to adjust the core inventory
841  RDCORSCA001      1.0
     *
     * APLFRC, Specifies how release fractions are applied to daughter ingrowth products
842  RDAPLFRC001    PARENT
     *
     * GRPNAM, user assigned name of each chemical group.  May have been imported from MelMACCS
     *ISGRPNAM001    Xe
     *ISGRPNAM002    Cs
     *ISGRPNAM003    Ba
     *ISGRPNAM004    I
     *ISGRPNAM005    Te
     *ISGRPNAM006    Ru
     *ISGRPNAM007    Mo
     *ISGRPNAM008    Ce
     *ISGRPNAM009    La
     *
     * Form 'Release Fractions' Comment:
     * These values come from MELCOR PTF file.  Plume discretization is done by user.  MACCS2 Radionuclide Inventory will account for the correct release magnitude on a isotope-by-isotope basis .
     *
     * RELFRC, release fractions for each of the plume segments for each chemical group
843  RDRELFRC001      7.5961E-05      5.2466E-05      1.3308E-05      2.207E-04       2.2286E-04      1.4088E-09      1.5567E-07      3.3224E-12      3.1904E-12
844  RDRELFRC002      7.9235E-05      5.9335E-05      1.4929E-05      2.4366E-04      2.5097E-04      7.4808E-09      8.2596E-07      4.3421E-12      4.1695E-12
845  RDRELFRC003      8.9016E-05      6.6622E-05      1.6567E-05      2.7027E-04      2.8053E-04      1.6716E-08      1.8452E-06      6.8548E-12      6.5824E-12
846  RDRELFRC004      3.1377E-04      2.7595E-04      5.7815E-05      1.7603E-04      1.8268E-04      2.3905E-08      2.634E-06       8.6614E-12      8.3186E-12
847  RDRELFRC005      1.4321E-04      8.4921E-05      1.845E-05       1.4068E-04      1.4472E-04      3.2621E-08      5.0471E-06      2.1956E-11      2.1199E-11
848  RDRELFRC006      1.5396E-04      7.6259E-05      1.4436E-05      1.4628E-04      1.492E-04       2.8201E-08      9.0064E-06      3.4484E-11      3.3666E-11
849  RDRELFRC007      0.0010715       7.0659E-04      1.6201E-04      0.0013916       0.0014231       4.6228E-08      1.5943E-05      2.5914E-11      2.5007E-11
850  RDRELFRC008      3.3774E-04      1.5054E-04      2.8216E-05      3.111E-04       3.0933E-04      3.0165E-08      2.8771E-05      3.3034E-11      3.1995E-11
851  RDRELFRC009      4.7978E-04      1.7831E-04      3.4121E-05      3.4744E-04      3.4782E-04      6.1965E-08      4.4951E-05      2.982E-12       2.986E-12
852  RDRELFRC010      5.9111E-04      1.6149E-04      2.8065E-05      3.066E-04       3.0754E-04      7.3842E-08      5.0126E-05      6.72E-13        6.46E-13
853  RDRELFRC011      7.3827E-04      2.8872E-04      2.9673E-05      5.3545E-04      5.2067E-04      7.9815E-08      6.9166E-05      8.092E-12       7.836E-12
854  RDRELFRC012      6.7634E-04      2.4744E-04      2.5622E-05      4.6027E-04      4.6357E-04      7.0409E-08      6.4508E-05      5.784E-12 5.595E-12
855  RDRELFRC013      0.0014218       7.601E-04       1.005E-04       2.4836E-04      2.6389E-04      8.1907E-08      7.0973E-05      5.042E-12       4.876E-12
856  RDRELFRC014      0.05027 0.021969        0.0036026       0.021924        0.022196        3.4581E-06      0.0022792       2.725E-10       2.6462E-10
857  RDRELFRC015      0.0034485       0.0025822       1.16E-04        0.0025196       0.0025587       8.0955E-07      8.3555E-04      1.1724E-10      1.1521E-10
858  RDRELFRC016      0.0052627       0.0045705       1.351E-05       0.0030243       0.0030605       6.7193E-07      0.0012451       2.0381E-10      2.0377E-10
859  RDRELFRC017      0.0077725       0.0081142       1.248E-05       0.0082127       0.0083206       8.576E-06       0.0033295       5.0613E-10      4.9585E-10
860  RDRELFRC018      0.012392        0.011946        1.501E-05       0.012337        0.012485        2.6051E-05      0.004813        8.2471E-10      8.084E-10
861  RDRELFRC019      0.011118        0.010465        1.477E-05       0.010191        0.010322        2.4924E-05      0.003787        7.2951E-10      7.1753E-10
862  RDRELFRC020      0.016103        0.014598        1.974E-05       0.012786        0.012869        3.581E-05       0.0044616       9.9006E-10      9.8254E-10
863  RDRELFRC021      0.016262        0.015053        1.732E-05       0.014435        0.014458        4.8393E-05      0.0048829       1.176E-09       1.1636E-09
864  RDRELFRC022      0.017191        0.018057        1.491E-05       0.013473        0.013165        5.1549E-05      0.0054123       1.029E-09       1.0223E-09
865  RDRELFRC023      0.014375        0.007644        1.004E-05       0.015353        0.015104        3.6754E-05      0.0043098       6.532E-10       6.1865E-10
866  RDRELFRC024      0.01778 0.00906 1.493E-05       0.017358        0.019943        4.1636E-05      0.0042059       1.0167E-09      1.2388E-09
867  RDRELFRC025      0.015195        0.006967        1.414E-05       0.014454        0.015447        2.5759E-05      0.0024347       1.0674E-09      1.0427E-09
868  RDRELFRC026      0.014964        0.008646        2.053E-05       0.011098        0.011025        1.8312E-05      0.0016556       7.1398E-10      7.4999E-10
869  RDRELFRC027      0.009121        0.006792        2.268E-05       0.005822        0.006317        1.5325E-05      0.0013118       5.3002E-10      5.1829E-10
870  RDRELFRC028      0.008736        0.007426        2.502E-05       0.005313        0.006137        1.7194E-05      0.0014796       5.5367E-10      5.559E-10
871  RDRELFRC029      0.00759 0.006755        1.003E-05       0.00491 0.005493        1.6491E-05      0.0013529       4.782E-10       5.36E-10
872  RDRELFRC030      0.007569        0.006938        9.36E-06        0.006209        0.006733        2.0328E-05      0.0016935       5.082E-10       5.815E-10
873  RDRELFRC031      0.006937        0.006289        1.63E-05        0.005473        0.005498        1.7407E-05      0.0014406       3.819E-10       4.303E-10
874  RDRELFRC032      0.009558        0.008695        3.192E-05       0.007119        0.00712 2.2749E-05      0.0018651       4.512E-10       5.566E-10
875  RDRELFRC033      0.009448        0.008436        4.405E-05       0.006502        0.006794        2.2937E-05      0.0017504       4.052E-10       5.283E-10
876  RDRELFRC034      0.011707        0.010366        2.78E-05        0.008208        0.008395        2.9231E-05      0.0021355       5.447E-10       6.805E-10
877  RDRELFRC035      0.011125        0.009595        1.561E-05       0.007678        0.007858        2.8565E-05      0.0019645       4.595E-10       5.835E-10
878  RDRELFRC036      0.013175        0.011244        1.641E-05       0.009459        0.009455        3.6449E-05      0.0023808       5.47E-10        7.005E-10
879  RDRELFRC037      0.011785        0.009822        1.509E-05       0.00859 0.008582        3.5306E-05      0.0021216       4.902E-10       6.444E-10
880  RDRELFRC038      0.01378 0.011095        2.107E-05       0.010217        0.010215        4.422E-05       0.0024746       5.972E-10       7.807E-10
881  RDRELFRC039      0.012299        0.009601        2.179E-05       0.00927 0.009273        4.2379E-05      0.002204        5.594E-10       7.132E-10
882  RDRELFRC040      0.014418        0.010962        2.416E-05       0.01097 0.010977        5.2931E-05      0.0025582       7.092E-10       8.913E-10
883  RDRELFRC041      0.012566        0.009286        2.135E-05       0.009856        0.009869        4.9851E-05      0.002252        7.429E-10       9.099E-10
884  RDRELFRC042      0.01439 0.010379        1.962E-05       0.011646        0.011667        6.0294E-05      0.0026006       1.046E-09       1.2052E-09
885  RDRELFRC043      0.012397        0.008652        1.652E-05       0.009874        0.009898        5.5079E-05      0.0021933       9.599E-10       1.0835E-09
886  RDRELFRC044      0.014089        0.009398        1.937E-05       0.011343        0.011377        6.5656E-05      0.0024312       1.1665E-09      1.3023E-09
887  RDRELFRC045      0.012367        0.007869        1.744E-05       0.010006        0.010043        5.9679E-05      0.0020797       1.0773E-09      1.1988E-09
888  RDRELFRC046      0.014007        0.008549        2.037E-05       0.011518        0.011567        7.027E-05       0.0023233       1.2749E-09      1.4201E-09
889  RDRELFRC047      0.012171        0.007139        1.816E-05       0.009888        0.009933        6.23E-05        0.0019492       1.1079E-09      1.2487E-09
890  RDRELFRC048      0.013823        0.007712        2.114E-05       0.011172        0.011213        7.247E-05       0.0021548       1.0109E-09      1.1138E-09
891  RDRELFRC049      0.011929        0.006428        1.891E-05       0.009783        0.009827        6.683E-05       0.0018327       9.222E-10       1.0824E-09
892  RDRELFRC050      0.012229        0.006949        2.096E-05       0.011097        0.011195        7.989E-05       0.002045        1.6255E-09      1.7289E-09
893  RDRELFRC051      0.011071        0.005815        2.055E-05       0.010094        0.009966        7.294E-05       0.0017712       1.7567E-09      1.7644E-09
894  RDRELFRC052      0.044679        0.00686 6.727E-05       0.016235        0.014031        8.813E-05       0.0020563       2.8212E-09      2.5829E-09
895  RDRELFRC053      0.012261        0.005631        2.479E-05       0.01002 0.009979        7.717E-05       0.0017293       1.7075E-09      1.7654E-09
896  RDRELFRC054      0.011436        0.006075        2.716E-05       0.010772        0.010816        8.977E-05       0.001821        1.8741E-09      1.9624E-09
897  RDRELFRC055      0.009807        0.005073        2.441E-05       0.009107        0.00915 7.954E-05       0.0015011       1.5756E-09      1.6708E-09
898  RDRELFRC056      0.011181        0.005741        2.969E-05       0.01041 0.010462        9.295E-05       0.001689        1.7357E-09      1.8843E-09
899  RDRELFRC057      0.009886        0.005137        2.787E-05       0.009459        0.009496        8.454E-05       0.001541        1.5488E-09      1.633E-09
900  RDRELFRC058      0.011414        0.005913        3.39E-05        0.010744        0.010807        9.792E-05       0.00168 2.5148E-09      2.5109E-09
901  RDRELFRC059      0.009589        0.004729        3.088E-05       0.008955        0.009044        8.335E-05       0.001296        2.4482E-09      2.3894E-09
902  RDRELFRC060      0.010243        0.004838        3.541E-05       0.009385        0.009488        9.383E-05       0.001304        2.394E-09       2.4291E-09
903  RDRELFRC061      0.008446        0.003916        3.141E-05       0.007804        0.007891        8.008E-05       0.001066        2.0536E-09      2.0931E-09
904  RDRELFRC062      0.009273        0.004214        3.679E-05       0.00874 0.008835        8.86E-05        0.001161        2.2698E-09      2.3402E-09
905  RDRELFRC063      0.007795        0.003498        3.285E-05       0.007557        0.007635        7.603E-05       9.79E-04        1.8496E-09      1.9578E-09
906  RDRELFRC064      0.00869 0.003803        3.847E-05       0.008764        0.008813        8.785E-05       0.001096        1.9805E-09      2.1672E-09
     *
     * ENDAT1, flag indicating whether only atmos is run
907  OCENDAT1001      .FALSE.
     *
     * IDEBUG, specifies set of debug results to report
908  OCIDEBUG001      0
     *
     * NUCOUT, name of the nuclide to be listed on the dispersion listings
909  OCNUCOUT001      Cs-137
```

```
*
* METCOD, meteorological sampling option code
910  M1METCOD001      2
*
* Form 'Boundary Limit' Comment:
* From NUREG-1150.
*
* LIMSPA, last spatial interval for measured weather
911  M2LIMSPA001      25
*
* Form 'Constant or Boundary Conditions' Comment:
* Stability class 5 is the most prevalent in the PB data.  2.2 is average speed data, and other values are from NUREG-1150 data.
*
* BNDMXH, boundary weather mixing layer height (meters)
912  M2BNDMXH001      1000.
*
* IBDSTB, boundary weather stability class index
913  M2IBDSTB001      4
*
* BNDRAN, boundary weather rain rate (mm/hr)
914  M2BNDRAN001      5.
*
* BNDWND, boundary weather wind speed (m/sec)
915  M2BNDWND001      2.2
*
* MAXHGT, if equal DAY_AND_NIGHT, then time of sunrise/sunset is used to calculate max mixing height. DAY_ONLY uses MACCS2 1.12 model
916  M1MAXHGT001      DAY_AND_NIGHT
*
* Form 'Site Location' Comment:
* Consistent with PB site file.
*
* LATITUDE_DEG, LATITUDE_MIN, LATITUDE_SEC, indicates latitude of site, used with longitude
917  M1LATITU001      39.
*
* LATITU_MIN minutes portion of latitude
918  M1LATITU002      45.
*
* LATITU_SEC, seconds portion of latitude
919  M1LATITU003      32.
*
* LONGIT_DEG, LONGIT_MIN, LONGIT_SEC, indicates longitude of site, used with latitude
920  M1LONGIT001      76.
*
* LONGIT_MIN, minutes portion of longitude
921  M1LONGIT002      16.
*
* LONGIT_SEC, seconds portion of longitude
922  M1LONGIT003      9.
*
* Form 'Rain Distances' Comment:
* From NUREG-1150.
*
* NRNINT, number of rain distance intervals for binning
923  M4NRNINT001      5
*
* RNDSTS, endpoints of the rain distance intervals (km)
924  M4RNDSTS001      3.22
925  M4RNDSTS002      5.63
926  M4RNDSTS003      11.27
927  M4RNDSTS004      20.92
928  M4RNDSTS005      32.19
*
* Form 'Rain Intensities' Comment:
* From NUREG-1150.
*
* NRINTN, number of rain intensity breakpoints
929  M4NRINTN001      3
*
* RNRATE, rain intensity breakpoints for weather binning (mm/hr)
930  M4RNRATE001      2.
931  M4RNRATE002      4.
932  M4RNRATE003      6.
*
* IRSEED, initial seed for random number generator
933  M4IRSEED001      79
*
* Form 'Bins' Comment:
* Minimum of 12 or 10% of samples in bin.
*
* NSBINS, number of bins to be sampled when NSMPLS = 0
934  M4NSBINS001      36
*
* INDXBN, INWGHT, number of weather sequences to be selected from specific weather bins
935  M4SMPLDF001      1      71
936  M4SMPLDF002      2      42
937  M4SMPLDF003      3      12
938  M4SMPLDF004      4      52
939  M4SMPLDF005      5      57
940  M4SMPLDF006      6      74
941  M4SMPLDF007      7      21
942  M4SMPLDF008      8      12
943  M4SMPLDF009      9      49
944  M4SMPLDF010      10     103
945  M4SMPLDF011      11     77
946  M4SMPLDF012      12     35
947  M4SMPLDF013      13     51
948  M4SMPLDF014      14     75
949  M4SMPLDF015      15     14
950  M4SMPLDF016      16     4
951  M4SMPLDF017      17     44
952  M4SMPLDF018      18     12
953  M4SMPLDF019      19     17
954  M4SMPLDF020      20     24
955  M4SMPLDF021      21     24
956  M4SMPLDF022      22     12
957  M4SMPLDF023      23     4
958  M4SMPLDF024      24     8
959  M4SMPLDF025      25     12
960  M4SMPLDF026      26     12
961  M4SMPLDF027      27     12
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 962  M4SMPLDF028      28      1
 963  M4SMPLDF029      29      3
 964  M4SMPLDF030      30      5
 965  M4SMPLDF031      31      4
 966  M4SMPLDF032      32      12
 967  M4SMPLDF033      33      1
 968  M4SMPLDF034      34      7
 969  M4SMPLDF035      35      9
 970  M4SMPLDF036      36      12
      *
      * ATMOS_ZERO = 0
 971  TYPE0NUMBER      0
      *
      * NUM0, number of results
 972  TYPE0NUMBER      14
********   RECORD NUMBER   972 REPLACES RECORD NUMBER   971   ********
      *
      * INDREL, INDRAD, CCDF, ATMOS release and spatial interval
 973  TYPE0OUT001     1      1      NONE
 974  TYPE0OUT002     1      2      NONE
 975  TYPE0OUT003     1      3      NONE
 976  TYPE0OUT004     1      4      NONE
 977  TYPE0OUT005     1      5      NONE
 978  TYPE0OUT006     1      6      NONE
 979  TYPE0OUT007     1      7      NONE
 980  TYPE0OUT008     1      8      NONE
 981  TYPE0OUT009     1      9      NONE
 982  TYPE0OUT010     1      10     NONE
 983  TYPE0OUT011     1      11     NONE
 984  TYPE0OUT012     1      12     NONE
 985  TYPE0OUT013     1      19     NONE
 986  TYPE0OUT014     1      21     NONE
      *
      * NUM_DIST2, used for Dispersion Power Law (always 0)
 987  NUM_DIST001      0
      *
      * NSMPLS2, used for non-uniform Bin Sampling (always 0)
 988  M4NSMPLS001      0
      .
******** TERMINATOR RECORD ENCOUNTERED -- END OF BASE CASE USER INPUT ********


USER INPUT PROCESSING SUMMARY - BASE CASE

NUMBER OF RECORDS READ                          = 1229
NUMBER OF BLANK OR COMMENT RECORDS READ         =  240
NUMBER OF TERMINATOR RECORDS                    =    1
NUMBER OF RECORDS PROCESSED                     =  988
    NUMBER OF PROCESSED RECORDS DUPLICATED      =    2
    NUMBER OF PROCESSED RECORDS SORTED          =  986
****************************************************************************************************



Decay Chain #  Ba-139

Decay Chain #  Ba-140   La-140
Fraction of Ba-140  going to La-140  in this chain = 1.000000

Decay Chain #  Ce-143   Pr-143
Fraction of Ce-143  going to Pr-143  in this chain = 1.000000

Decay Chain #  Ce-144   Pr-144
Fraction of Ce-144  going to Pr-144  in this chain = 0.982200

Decay Chain #  Ce-144   Pr-144m  Pr-144
Fraction of Ce-144  going to Pr-144m in this chain = 0.017800
Fraction of Ce-144  going to Pr-144  in this chain = 0.017782
Fraction of Pr-144m going to Pr-144  in this chain = 0.999000

Decay Chain #  Cm-242   Pu-238
Fraction of Cm-242  going to Pu-238  in this chain = 1.000000

Decay Chain #  Cm-244   Pu-240
Fraction of Cm-244  going to Pu-240  in this chain = 1.000000

Decay Chain #  Co-58

Decay Chain #  Co-60

Decay Chain #  Cs-134

Decay Chain #  Cs-136

Decay Chain #  Cs-137   Ba-137m
Fraction of Cs-137  going to Ba-137m in this chain = 0.946000

Decay Chain #  I-133    Xe-133
Fraction of I-133   going to Xe-133  in this chain = 0.971000

Decay Chain #  I-134

Decay Chain #  I-135    Xe-135
Fraction of I-135   going to Xe-135  in this chain = 0.846000

Decay Chain #  I-135    Xe-135m  Xe-135
Fraction of I-135   going to Xe-135m in this chain = 0.154000
Fraction of I-135   going to Xe-135  in this chain = 0.153985
Fraction of Xe-135m going to Xe-135  in this chain = 0.999900

Decay Chain #  Kr-85m   Kr-85
Fraction of Kr-85m  going to Kr-85   in this chain = 0.211000

Decay Chain #  Kr-87

Decay Chain #  Kr-88    Rb-88
Fraction of Kr-88   going to Rb-88   in this chain = 1.000000

Decay Chain #  La-141   Ce-141
Fraction of La-141  going to Ce-141  in this chain = 1.000000
```

Decay Chain # La-142

Decay Chain # Mo-99 Tc-99m
Fraction of Mo-99 going to Tc-99m in this chain = 0.876000

Decay Chain # Nd-147

Decay Chain # Np-239 Pu-239
Fraction of Np-239 going to Pu-239 in this chain = 1.000000

Decay Chain # Pu-241 Am-241
Fraction of Pu-241 going to Am-241 in this chain = 1.000000

Decay Chain # Rb-86

Decay Chain # Ru-103 Rh-103m
Fraction of Ru-103 going to Rh-103m in this chain = 0.997000

Decay Chain # Ru-105 Rh-105
Fraction of Ru-105 going to Rh-105 in this chain = 1.000000

Decay Chain # Ru-106 Rh-106
Fraction of Ru-106 going to Rh-106 in this chain = 1.000000

Decay Chain # Sr-89

Decay Chain # Sr-90 Y-90
Fraction of Sr-90 going to Y-90 in this chain = 1.000000

Decay Chain # Sr-91 Y-91
Fraction of Sr-91 going to Y-91 in this chain = 0.422000

Decay Chain # Sr-91 Y-91m Y-91
Fraction of Sr-91 going to Y-91m in this chain = 0.578000
Fraction of Sr-91 going to Y-91 in this chain = 0.578000
Fraction of Y-91m going to Y-91 in this chain = 1.000000

Decay Chain # Sr-92 Y-92
Fraction of Sr-92 going to Y-92 in this chain = 1.000000

Decay Chain # Te-127m Te-127
Fraction of Te-127m going to Te-127 in this chain = 0.976000

Decay Chain # Te-129m Te-129
Fraction of Te-129m going to Te-129 in this chain = 0.650000

Decay Chain # Te-131m I-131
Fraction of Te-131m going to I-131 in this chain = 0.778000

Decay Chain # Te-131m Te-131 I-131
Fraction of Te-131m going to Te-131 in this chain = 0.222000
Fraction of Te-131m going to I-131 in this chain = 0.222000
Fraction of Te-131 going to I-131 in this chain = 1.000000

Decay Chain # Te-132 I-132
Fraction of Te-132 going to I-132 in this chain = 1.000000

Decay Chain # Y-93

Decay Chain # Zr-95 Nb-95
Fraction of Zr-95 going to Nb-95 in this chain = 0.993000

Decay Chain # Zr-97 Nb-97
Fraction of Zr-97 going to Nb-97 in this chain = 0.053000

Decay Chain # Zr-97 Nb-97m Nb-97
Fraction of Zr-97 going to Nb-97m in this chain = 0.947000
Fraction of Zr-97 going to Nb-97 in this chain = 0.947000
Fraction of Nb-97m going to Nb-97 in this chain = 1.000000

Using distance dispersion model for sigma-y/sigma-z

Using NEW Plume Meander model for sigma-y

THE DENSITY PLUME BUOYANCY MODEL IS IN EFFECT

RELEASED INVENTORY OF ALL PLUMES

Rel #	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25
Kr-85	1.04E+13	1.09E+13	1.22E+13	4.30E+13	1.96E+13	2.11E+13	1.47E+14	4.63E+13	6.57E+13	8.10E+13	1.01E+14	9.26E+13	1.95E+14	6.88E+15	4.72E+14	7.21E+14	1.06E+15	1.70E+15	1.52E+15	2.21E+15	2.23E+15	2.35E+15	1.97E+15	2.43E+15	2.08E+15
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	1.52E+12	1.58E+12	1.78E+12	6.24E+12	2.84E+12	3.05E+12	2.11E+13	6.64E+12	9.41E+12	1.16E+13	1.44E+13	1.32E+13	2.76E+13	9.73E+14	6.66E+13	1.01E+14	1.50E+14	2.38E+14	2.13E+14	3.07E+14	3.10E+14	3.26E+14	2.72E+14	3.36E+14	2.86E+14
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	4.09E+13	4.62E+13	5.19E+13	2.15E+14	6.61E+13	5.94E+13	5.50E+14	1.17E+14	1.39E+14	1.26E+14	2.25E+14	1.93E+14	5.92E+14	1.71E+16	2.01E+15	3.56E+15	6.32E+15	9.30E+15	8.15E+15	1.14E+16	1.17E+16	1.41E+16	5.95E+15	7.05E+15	5.42E+15
Cs-136	8.73E+11	9.86E+11	1.11E+12	4.58E+12	1.41E+12	1.26E+12	1.17E+13	2.49E+12	2.94E+12	2.66E+12	4.75E+12	4.07E+12	1.25E+13	3.60E+14	4.23E+13	7.49E+13	1.33E+14	1.95E+14	1.71E+14	2.38E+14	2.45E+14	2.94E+14	1.24E+14	1.47E+14	1.13E+14
Cs-137	1.14E+14	1.29E+14	1.45E+14	5.99E+14	1.84E+14	1.65E+14	1.53E+15	3.27E+14	3.87E+14	3.50E+14	6.26E+14	5.37E+14	1.65E+15	4.77E+16	5.60E+15	9.92E+15	1.76E+16	2.59E+16	2.27E+16	3.17E+16	3.27E+16	3.92E+16	1.66E+16	1.97E+16	1.51E+16
Rb-86	4.98E+10	5.62E+10	6.31E+10	2.61E+11	8.03E+10	7.20E+10	6.67E+11	1.42E+11	1.68E+11	1.52E+11	2.72E+11	2.33E+11	7.14E+11	2.06E+13	2.42E+12	4.29E+12	7.61E+12	1.12E+13	9.79E+12	1.37E+13	1.41E+13	1.69E+13	7.13E+12	8.45E+12	6.49E+12
Rb-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	3.20E+12	3.59E+12	3.98E+12	1.39E+13	4.42E+12	3.46E+12	3.87E+13	6.74E+12	8.14E+12	6.69E+12	7.06E+12	6.09E+12	2.39E+13	8.54E+14	2.75E+13	3.20E+12	2.95E+12	3.55E+12	3.49E+12	4.66E+12	4.08E+12	3.51E+12	2.36E+12	3.51E+12	3.32E+12
Sr-89	7.15E+12	8.02E+12	8.90E+12	3.10E+13	9.90E+12	7.75E+12	8.69E+13	1.51E+13	1.83E+13	1.50E+13	1.59E+13	1.37E+13	5.38E+13	1.93E+15	6.21E+13	7.23E+12	6.68E+12	8.03E+12	7.90E+12	1.06E+13	9.26E+12	7.97E+12	5.36E+12	7.97E+12	7.55E+12
Sr-90	2.05E+13	2.30E+13	2.55E+13	8.90E+13	2.84E+13	2.22E+13	2.49E+14	4.34E+13	5.25E+13	4.32E+13	4.57E+13	3.95E+13	1.55E+14	5.55E+15	1.79E+14	2.08E+13	1.92E+13	2.31E+13	2.27E+13	3.04E+13	2.67E+13	2.30E+13	1.55E+13	2.30E+13	2.18E+13
Sr-91	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-137m	1.08E+14	1.22E+14	1.37E+14	5.66E+14	1.74E+14	1.57E+14	1.45E+15	3.09E+14	3.66E+14	3.31E+14	5.93E+14	5.08E+14	1.56E+15	4.51E+16	5.30E+15	9.38E+15	1.67E+16	2.45E+16	2.15E+16	3.00E+16	3.09E+16	3.71E+16	1.57E+16	1.86E+16	1.43E+16
I-131	1.01E+13	1.11E+13	1.23E+13	7.97E+12	6.36E+12	6.60E+12	6.27E+13	1.40E+13	1.56E+13	1.37E+13	2.39E+13	2.05E+13	1.11E+13	9.75E+14	1.12E+14	1.34E+14	3.64E+14	5.46E+14	4.50E+14	5.64E+14	6.35E+14	5.92E+14	6.73E+14	7.60E+14	6.31E+14
I-132	1.06E+11	1.19E+11	1.32E+11	8.55E+10	6.75E+10	6.93E+10	6.58E+11	1.42E+11	1.59E+11	1.40E+11	2.36E+11	2.10E+11	1.19E+11	9.94E+12	1.14E+12	1.36E+12	3.69E+12	5.52E+12	4.54E+12	5.64E+12	6.31E+12	5.72E+12	6.53E+12	8.59E+12	6.62E+12
I-133	2.02E+01	2.18E+01	2.38E+01	1.52E+01	1.20E+01	1.23E+01	1.15E+02	2.52E+01	2.77E+01	2.40E+01	4.13E+01	3.49E+01	1.85E+01	1.61E+03	1.82E+02	2.17E+02	5.82E+02	8.60E+02	6.99E+02	8.62E+02	9.58E+02	8.79E+02	9.85E+02	1.10E+03	8.97E+02
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-127	3.67E+12	4.14E+12	4.62E+12	3.01E+12	2.38E+12	2.46E+12	2.34E+13	5.09E+12	5.73E+12	5.06E+12	8.57E+12	7.63E+12	4.34E+12	3.65E+14	4.21E+13	5.03E+13	1.37E+14	2.05E+14	1.70E+14	2.12E+14	2.38E+14	2.16E+14	2.48E+14	3.28E+14	2.54E+14
Te-127m	3.75E+12	4.22E+12	4.72E+12	3.07E+12	2.43E+12	2.51E+12	2.39E+13	5.20E+12	5.84E+12	5.17E+12	8.74E+12	7.78E+12	4.43E+12	3.73E+14	4.30E+13	5.14E+13	1.40E+14	2.10E+14	1.73E+14	2.16E+14	2.43E+14	2.21E+14	2.53E+14	3.34E+14	2.59E+14
Te-129	4.41E+12	4.97E+12	5.55E+12	3.61E+12	2.86E+12	2.95E+12	2.81E+13	6.11E+12	6.86E+12	6.07E+12	1.03E+13	9.13E+12	5.20E+12	4.37E+14	5.04E+13	6.02E+13	1.64E+14	2.45E+14	2.03E+14	2.53E+14	2.84E+14	2.58E+14	2.96E+14	3.91E+14	3.03E+14
Te-129m	6.78E+12	7.63E+12	8.53E+12	5.55E+12	4.39E+12	4.53E+12	4.32E+13	9.38E+12	1.05E+13	9.32E+12	1.58E+13	1.40E+13	7.99E+12	6.71E+14	7.74E+13	9.25E+13	2.51E+14	3.77E+14	3.12E+14	3.88E+14	4.36E+14	3.97E+14	4.55E+14	6.01E+14	4.65E+14
Te-131m	2.66E+04	2.95E+04	3.26E+04	2.10E+04	1.64E+04	1.68E+04	1.58E+05	3.39E+04	3.77E+04	3.30E+04	5.52E+04	4.86E+04	2.73E+04	2.27E+06	2.60E+05	3.09E+05	8.32E+05	1.23E+06	1.01E+06	1.24E+06	1.38E+06	1.24E+06	1.41E+06	1.84E+06	1.41E+06

Te-132	1.03E+11	1.15E+11	1.28E+11	8.30E+10	6.55E+10	6.72E+10	6.38E+11	1.38E+11	1.55E+11	1.36E+11	2.29E+11	2.03E+11	1.15E+11	9.65E+12	1.11E+12	1.32E+12	3.59E+12	5.36E+12	4.41E+12	5.47E+12	6.12E+12	5.55E+12	6.34E+12	8.33E+12	6.43E+12
Te-131	5.99E+03	6.64E+03	7.34E+03	4.73E+03	3.70E+03	3.77E+03	3.56E+04	7.64E+03	8.49E+03	7.42E+03	1.24E+04	1.09E+04	6.15E+03	5.12E+05	5.85E+04	6.95E+04	1.87E+05	2.78E+05	2.27E+05	2.80E+05	3.11E+05	2.80E+05	3.17E+05	4.14E+05	3.17E+05
Rh-105	3.06E+01	1.60E+02	3.55E+02	5.02E+02	6.79E+02	5.81E+02	9.43E+02	6.09E+02	1.24E+03	1.46E+03	1.57E+03	1.37E+03	1.58E+03	6.59E+04	1.53E+04	1.26E+04	1.60E+05	4.82E+05	4.57E+05	6.50E+05	8.69E+05	9.17E+05	6.47E+05	7.26E+05	4.45E+05
Ru-103	1.57E+09	8.35E+09	1.86E+10	2.66E+10	3.64E+10	3.14E+10	5.15E+10	3.36E+10	6.90E+10	8.21E+10	8.88E+10	7.83E+10	9.10E+10	3.84E+12	8.99E+11	7.46E+11	9.52E+12	2.89E+13	2.76E+13	3.97E+13	5.36E+13	5.71E+13	4.07E+13	4.61E+13	2.85E+13
Ru-105	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.64E+09	8.72E+09	1.95E+10	2.79E+10	3.80E+10	3.29E+10	5.39E+10	3.52E+10	7.22E+10	8.61E+10	9.30E+10	8.21E+10	9.55E+10	4.03E+12	9.44E+11	7.83E+11	1.00E+13	3.04E+13	2.90E+13	4.17E+13	5.64E+13	6.01E+13	4.28E+13	4.85E+13	3.00E+13
Rh-103m	1.57E+09	8.33E+09	1.86E+10	2.66E+10	3.63E+10	3.14E+10	5.14E+10	3.35E+10	6.88E+10	8.20E+10	8.86E+10	7.81E+10	9.08E+10	3.83E+12	8.97E+11	7.44E+11	9.50E+12	2.88E+13	2.76E+13	3.96E+13	5.35E+13	5.70E+13	4.06E+13	4.60E+13	2.84E+13
Rh-106	1.64E+09	8.72E+09	1.95E+10	2.79E+10	3.80E+10	3.29E+10	5.39E+10	3.52E+10	7.22E+10	8.61E+10	9.30E+10	8.21E+10	9.55E+10	4.03E+12	9.44E+11	7.83E+11	1.00E+13	3.04E+13	2.90E+13	4.17E+13	5.64E+13	6.01E+13	4.28E+13	4.85E+13	3.00E+13
Nb-95	2.59E+11	1.37E+12	3.07E+12	4.38E+12	8.38E+12	1.50E+13	2.65E+13	4.77E+13	7.45E+13	8.31E+13	1.15E+14	1.07E+14	1.17E+14	3.77E+15	1.38E+15	2.06E+15	5.50E+15	7.95E+15	6.26E+15	7.37E+15	8.06E+15	8.93E+15	7.11E+15	6.93E+15	4.01E+15
Co-58	3.14E+07	1.66E+08	3.72E+08	5.31E+08	1.02E+09	1.81E+09	3.21E+09	5.79E+09	9.05E+09	1.01E+10	1.39E+10	1.30E+10	1.43E+10	4.58E+11	1.68E+11	2.50E+11	6.69E+11	9.67E+11	7.61E+11	8.96E+11	9.81E+11	1.09E+12	8.65E+11	8.44E+11	4.89E+11
Co-60	4.67E+08	2.48E+09	5.53E+09	7.90E+09	1.51E+10	2.70E+10	4.78E+10	8.63E+10	1.35E+11	1.50E+11	2.07E+11	1.93E+11	2.13E+11	6.83E+12	2.50E+12	3.73E+12	9.98E+12	1.44E+13	1.14E+13	1.34E+13	1.46E+13	1.62E+13	1.29E+13	1.26E+13	7.30E+12
Mo-99	1.99E+07	1.05E+08	2.33E+08	3.31E+08	6.31E+08	1.12E+09	1.97E+09	3.54E+09	5.51E+09	6.11E+09	8.38E+09	7.78E+09	8.51E+09	2.72E+11	9.93E+10	1.48E+11	3.93E+11	5.65E+11	4.43E+11	5.19E+11	5.65E+11	6.23E+11	4.93E+11	4.79E+11	2.76E+11
Tc-99m	1.92E+07	1.01E+08	2.25E+08	3.19E+08	6.09E+08	1.08E+09	1.90E+09	3.42E+09	5.31E+09	5.89E+09	8.08E+09	7.50E+09	8.21E+09	2.62E+11	9.57E+10	1.42E+11	3.79E+11	5.45E+11	4.27E+11	5.00E+11	5.44E+11	6.00E+11	4.75E+11	4.61E+11	2.66E+11
Nb-97	0.00E+00	1.25E-14	2.10E-14	2.24E-14	3.22E-14	4.31E-14	5.72E-14	7.83E-14	9.17E-14	7.66E-14	7.92E-14	5.54E-14	4.57E-14	1.10E-12	3.25E-13	4.19E-13	9.02E-13	9.77E-13	5.76E-13	5.09E-13	4.17E-13	3.47E-13	2.07E-13	1.51E-13	6.56E-14
Nb-97m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	3.03E+06	3.95E+06	6.24E+06	7.88E+06	2.00E+07	3.13E+07	2.35E+07	3.00E+07	2.71E+06	6.09E+05	7.33E+06	5.24E+06	4.57E+06	2.47E+08	1.06E+08	1.84E+08	4.58E+08	7.46E+08	6.59E+08	8.94E+08	1.06E+09	9.29E+08	5.89E+08	9.17E+08	9.62E+08
Ce-143	2.11E-02	2.72E-02	4.25E-02	5.31E-02	1.33E-01	2.07E-01	1.54E-01	1.94E-01	1.74E-02	3.87E-03	4.61E-02	3.26E-02	2.81E-02	1.50E+00	6.42E-01	1.11E+00	2.74E+00	4.41E+00	3.86E+00	5.19E+00	6.10E+00	5.28E+00	3.32E+00	5.11E+00	5.31E+00
Ce-144	5.96E+06	7.78E+06	1.23E+07	1.55E+07	3.93E+07	6.18E+07	4.64E+07	5.92E+07	5.34E+06	1.20E+06	1.45E+07	1.04E+07	9.03E+06	4.88E+08	2.10E+08	3.65E+08	9.07E+08	1.48E+09	1.31E+09	1.77E+09	2.11E+09	1.84E+09	1.17E+09	1.82E+09	1.91E+09
Np-239	2.28E+03	2.96E+03	4.64E+03	5.83E+03	1.47E+04	2.29E+04	1.71E+04	2.17E+04	1.95E+03	4.36E+02	5.22E+03	3.71E+03	3.21E+03	1.72E+05	7.39E+04	1.28E+05	3.16E+05	5.13E+05	4.51E+05	6.08E+05	7.18E+05	6.24E+05	3.94E+05	6.09E+05	6.36E+05
Pu-238	2.23E+05	2.91E+05	4.59E+05	5.80E+05	1.47E+06	2.31E+06	1.74E+06	2.21E+06	2.00E+05	4.50E+04	5.42E+05	3.88E+05	3.38E+05	1.83E+07	7.86E+06	1.37E+07	3.39E+07	5.53E+07	4.89E+07	6.63E+07	7.88E+07	6.89E+07	4.38E+07	6.81E+07	7.15E+07
Pu-239	1.76E+04	2.30E+04	3.63E+04	4.58E+04	1.16E+05	1.82E+05	1.37E+05	1.75E+05	1.58E+04	3.55E+03	4.28E+04	3.06E+04	2.67E+04	1.44E+06	6.20E+05	1.08E+06	2.68E+06	4.36E+06	3.86E+06	5.24E+06	6.22E+06	5.44E+06	3.46E+06	5.38E+06	5.65E+06
Pu-240	3.59E+04	4.69E+04	7.40E+04	9.35E+04	2.37E+05	3.72E+05	2.80E+05	3.57E+05	3.22E+04	7.26E+03	8.74E+04	6.25E+04	5.45E+04	2.94E+06	1.27E+06	2.20E+06	5.47E+06	8.91E+06	7.88E+06	1.07E+07	1.27E+07	1.11E+07	7.05E+06	1.10E+07	1.15E+07
Pu-241	5.02E+06	6.56E+06	1.03E+07	1.31E+07	3.31E+07	5.21E+07	3.91E+07	4.99E+07	4.50E+06	1.01E+06	1.22E+07	8.73E+06	7.61E+06	4.11E+08	1.77E+08	3.08E+08	7.64E+08	1.24E+09	1.10E+09	1.49E+09	1.78E+09	1.55E+09	9.86E+08	1.53E+09	1.61E+09
Zr-95	4.40E+06	5.75E+06	9.08E+06	1.15E+07	2.91E+07	4.57E+07	3.43E+07	4.37E+07	3.95E+06	8.89E+05	1.07E+07	7.65E+06	6.66E+06	3.60E+08	1.55E+08	2.69E+08	6.69E+08	1.09E+09	9.63E+08	1.31E+09	1.55E+09	1.36E+09	8.62E+08	1.34E+09	1.41E+09
Zr-97	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am-241	9.13E+04	1.19E+05	1.88E+05	2.38E+05	6.07E+05	9.63E+05	7.16E+05	9.15E+05	8.54E+04	1.85E+04	2.24E+05	1.60E+05	1.40E+05	7.57E+06	3.30E+06	5.83E+06	1.42E+07	2.31E+07	2.05E+07	2.81E+07	3.33E+07	2.93E+07	1.77E+07	3.54E+07	2.98E+07
Cm-242	3.96E+05	5.17E+05	8.17E+05	1.03E+06	2.63E+06	4.18E+06	3.10E+06	3.97E+06	3.70E+05	8.01E+04	9.72E+05	6.94E+05	6.04E+05	3.28E+07	1.43E+07	2.53E+07	6.15E+07	1.00E+08	8.89E+07	1.22E+08	1.44E+08	1.27E+08	7.66E+07	1.53E+08	1.29E+08
Cm-244	2.11E+05	2.75E+05	4.34E+05	5.49E+05	1.40E+06	2.22E+06	1.65E+06	2.11E+06	1.97E+05	4.26E+04	5.17E+05	3.69E+05	3.22E+05	1.75E+07	7.60E+06	1.34E+07	3.27E+07	5.33E+07	4.73E+07	6.48E+07	7.68E+07	6.75E+07	4.08E+07	8.17E+07	6.88E+07
La-140	1.68E+12	1.90E+12	2.13E+12	7.48E+12	2.40E+12	1.89E+12	2.14E+13	3.75E+12	4.57E+12	3.78E+12	4.03E+12	3.50E+12	1.38E+13	4.98E+14	1.61E+13	1.88E+12	1.75E+12	2.11E+12	2.09E+12	2.81E+12	2.48E+12	2.15E+12	1.45E+12	2.17E+12	2.07E+12
La-141	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nd-147	2.34E+05	3.05E+05	4.81E+05	6.07E+05	1.54E+06	2.45E+06	1.82E+06	2.32E+06	2.16E+05	4.67E+04	5.66E+05	4.04E+05	3.51E+05	1.90E+07	8.28E+06	1.46E+07	3.56E+07	5.80E+07	5.14E+07	7.03E+07	8.31E+07	7.29E+07	4.41E+07	8.81E+07	7.41E+07
Pr-143	9.10E+05	1.19E+06	1.87E+06	2.36E+06	6.02E+06	9.55E+06	7.09E+06	9.06E+06	8.44E+05	1.82E+05	2.21E+06	1.58E+06	1.37E+06	7.44E+07	3.24E+07	5.72E+07	1.39E+08	2.27E+08	2.01E+08	2.75E+08	3.25E+08	2.85E+08	1.73E+08	3.45E+08	2.90E+08
Y-90	7.29E+12	8.28E+12	9.28E+12	3.27E+13	1.05E+13	8.31E+12	9.42E+13	1.65E+13	2.02E+13	1.67E+13	1.79E+13	1.55E+13	6.15E+13	2.22E+15	7.20E+13	8.42E+12	7.82E+12	9.48E+12	9.40E+12	1.27E+13	1.12E+13	9.71E+12	6.59E+12	9.87E+12	9.41E+12
Y-91	2.78E+06	3.63E+06	5.73E+06	7.24E+06	1.85E+07	2.93E+07	2.18E+07	2.78E+07	2.60E+06	5.62E+05	6.81E+06	4.86E+06	4.24E+06	2.30E+08	1.00E+08	1.77E+08	4.30E+08	7.02E+08	6.23E+08	8.52E+08	1.01E+09	8.86E+08	5.36E+08	1.07E+09	9.03E+08
Y-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	1.16E-21	1.45E-21	2.21E-21	2.70E-21	6.64E-21	1.02E-20	7.31E-21	9.04E-21	8.16E-22	1.70E-22	2.00E-21	1.38E-21	1.16E-21	6.09E-20	2.58E-20	4.49E-20	1.07E-19	1.68E-19	1.44E-19	1.90E-19	2.18E-19	1.85E-19	1.08E-19	2.09E-19	1.70E-19
Y-91m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	5.96E+06	7.78E+06	1.23E+07	1.55E+07	3.94E+07	6.18E+07	4.64E+07	5.92E+07	5.34E+06	1.20E+06	1.45E+07	1.04E+07	9.03E+06	4.88E+08	2.10E+08	3.65E+08	9.07E+08	1.48E+09	1.31E+09	1.77E+09	2.11E+09	1.84E+09	1.17E+09	1.82E+09	1.91E+09
Pr-144m	1.06E+05	1.39E+05	2.19E+05	2.76E+05	7.00E+05	1.10E+06	8.27E+05	1.05E+06	9.51E+04	2.14E+04	2.58E+05	1.84E+05	1.61E+05	8.69E+06	3.74E+06	6.50E+06	1.61E+07	2.63E+07	2.33E+07	3.16E+07	3.75E+07	3.28E+07	2.08E+07	3.24E+07	3.40E+07

Rel #	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48	49	50
Kr-85	2.05E+15	1.25E+15	1.20E+15	1.04E+15	1.04E+15	9.50E+14	1.31E+15	1.29E+15	1.60E+15	1.52E+15	1.80E+15	1.61E+15	1.89E+15	1.68E+15	1.97E+15	1.72E+15	1.97E+15	1.70E+15	1.93E+15	1.69E+15	1.92E+15	1.67E+15	1.89E+15	1.63E+15	1.67E+15
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	2.81E+14	1.71E+14	1.63E+14	1.41E+14	1.41E+14	1.28E+14	1.76E+14	1.74E+14	2.15E+14	2.04E+14	2.41E+14	2.15E+14	2.50E+14	2.23E+14	2.60E+14	2.26E+14	2.58E+14	2.22E+14	2.52E+14	2.20E+14	2.49E+14	2.16E+14	2.44E+14	2.10E+14	2.15E+14
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	6.73E+15	5.29E+15	5.78E+15	5.26E+15	5.40E+15	4.90E+15	6.77E+15	6.57E+15	8.07E+15	7.47E+15	8.75E+15	7.64E+15	8.63E+15	7.47E+15	8.53E+15	7.23E+15	8.08E+15	6.73E+15	7.31E+15	6.12E+15	6.65E+15	5.55E+15	6.00E+15	5.00E+15	5.41E+15
Cs-136	1.40E+14	1.10E+14	1.20E+14	1.09E+14	1.12E+14	1.01E+14	1.40E+14	1.36E+14	1.66E+14	1.54E+14	1.80E+14	1.57E+14	1.77E+14	1.53E+14	1.75E+14	1.48E+14	1.65E+14	1.38E+14	1.49E+14	1.25E+14	1.35E+14	1.13E+14	1.22E+14	1.02E+14	1.10E+14
Cs-137	1.88E+16	1.47E+16	1.61E+16	1.47E+16	1.51E+16	1.36E+16	1.89E+16	1.83E+16	2.25E+16	2.08E+16	2.44E+16	2.13E+16	2.41E+16	2.08E+16	2.38E+16	2.01E+16	2.25E+16	1.88E+16	2.04E+16	1.71E+16	1.85E+16	1.55E+16	1.67E+16	1.39E+16	1.51E+16
Rb-86	8.05E+12	6.32E+12	6.90E+12	6.27E+12	6.44E+12	5.83E+12	8.06E+12	7.81E+12	9.59E+12	8.87E+12	1.04E+13	9.06E+12	1.02E+13	8.85E+12	1.01E+13	8.54E+12	9.54E+12	7.95E+12	8.63E+12	7.22E+12	7.84E+12	6.54E+12	7.06E+12	5.88E+12	6.35E+12
Rb-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	4.81E+12	5.31E+12	5.85E+12	2.34E+12	2.18E+12	3.80E+12	7.43E+12	1.02E+13	6.45E+12	3.62E+12	3.80E+12	3.49E+12	4.87E+12	5.03E+12	5.57E+12	4.92E+12	4.51E+12	3.80E+12	4.45E+12	4.00E+12	4.66E+12	4.15E+12	4.83E+12	4.32E+12	4.78E+12
Sr-89	1.10E+13	1.21E+13	1.33E+13	5.35E+12	4.99E+12	8.69E+12	1.70E+13	2.35E+13	1.48E+13	8.31E+12	8.73E+12	8.03E+12	1.12E+13	1.16E+13	1.28E+13	1.13E+13	1.04E+13	8.77E+12	1.03E+13	9.26E+12	1.08E+13	9.63E+12	1.12E+13	1.00E+13	1.11E+13
Sr-90	3.16E+13	3.49E+13	3.85E+13	1.54E+13	1.44E+13	2.51E+13	4.91E+13	6.78E+13	4.28E+13	2.40E+13	2.53E+13	2.32E+13	3.24E+13	3.36E+13	3.72E+13	3.29E+13	3.02E+13	2.54E+13	2.98E+13	2.69E+13	3.14E+13	2.80E+13	3.25E+13	2.91E+13	3.23E+13
Sr-91	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-137m	1.77E+16	1.39E+16	1.52E+16	1.39E+16	1.42E+16	1.29E+16	1.78E+16	1.73E+16	2.13E+16	1.97E+16	2.31E+16	2.02E+16	2.28E+16	1.97E+16	2.25E+16	1.91E+16	2.13E+16	1.78E+16	1.93E+16	1.62E+16	1.75E+16	1.47E+16	1.58E+16	1.32E+16	1.43E+16
I-131	4.84E+14	2.53E+14	2.31E+14	2.13E+14	2.69E+14	2.37E+14	3.07E+14	2.80E+14	3.53E+14	3.29E+14	4.05E+14	3.67E+14	4.36E+14	3.95E+14	4.66E+14	4.18E+14	4.93E+14	4.18E+14	4.79E+14	4.22E+14	4.84E+14	4.15E+14	4.68E+14	4.09E+14	4.63E+14
I-132	4.70E+12	2.68E+12	2.60E+12	2.31E+12	2.82E+12	2.29E+12	2.96E+12	2.81E+12	3.46E+12	3.22E+12	3.86E+12	3.49E+12	4.13E+12	3.74E+12	4.40E+12	3.94E+12	4.64E+12	3.92E+12	4.48E+12	3.94E+12	4.52E+12	3.86E+12	4.34E+12	3.79E+12	4.29E+12
I-133	6.77E+02	3.49E+02	3.14E+02	2.85E+02	3.55E+02	3.07E+02	3.93E+02	3.53E+02	4.38E+02	4.03E+02	4.89E+02	4.36E+02	5.11E+02	4.56E+02	5.30E+02	4.69E+02	5.44E+02	4.54E+02	5.13E+02	4.45E+02	5.04E+02	4.25E+02	4.73E+02	4.07E+02	4.54E+02
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-127	1.81E+14	1.04E+14	1.01E+14	9.02E+13	1.11E+14	9.02E+13	1.17E+14	1.11E+14	1.38E+14	1.29E+14	1.55E+14	1.41E+14	1.67E+14	1.52E+14	1.80E+14	1.62E+14	1.91E+14	1.62E+14	1.86E+14	1.65E+14	1.89E+14	1.63E+14	1.84E+14	1.61E+14	1.83E+14
Te-127m	1.85E+14	1.06E+14	1.03E+14	9.20E+13	1.13E+14	9.21E+13	1.19E+14	1.14E+14	1.41E+14	1.32E+14	1.58E+14	1.44E+14	1.71E+14	1.55E+14	1.84E+14	1.65E+14	1.95E+14	1.66E+14	1.90E+14	1.68E+14	1.93E+14	1.66E+14	1.87E+14	1.64E+14	1.87E+14
Te-129	2.16E+14	1.24E+14	1.20E+14	1.07E+14	1.32E+14	1.08E+14	1.39E+14	1.33E+14	1.64E+14	1.53E+14	1.84E+14	1.67E+14	1.99E+14	1.81E+14	2.14E+14	1.92E+14	2.27E+14	1.93E+14	2.21E+14	1.95E+14	2.25E+14	1.93E+14	2.18E+14	1.91E+14	2.17E+14
Te-129m	3.32E+14	1.90E+14	1.85E+14	1.65E+14	2.02E+14	1.65E+14	2.14E+14	2.04E+14	2.52E+14	2.36E+14	2.83E+14	2.57E+14	3.06E+14	2.78E+14	3.28E+14	2.95E+14	3.49E+14	2.96E+14	3.40E+14	3.00E+14	3.45E+14	2.96E+14	3.34E+14	2.93E+14	3.34E+14
Te-131m	9.94E+05	5.63E+05	5.41E+05	4.78E+05	5.80E+05	4.68E+05	5.99E+05	5.65E+05	6.90E+05	6.38E+05	7.59E+05	6.81E+05	8.02E+05	7.19E+05	8.42E+05	7.48E+05	8.74E+05	7.33E+05	8.33E+05	7.27E+05	8.28E+05	7.03E+05	7.84E+05	6.79E+05	7.65E+05
Te-132	4.57E+12	2.60E+12	2.52E+12	2.24E+12	2.74E+12	2.23E+12	2.87E+12	2.73E+12	3.36E+12	3.13E+12	3.75E+12	3.38E+12	4.01E+12	3.63E+12	4.27E+12	3.82E+12	4.50E+12	3.80E+12	4.35E+12	3.82E+12	4.38E+12	3.75E+12	4.21E+12	3.68E+12	4.17E+12
Te-131	2.24E+05	1.27E+05	1.22E+05	1.08E+05	1.30E+05	1.05E+05	1.35E+05	1.27E+05	1.55E+05	1.44E+05	1.71E+05	1.53E+05	1.80E+05	1.62E+05	1.90E+05	1.68E+05	1.97E+05	1.65E+05	1.88E+05	1.64E+05	1.86E+05	1.58E+05	1.77E+05	1.53E+05	1.72E+05
Rh-105	3.13E+05	2.60E+05	2.88E+05	2.74E+05	3.34E+05	2.83E+05	3.67E+05	3.66E+05	4.62E+05	4.47E+05	5.65E+05	5.42E+05	6.72E+05	6.38E+05	7.89E+05	7.36E+05	8.82E+05	7.97E+05	9.41E+05	8.47E+05	9.88E+05	8.67E+05	9.99E+05	9.12E+05	1.08E+06
Ru-103	2.03E+13	1.69E+13	1.90E+13	1.82E+13	2.25E+13	1.92E+13	2.51E+13	2.53E+13	3.22E+13	3.15E+13	4.02E+13	3.89E+13	4.87E+13	4.67E+13	5.83E+13	5.48E+13	6.63E+13	6.06E+13	7.22E+13	6.56E+13	7.72E+13	6.84E+13	7.95E+13	7.33E+13	8.76E+13
Ru-105	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	2.13E+13	1.79E+13	2.00E+13	1.92E+13	2.37E+13	2.03E+13	2.65E+13	2.67E+13	3.40E+13	3.33E+13	4.25E+13	4.11E+13	5.15E+13	4.94E+13	6.16E+13	5.81E+13	7.02E+13	6.41E+13	7.64E+13	6.95E+13	8.18E+13	7.25E+13	8.44E+13	7.78E+13	9.30E+13
Rh-103m	2.02E+13	1.69E+13	1.90E+13	1.82E+13	2.24E+13	1.92E+13	2.51E+13	2.53E+13	3.22E+13	3.14E+13	4.01E+13	3.88E+13	4.86E+13	4.66E+13	5.81E+13	5.47E+13	6.62E+13	6.04E+13	7.20E+13	6.54E+13	7.70E+13	6.83E+13	7.94E+13	7.32E+13	8.74E+13
Rh-106	2.13E+13	1.79E+13	2.00E+13	1.92E+13	2.37E+13	2.03E+13	2.65E+13	2.67E+13	3.40E+13	3.33E+13	4.25E+13	4.11E+13	5.15E+13	4.94E+13	6.16E+13	5.81E+13	7.02E+13	6.41E+13	7.64E+13	6.95E+13	8.18E+13	7.25E+13	8.44E+13	7.78E+13	9.30E+13
Nb-95	2.73E+15	2.16E+15	2.43E+15	2.23E+15	2.78E+15	2.37E+15	3.06E+15	2.87E+15	3.51E+15	3.22E+15	3.91E+15	3.48E+15	4.06E+15	3.61E+15	4.19E+15	3.69E+15	4.26E+15	3.59E+15	3.97E+15	3.40E+15	3.80E+15	3.18E+15	3.52E+15	2.99E+15	3.34E+15
Co-58	3.32E+11	2.63E+11	2.97E+11	2.71E+11	3.39E+11	2.89E+11	3.74E+11	3.51E+11	4.28E+11	3.93E+11	4.77E+11	4.25E+11	4.95E+11	4.41E+11	5.12E+11	4.50E+11	5.20E+11	4.39E+11	4.86E+11	4.16E+11	4.64E+11	3.89E+11	4.30E+11	3.66E+11	4.08E+11
Co-60	4.96E+12	3.93E+12	4.44E+12	4.06E+12	5.08E+12	4.32E+12	5.59E+12	5.25E+12	6.40E+12	5.89E+12	7.14E+12	6.36E+12	7.42E+12	6.61E+12	7.67E+12	6.75E+12	7.79E+12	6.57E+12	7.29E+12	6.23E+12	6.96E+12	5.84E+12	6.46E+12	5.49E+12	6.13E+12
Mo-99	1.86E+11	1.47E+11	1.65E+11	1.50E+11	1.87E+11	1.58E+11	2.04E+11	1.90E+11	2.31E+11	2.11E+11	2.54E+11	2.26E+11	2.62E+11	2.32E+11	2.68E+11	2.34E+11	2.69E+11	2.26E+11	2.49E+11	2.12E+11	2.36E+11	1.97E+11	2.16E+11	1.83E+11	2.03E+11
Tc-99m	1.80E+11	1.42E+11	1.59E+11	1.45E+11	1.80E+11	1.52E+11	1.96E+11	1.83E+11	2.22E+11	2.03E+11	2.45E+11	2.17E+11	2.52E+11	2.23E+11	2.58E+11	2.26E+11	2.60E+11	2.18E+11	2.40E+11	2.04E+11	2.27E+11	1.90E+11	2.08E+11	1.76E+11	1.96E+11
Nb-97	3.35E-14	1.99E-14	1.68E-14	1.15E-14	1.08E-14	6.88E-15	6.68E-15	4.70E-15	4.29E-15	2.96E-15	2.69E-15	1.80E-15	1.57E-15	1.05E-15	9.12E-16	6.01E-16	5.21E-16	3.29E-16	2.73E-16	1.75E-16	1.47E-16	9.23E-17	7.64E-17	4.87E-17	4.07E-17
Nb-97m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	6.43E+08	4.77E+08	4.98E+08	4.30E+08	4.57E+08	3.43E+08	4.05E+08	3.64E+08	4.89E+08	4.12E+08	4.91E+08	4.39E+08	5.35E+08	5.01E+08	6.35E+08	6.65E+08	9.36E+08	8.58E+08	1.04E+09	9.62E+08	1.14E+09	9.89E+08	9.02E+08	8.22E+08	1.45E+09
Ce-143	3.51E+00	2.58E+00	2.67E+00	2.28E+00	2.40E+00	1.78E+00	2.08E+00	1.85E+00	2.46E+00	2.06E+00	2.42E+00	2.15E+00	2.59E+00	2.40E+00	3.01E+00	3.12E+00	4.35E+00	3.95E+00	4.75E+00	4.34E+00	5.08E+00	4.37E+00	3.95E+00	3.56E+00	6.22E+00
Ce-144	1.28E+09	9.49E+08	9.91E+08	8.56E+08	9.10E+08	6.84E+08	8.08E+08	7.25E+08	9.75E+08	8.22E+08	9.79E+08	8.77E+08	1.07E+09	1.00E+09	1.27E+09	1.33E+09	1.87E+09	1.72E+09	2.09E+09	1.93E+09	2.28E+09	1.98E+09	1.81E+09	1.65E+09	2.91E+09
Np-239	4.22E+05	3.12E+05	3.24E+05	2.78E+05	2.93E+05	2.19E+05	2.57E+05	2.30E+05	3.07E+05	2.57E+05	3.04E+05	2.71E+05	3.28E+05	3.06E+05	3.85E+05	4.01E+05	5.61E+05	5.12E+05	6.18E+05	5.67E+05	6.67E+05	5.76E+05	5.23E+05	4.74E+05	8.30E+05
Pu-238	4.78E+07	3.55E+07	3.71E+07	3.20E+07	3.41E+07	2.56E+07	3.02E+07	2.72E+07	3.65E+07	3.08E+07	3.67E+07	3.28E+07	4.00E+07	3.75E+07	4.75E+07	4.98E+07	7.01E+07	6.43E+07	7.82E+07	7.22E+07	8.54E+07	7.42E+07	6.77E+07	6.18E+07	1.09E+08
Pu-239	3.78E+06	2.80E+06	2.93E+06	2.53E+06	2.69E+06	2.02E+06	2.39E+06	2.14E+06	2.88E+06	2.43E+06	2.89E+06	2.59E+06	3.16E+06	2.96E+06	3.75E+06	3.93E+06	5.53E+06	5.08E+06	6.17E+06	5.70E+06	6.74E+06	5.86E+06	5.35E+06	4.88E+06	8.60E+06
Pu-240	7.71E+06	5.72E+06	5.98E+06	5.16E+06	5.49E+06	4.12E+06	4.87E+06	4.38E+06	5.88E+06	4.96E+06	5.91E+06	5.29E+06	6.45E+06	6.04E+06	7.66E+06	8.02E+06	1.13E+07	1.04E+07	1.26E+07	1.16E+07	1.38E+07	1.20E+07	1.09E+07	9.96E+06	1.76E+07
Pu-241	1.08E+09	8.00E+08	8.36E+08	7.22E+08	7.67E+08	5.76E+08	6.81E+08	6.12E+08	8.22E+08	6.94E+08	8.26E+08	7.40E+08	9.01E+08	8.44E+08	1.07E+09	1.12E+09	1.58E+09	1.45E+09	1.76E+09	1.63E+09	1.92E+09	1.67E+09	1.53E+09	1.39E+09	2.45E+09
Zr-95	9.41E+08	6.99E+08	7.30E+08	6.30E+08	6.69E+08	5.03E+08	5.94E+08	5.33E+08	7.17E+08	6.04E+08	7.19E+08	6.45E+08	7.85E+08	7.35E+08	9.32E+08	9.76E+08	1.37E+09	1.26E+09	1.53E+09	1.41E+09	1.67E+09	1.45E+09	1.33E+09	1.21E+09	2.13E+09
Zr-97	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am-241	2.15E+07	1.48E+07	1.59E+07	1.53E+07	1.66E+07	1.23E+07	1.59E+07	1.51E+07	1.95E+07	1.67E+07	2.00E+07	1.84E+07	2.23E+07	2.04E+07	2.55E+07	2.60E+07	3.45E+07	3.10E+07	3.73E+07	3.43E+07	4.06E+07	3.57E+07	3.19E+07	3.10E+07	4.95E+07
Cm-242	9.29E+07	6.42E+07	6.88E+07	6.64E+07	7.20E+07	5.33E+07	6.89E+07	6.54E+07	8.42E+07	7.22E+07	8.67E+07	7.97E+07	9.66E+07	8.82E+07	1.10E+08	1.13E+08	1.49E+08	1.34E+08	1.61E+08	1.48E+08	1.76E+08	1.54E+08	1.38E+08	1.34E+08	2.14E+08
Cm-244	4.95E+07	3.42E+07	3.67E+07	3.54E+07	3.84E+07	2.84E+07	3.67E+07	3.49E+07	4.49E+07	3.85E+07	4.62E+07	4.25E+07	5.15E+07	4.71E+07	5.88E+07	6.00E+07	7.95E+07	7.15E+07	8.59E+07	7.91E+07	9.37E+07	8.24E+07	7.35E+07	7.14E+07	1.14E+08
La-140	3.02E+12	3.35E+12	3.72E+12	1.50E+12	1.40E+12	2.46E+12	4.84E+12	6.70E+12	4.25E+12	2.40E+12	2.53E+12	2.34E+12	3.28E+12	3.40E+12	3.79E+12	3.36E+12	3.10E+12	2.62E+12	3.09E+12	2.79E+12	3.27E+12	2.93E+12	3.42E+12	3.07E+12	3.41E+12
La-141	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nd-147	5.32E+07	3.67E+07	3.93E+07	3.79E+07	4.10E+07	3.03E+07	3.92E+07	3.71E+07	4.78E+07	4.09E+07	4.90E+07	4.51E+07	5.45E+07	4.97E+07	6.21E+07	6.33E+07	8.37E+07	7.52E+07	9.02E+07	8.29E+07	9.81E+07	8.62E+07	7.68E+07	7.45E+07	1.19E+08
Pr-143	2.08E+08	1.44E+08	1.54E+08	1.49E+08	1.61E+08	1.19E+08	1.54E+08	1.46E+08	1.88E+08	1.61E+08	1.93E+08	1.77E+08	2.14E+08	1.96E+08	2.44E+08	2.49E+08	3.29E+08	2.96E+08	3.55E+08	3.27E+08	3.86E+08	3.39E+08	3.02E+08	2.94E+08	4.68E+08
Y-90	1.38E+13	1.53E+13	1.70E+13	6.86E+12	6.45E+12	1.13E+13	2.23E+13	3.09E+13	1.97E+13	1.11E+13	1.18E+13	1.09E+13	1.53E+13	1.59E+13	1.77E+13	1.58E+13	1.46E+13	1.23E+13	1.45E+13	1.32E+13	1.55E+13	1.39E+13	1.62E+13	1.46E+13	1.63E+13
Y-91	6.50E+08	4.49E+08	4.81E+08	4.64E+08	5.03E+08	3.72E+08	4.81E+08	4.57E+08	5.88E+08	5.04E+08	6.05E+08	5.57E+08	6.74E+08	6.16E+08	7.69E+08	7.85E+08	1.04E+09	9.35E+08	1.12E+09	1.03E+09	1.22E+09	1.08E+09	9.59E+08	9.32E+08	1.49E+09
Y-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	1.18E-19	7.90E-20	8.19E-20	7.63E-20	8.00E-20	5.72E-20	7.15E-20	6.55E-20	8.16E-20	6.76E-20	7.84E-20	6.97E-20	8.16E-20	7.20E-20	8.70E-20	8.58E-20	1.10E-19	9.54E-20	1.11E-19	9.85E-20	1.13E-19	9.58E-20	8.26E-20	7.75E-20	1.20E-19
Y-91m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	1.28E+09	9.49E+08	9.91E+08	8.56E+08	9.10E+08	6.84E+08	8.08E+08	7.25E+08	9.75E+08	8.22E+08	9.79E+08	8.77E+08	1.07E+09	1.00E+09	1.27E+09	1.33E+09	1.87E+09	1.72E+09	2.09E+09	1.93E+09	2.28E+09	1.98E+09	1.81E+09	1.65E+09	2.91E+09

Pr-144m 2.28E+07 1.69E+07 1.76E+07 1.52E+07 1.62E+07 1.22E+07 1.44E+07 1.29E+07 1.74E+07 1.46E+07 1.74E+07 1.56E+07 1.90E+07 1.78E+07 2.26E+07 2.37E+07 3.33E+07 3.06E+07 3.71E+07 3.43E+07 4.06E+07 3.53E+07 3.22E+07 2.94E+07 5.17E+07

Rel #	51	52	53	54	55	56	57	58	59	60	61	62	63	64
Kr-85	1.52E+15	6.12E+15	1.68E+15	1.57E+15	1.34E+15	1.53E+15	1.35E+15	1.56E+15	1.31E+15	1.40E+15	1.16E+15	1.27E+15	1.07E+15	1.19E+15
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	1.94E+14	7.81E+14	2.14E+14	1.99E+14	1.70E+14	1.93E+14	1.70E+14	1.96E+14	1.64E+14	1.75E+14	1.44E+14	1.58E+14	1.32E+14	1.47E+14
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	4.52E+15	5.34E+15	4.38E+15	4.73E+15	3.95E+15	4.47E+15	4.00E+15	4.60E+15	3.68E+15	3.76E+15	3.05E+15	3.28E+15	2.72E+15	2.96E+15
Cs-136	9.17E+13	1.08E+14	8.86E+13	9.54E+13	7.96E+13	9.00E+13	8.04E+13	9.25E+13	7.39E+13	7.55E+13	6.10E+13	6.56E+13	5.44E+13	5.91E+13
Cs-137	1.26E+16	1.49E+16	1.22E+16	1.32E+16	1.10E+16	1.25E+16	1.11E+16	1.28E+16	1.03E+16	1.05E+16	8.50E+15	9.14E+15	7.59E+15	8.25E+15
Rb-86	5.31E+12	6.26E+12	5.13E+12	5.53E+12	4.62E+12	5.22E+12	4.67E+12	5.37E+12	4.29E+12	4.39E+12	3.55E+12	3.81E+12	3.16E+12	3.44E+12
Rb-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-139	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-140	4.68E+12	1.53E+13	5.63E+12	6.16E+12	5.53E+12	6.72E+12	6.30E+12	7.66E+12	6.97E+12	7.98E+12	7.07E+12	8.27E+12	7.38E+12	8.63E+12
Sr-89	1.09E+13	3.56E+13	1.31E+13	1.44E+13	1.29E+13	1.57E+13	1.47E+13	1.79E+13	1.63E+13	1.87E+13	1.66E+13	1.94E+13	1.73E+13	2.03E+13
Sr-90	3.16E+13	1.04E+14	3.82E+13	4.18E+13	3.76E+13	4.57E+13	4.29E+13	5.22E+13	4.75E+13	5.45E+13	4.84E+13	5.66E+13	5.06E+13	5.92E+13
Sr-91	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ba-137m	1.19E+16	1.41E+16	1.16E+16	1.25E+16	1.04E+16	1.18E+16	1.05E+16	1.21E+16	9.71E+15	9.93E+15	8.04E+15	8.65E+15	7.18E+15	7.81E+15
I-131	4.21E+14	6.76E+14	4.16E+14	4.47E+14	3.77E+14	4.30E+14	3.90E+14	4.42E+14	3.68E+14	3.85E+14	3.20E+14	3.57E+14	3.08E+14	3.57E+14
I-132	3.81E+12	5.34E+12	3.78E+12	4.08E+12	3.43E+12	3.91E+12	3.53E+12	4.00E+12	3.33E+12	3.48E+12	2.88E+12	3.21E+12	2.77E+12	3.18E+12
I-133	4.06E+02	6.43E+02	3.90E+02	4.12E+02	3.43E+02	3.85E+02	3.44E+02	3.85E+02	3.15E+02	3.25E+02	2.66E+02	2.93E+02	2.49E+02	2.84E+02
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-127	1.63E+14	2.30E+14	1.63E+14	1.77E+14	1.50E+14	1.71E+14	1.55E+14	1.77E+14	1.48E+14	1.55E+14	1.29E+14	1.44E+14	1.25E+14	1.44E+14
Te-127m	1.67E+14	2.34E+14	1.67E+14	1.81E+14	1.53E+14	1.75E+14	1.59E+14	1.80E+14	1.51E+14	1.58E+14	1.32E+14	1.47E+14	1.27E+14	1.47E+14
Te-129	1.93E+14	2.72E+14	1.93E+14	2.09E+14	1.77E+14	2.02E+14	1.84E+14	2.09E+14	1.75E+14	1.83E+14	1.52E+14	1.70E+14	1.47E+14	1.70E+14
Te-129m	2.97E+14	4.18E+14	2.97E+14	3.22E+14	2.72E+14	3.11E+14	2.82E+14	3.21E+14	2.68E+14	2.81E+14	2.34E+14	2.62E+14	2.26E+14	2.61E+14
Te-131m	6.73E+05	9.37E+05	6.59E+05	7.06E+05	5.90E+05	6.67E+05	5.98E+05	6.73E+05	5.57E+05	5.78E+05	4.75E+05	5.26E+05	4.49E+05	5.12E+05
Te-132	3.69E+12	5.18E+12	3.67E+12	3.96E+12	3.33E+12	3.79E+12	3.43E+12	3.88E+12	3.24E+12	3.38E+12	2.80E+12	3.12E+12	2.68E+12	3.08E+12
Te-131	1.52E+05	2.11E+05	1.48E+05	1.59E+05	1.33E+05	1.50E+05	1.35E+05	1.52E+05	1.25E+05	1.30E+05	1.07E+05	1.18E+05	1.01E+05	1.15E+05
Rh-105	9.76E+05	1.17E+06	1.01E+06	1.17E+06	1.02E+06	1.18E+06	1.07E+06	1.22E+06	1.03E+06	1.15E+06	9.72E+05	1.06E+06	9.05E+05	1.04E+06
Ru-103	8.00E+13	9.66E+13	8.45E+13	9.83E+13	8.71E+13	1.02E+14	9.25E+13	1.07E+14	9.11E+13	1.03E+14	8.75E+13	9.67E+13	8.30E+13	9.58E+13
Ru-105	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	8.49E+13	1.03E+14	8.98E+13	1.04E+14	9.26E+13	1.08E+14	9.84E+13	1.14E+14	9.70E+13	1.09E+14	9.32E+13	1.03E+14	8.85E+13	1.02E+14
Rh-103m	7.98E+13	9.64E+13	8.44E+13	9.81E+13	8.69E+13	1.01E+14	9.23E+13	1.07E+14	9.09E+13	1.02E+14	8.73E+13	9.65E+13	8.28E+13	9.56E+13
Rh-106	8.49E+13	1.03E+14	8.98E+13	1.04E+14	9.26E+13	1.08E+14	9.84E+13	1.14E+14	9.70E+13	1.09E+14	9.32E+13	1.03E+14	8.85E+13	1.02E+14
Nb-95	2.89E+15	3.35E+15	2.82E+15	2.96E+15	2.44E+15	2.75E+15	2.51E+15	2.73E+15	2.11E+15	2.12E+15	1.73E+15	1.88E+15	1.59E+15	1.78E+15
Co-58	3.54E+11	4.10E+11	3.45E+11	3.63E+11	2.99E+11	3.37E+11	3.07E+11	3.35E+11	2.58E+11	2.60E+11	2.12E+11	2.31E+11	1.95E+11	2.18E+11
Co-60	5.31E+12	6.16E+12	5.18E+12	5.46E+12	4.50E+12	5.06E+12	4.62E+12	5.03E+12	3.88E+12	3.91E+12	3.19E+12	3.48E+12	2.93E+12	3.28E+12
Mo-99	1.75E+11	2.02E+11	1.69E+11	1.77E+11	1.45E+11	1.63E+11	1.47E+11	1.60E+11	1.23E+11	1.23E+11	9.99E+10	1.08E+11	9.08E+10	1.01E+11
Tc-99m	1.69E+11	1.95E+11	1.63E+11	1.71E+11	1.40E+11	1.57E+11	1.42E+11	1.54E+11	1.18E+11	1.18E+11	9.63E+10	1.04E+11	8.75E+10	9.75E+10
Nb-97	2.64E-17	2.30E-17	1.45E-17	1.14E-17	7.07E-18	5.96E-18	4.08E-18	3.33E-18	1.93E-18	1.45E-18	8.90E-19	7.26E-19	4.59E-19	3.85E-19
Nb-97m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-141	1.56E+09	2.51E+09	1.52E+09	1.67E+09	1.40E+09	1.54E+09	1.38E+09	2.23E+09	2.17E+09	2.12E+09	1.82E+09	2.01E+09	1.64E+09	1.75E+09
Ce-143	6.65E+00	1.06E+01	6.33E+00	6.87E+00	5.72E+00	6.23E+00	5.50E+00	8.84E+00	8.52E+00	8.24E+00	7.00E+00	7.65E+00	6.17E+00	6.54E+00
Ce-144	3.14E+09	5.04E+09	3.05E+09	3.35E+09	2.82E+09	3.10E+09	2.77E+09	4.50E+09	4.38E+09	4.28E+09	3.67E+09	4.06E+09	3.31E+09	3.54E+09
Np-239	8.92E+05	1.42E+06	8.56E+05	9.34E+05	7.80E+05	8.54E+05	7.58E+05	1.22E+06	1.18E+06	1.15E+06	9.80E+05	1.08E+06	8.72E+05	9.28E+05
Pu-238	1.18E+08	1.89E+08	1.14E+08	1.26E+08	1.06E+08	1.16E+08	1.04E+08	1.69E+08	1.64E+08	1.60E+08	1.38E+08	1.52E+08	1.24E+08	1.33E+08
Pu-239	9.29E+06	1.49E+07	9.03E+06	9.91E+06	8.33E+06	9.18E+06	8.19E+06	1.33E+07	1.30E+07	1.27E+07	1.09E+07	1.20E+07	9.78E+06	1.05E+07
Pu-240	1.90E+07	3.05E+07	1.84E+07	2.02E+07	1.70E+07	1.87E+07	1.67E+07	2.72E+07	2.64E+07	2.59E+07	2.22E+07	2.45E+07	2.00E+07	2.14E+07
Pu-241	2.65E+09	4.26E+09	2.58E+09	2.83E+09	2.38E+09	2.62E+09	2.34E+09	3.80E+09	3.70E+09	3.61E+09	3.10E+09	3.43E+09	2.79E+09	2.99E+09
Zr-95	2.30E+09	3.70E+09	2.24E+09	2.45E+09	2.06E+09	2.27E+09	2.03E+09	3.29E+09	3.20E+09	3.13E+09	2.69E+09	2.97E+09	2.42E+09	2.59E+09
Zr-97	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am-241	5.05E+07	7.39E+07	5.05E+07	5.62E+07	4.78E+07	5.39E+07	4.67E+07	7.19E+07	6.84E+07	6.95E+07	5.99E+07	6.70E+07	5.60E+07	6.20E+07
Cm-242	2.18E+08	3.19E+08	2.18E+08	2.42E+08	2.06E+08	2.33E+08	2.02E+08	3.10E+08	2.95E+08	3.00E+08	2.58E+08	2.89E+08	2.42E+08	2.67E+08
Cm-244	1.16E+08	1.70E+08	1.16E+08	1.29E+08	1.10E+08	1.24E+08	1.08E+08	1.66E+08	1.58E+08	1.60E+08	1.38E+08	1.54E+08	1.29E+08	1.43E+08
La-140	3.36E+12	1.10E+13	4.08E+12	4.48E+12	4.04E+12	4.93E+12	4.64E+12	5.66E+12	5.17E+12	5.95E+12	5.29E+12	6.22E+12	5.57E+12	6.54E+12
La-141	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-142	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nd-147	1.21E+08	1.77E+08	1.21E+08	1.34E+08	1.14E+08	1.29E+08	1.11E+08	1.71E+08	1.62E+08	1.65E+08	1.42E+08	1.58E+08	1.32E+08	1.46E+08
Pr-143	4.78E+08	6.98E+08	4.77E+08	5.29E+08	4.50E+08	5.07E+08	4.39E+08	6.75E+08	6.41E+08	6.51E+08	5.60E+08	6.26E+08	5.23E+08	5.78E+08
Y-90	1.60E+13	5.28E+13	1.95E+13	2.15E+13	1.94E+13	2.38E+13	2.24E+13	2.74E+13	2.51E+13	2.89E+13	2.58E+13	3.03E+13	2.72E+13	3.20E+13
Y-91	1.52E+09	2.22E+09	1.52E+09	1.69E+09	1.44E+09	1.62E+09	1.40E+09	2.16E+09	2.05E+09	2.09E+09	1.80E+09	2.01E+09	1.68E+09	1.86E+09
Y-92	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-93	1.18E-19	1.67E-19	1.10E-19	1.18E-19	9.74E-20	1.06E-19	8.89E-20	1.32E-19	1.21E-19	1.19E-19	9.93E-20	1.07E-19	8.68E-20	9.28E-20
Y-91m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	3.14E+09	5.04E+09	3.05E+09	3.35E+09	2.82E+09	3.10E+09	2.77E+09	4.50E+09	4.38E+09	4.28E+09	3.67E+09	4.06E+09	3.31E+09	3.54E+09
Pr-144m	5.59E+07	8.98E+07	5.43E+07	5.96E+07	5.01E+07	5.52E+07	4.93E+07	8.00E+07	7.79E+07	7.62E+07	6.53E+07	7.22E+07	5.88E+07	6.30E+07

MAXIMUM HEIGHT PLUME RISE FLAG = DAY_AND_NIGHT

READING FROM A WEATHER FILE WITH THE FOLLOWING HEADER:
Peach Bottom MACCS2 2006 Data
64 WD for SOAR CA Trials
Weather file uses 60 minute intervals
Weather file uses 64 wind directions
METEOROLOGICAL DATA FILE CONTAINS 602 PERIODS OF OBSERVED RAIN DATA.
ACCUMULATED RAIN MEASUREMENTS TOTALED 44.42 INCHES FOR THE YEAR.
MORNING LID HEIGHTS (M) FOR 4 SEASONS = 760 650 500 570
AFTERNOON LID HEIGHTS (M) FOR 4 SEASONS = 770 1450 1620 1140
NON-ZERO WINDSPEEDS LESS THAN 0.5 M/S ARE SET TO 0.5 M/S
NUMTRI= 984

* * * * METEOROLOGICAL BIN SUMMARY * * * *

BIN PRIORITIES
RI XX - RAIN INTENSITY I WITHIN THE INTERVAL ENDING AT XX
INTERVAL ENDPOINTS ARE IN KILOMETERS FROM THE ACCIDENT SITE, THE 5 INTERVAL ENDPOINTS ARE 3 6 11 21 32
RAIN INTENSITIES ARE IN MILLIMETERS OF RAIN PER HOUR, THE 3 INTENSITY BREAKPOINTS ARE 2.0 4.0 6.0
S V - INITIAL WEATHER CONDITIONS WITH STABILITY CLASS S AND WIND SPEED INTERVAL V
STABILITY CLASSES ARE B = A/B, D = C/D, E = E, AND F = F
WIND SPEED INTERVALS ARE IN METERS PER SECOND, 1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (5-7), 6 (GT 7)

WIND DIRECTION

METBIN			1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	B	3	0.017	0.010	0.004	0.011	0.004	0.000	0.003	0.008	0.004	0.010	0.010	0.007	0.004	0.007	0.011	0.016
2	B	4	0.017	0.014	0.012	0.017	0.005	0.007	0.014	0.017	0.010	0.012	0.005	0.017	0.010	0.014	0.026	0.029
3	D	1	0.000	0.000	0.000	0.000	0.000	0.011	0.011	0.011	0.000	0.022	0.011	0.022	0.022	0.000	0.000	0.033
4	D	2	0.017	0.021	0.013	0.010	0.011	0.010	0.010	0.008	0.006	0.008	0.008	0.011	0.011	0.021	0.019	0.013
5	D	3	0.012	0.021	0.011	0.012	0.011	0.007	0.009	0.005	0.011	0.005	0.009	0.005	0.012	0.007	0.012	0.027
6	D	4	0.004	0.005	0.003	0.004	0.008	0.003	0.003	0.005	0.011	0.004	0.008	0.011	0.013	0.011	0.013	0.015
7	D	5	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.005	0.000	0.010	0.005	0.038	0.072
8	D	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
9	E	1	0.023	0.029	0.031	0.004	0.008	0.037	0.012	0.006	0.016	0.010	0.019	0.014	0.004	0.008	0.010	0.010
10	E	2	0.028	0.030	0.021	0.011	0.014	0.011	0.012	0.014	0.010	0.014	0.017	0.025	0.022	0.015	0.022	0.030
11	E	3	0.010	0.013	0.008	0.009	0.005	0.004	0.004	0.005	0.009	0.022	0.027	0.030	0.031	0.023	0.034	0.040

12 E 4	0.008	0.011	0.003	0.006	0.006	0.000	0.008	0.006	0.011	0.008	0.011	0.008	0.028	0.017	0.034	0.056
13 F 1	0.018	0.012	0.041	0.022	0.014	0.024	0.031	0.010	0.027	0.031	0.025	0.024	0.020	0.022	0.027	0.025
14 F 2	0.005	0.004	0.016	0.005	0.009	0.019	0.012	0.027	0.029	0.057	0.060	0.077	0.096	0.055	0.067	0.056
15 F 3	0.000	0.007	0.000	0.000	0.007	0.000	0.007	0.000	0.022	0.058	0.080	0.066	0.117	0.139	0.095	0.102
16 F 4	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.250	0.000	0.250	0.000	0.250	0.000
17 R1 3	0.018	0.018	0.009	0.007	0.007	0.005	0.009	0.002	0.009	0.018	0.014	0.011	0.005	0.002	0.014	0.009
18 R1 6	0.000	0.000	0.016	0.016	0.032	0.032	0.016	0.016	0.000	0.032	0.032	0.000	0.048	0.016	0.000	0.016
19 R1 11	0.018	0.042	0.024	0.012	0.000	0.024	0.024	0.018	0.006	0.018	0.006	0.012	0.024	0.036	0.012	0.024
20 R1 21	0.021	0.034	0.004	0.004	0.004	0.008	0.004	0.025	0.000	0.021	0.021	0.008	0.000	0.013	0.000	0.021
21 R1 32	0.008	0.021	0.000	0.004	0.030	0.017	0.034	0.008	0.004	0.004	0.030	0.004	0.008	0.004	0.021	0.030
22 R2 3	0.026	0.000	0.017	0.035	0.000	0.000	0.009	0.009	0.009	0.000	0.009	0.009	0.009	0.000	0.009	0.000
23 R2 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
24 R2 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000	0.000	0.000
25 R2 21	0.000	0.000	0.000	0.062	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
26 R2 32	0.080	0.040	0.040	0.000	0.000	0.040	0.000	0.000	0.000	0.200	0.000	0.000	0.000	0.000	0.000	0.000
27 R3 3	0.026	0.000	0.051	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.051	0.051	0.000	0.000	0.000
28 R3 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
29 R3 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
30 R3 21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.400	0.000	0.000	0.000	0.000
31 R3 32	0.000	0.250	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.250
32 R4 3	0.000	0.000	0.000	0.000	0.000	0.000	0.029	0.029	0.029	0.000	0.000	0.000	0.000	0.000	0.000	0.000
33 R4 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
34 R4 11	0.143	0.143	0.000	0.000	0.143	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.143	0.000
35 R4 21	0.000	0.222	0.111	0.222	0.000	0.000	0.000	0.000	0.000	0.000	0.111	0.000	0.000	0.000	0.000	0.000
36 R4 32	0.067	0.000	0.133	0.000	0.067	0.067	0.067	0.000	0.067	0.000	0.000	0.000	0.000	0.067	0.000	0.000
37 ALL	0.015	0.016	0.013	0.009	0.009	0.010	0.011	0.010	0.012	0.018	0.020	0.021	0.024	0.019	0.024	0.029

METBIN	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1 B 3	0.003	0.013	0.007	0.006	0.010	0.010	0.007	0.006	0.007	0.016	0.010	0.027	0.040	0.035	0.023	0.038
2 B 4	0.005	0.021	0.024	0.038	0.055	0.043	0.041	0.060	0.036	0.050	0.074	0.088	0.064	0.019	0.014	0.005
3 D 1	0.033	0.022	0.000	0.022	0.000	0.000	0.022	0.022	0.011	0.011	0.000	0.022	0.011	0.011	0.000	0.033
4 D 2	0.015	0.011	0.019	0.011	0.025	0.015	0.017	0.021	0.025	0.010	0.023	0.019	0.025	0.029	0.017	0.029
5 D 3	0.011	0.016	0.018	0.021	0.039	0.034	0.032	0.018	0.019	0.048	0.060	0.055	0.046	0.032	0.028	0.048
6 D 4	0.017	0.023	0.057	0.035	0.058	0.048	0.085	0.071	0.065	0.081	0.062	0.039	0.057	0.030	0.009	0.011
7 D 5	0.019	0.024	0.034	0.053	0.072	0.077	0.058	0.115	0.125	0.072	0.096	0.034	0.034	0.014	0.000	0.000
8 D 6	0.031	0.000	0.031	0.031	0.031	0.000	0.000	0.000	0.000	0.469	0.375	0.000	0.000	0.000	0.000	0.000
9 E 1	0.006	0.006	0.014	0.012	0.008	0.004	0.012	0.019	0.014	0.016	0.008	0.004	0.004	0.014	0.004	0.035
10 E 2	0.023	0.019	0.026	0.021	0.020	0.013	0.022	0.020	0.012	0.021	0.026	0.010	0.017	0.009	0.013	0.023
11 E 3	0.045	0.050	0.058	0.044	0.039	0.044	0.031	0.025	0.021	0.040	0.028	0.031	0.021	0.014	0.025	0.016
12 E 4	0.031	0.023	0.045	0.025	0.068	0.071	0.065	0.054	0.054	0.045	0.028	0.011	0.011	0.020	0.014	0.025
13 F 1	0.016	0.016	0.014	0.022	0.010	0.016	0.022	0.014	0.020	0.016	0.033	0.002	0.000	0.002	0.002	0.031
14 F 2	0.031	0.031	0.028	0.017	0.029	0.011	0.020	0.015	0.023	0.013	0.013	0.012	0.013	0.001	0.004	0.025
15 F 3	0.044	0.044	0.015	0.015	0.022	0.007	0.007	0.029	0.000	0.015	0.000	0.007	0.007	0.000	0.007	0.007
16 F 4	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
17 R1 3	0.005	0.007	0.007	0.014	0.009	0.009	0.009	0.011	0.016	0.018	0.025	0.027	0.020	0.018	0.007	0.039
18 R1 6	0.000	0.000	0.016	0.000	0.016	0.016	0.000	0.000	0.000	0.000	0.016	0.000	0.032	0.000	0.016	0.032
19 R1 11	0.000	0.018	0.006	0.006	0.006	0.000	0.018	0.012	0.024	0.006	0.000	0.012	0.012	0.012	0.006	0.012
20 R1 21	0.004	0.008	0.000	0.017	0.013	0.004	0.021	0.013	0.021	0.017	0.034	0.013	0.021	0.017	0.008	0.025
21 R1 32	0.008	0.017	0.008	0.000	0.013	0.008	0.021	0.000	0.013	0.021	0.004	0.004	0.017	0.021	0.034	0.021
22 R2 3	0.000	0.000	0.000	0.000	0.009	0.000	0.000	0.009	0.000	0.009	0.035	0.009	0.009	0.009	0.000	0.017
23 R2 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
24 R2 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000
25 R2 21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.063	0.000	0.000	0.063	0.000	0.000	0.000
26 R2 32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
27 R3 3	0.000	0.000	0.000	0.000	0.000	0.000	0.026	0.000	0.000	0.103	0.026	0.026	0.026	0.051	0.000	0.103
28 R3 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
29 R3 11	0.000	0.000	0.000	0.000	0.333	0.000	0.000	0.000	0.000	0.333	0.000	0.000	0.000	0.000	0.000	0.000
30 R3 21	0.000	0.000	0.000	0.000	0.000	0.200	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
31 R3 32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
32 R4 3	0.029	0.000	0.000	0.000	0.000	0.029	0.000	0.000	0.000	0.059	0.088	0.029	0.000	0.029	0.059	0.000
33 R4 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
34 R4 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
35 R4 21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
36 R4 32	0.000	0.000	0.000	0.133	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
37 ALL	0.018	0.020	0.025	0.021	0.028	0.023	0.028	0.026	0.025	0.032	0.032	0.024	0.025	0.017	0.013	0.025

METBIN	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1 B 3	0.024	0.048	0.045	0.049	0.021	0.018	0.017	0.008	0.007	0.020	0.024	0.018	0.024	0.020	0.023	0.028
2 B 4	0.000	0.019	0.010	0.010	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
3 D 1	0.000	0.022	0.000	0.044	0.033	0.022	0.033	0.011	0.000	0.022	0.033	0.022	0.044	0.022	0.044	0.056
4 D 2	0.019	0.044	0.040	0.031	0.029	0.013	0.015	0.006	0.006	0.015	0.010	0.008	0.013	0.004	0.013	0.013
5 D 3	0.027	0.016	0.032	0.007	0.007	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
6 D 4	0.011	0.012	0.024	0.001	0.001	0.000	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
7 D 5	0.000	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
8 D 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
9 E 1	0.012	0.008	0.010	0.023	0.006	0.004	0.006	0.006	0.008	0.012	0.008	0.004	0.027	0.019	0.029	0.025
10 E 2	0.011	0.011	0.017	0.006	0.009	0.008	0.003	0.002	0.000	0.005	0.002	0.002	0.002	0.001	0.002	0.008
11 E 3	0.009	0.004	0.008	0.006	0.004	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
12 E 4	0.017	0.011	0.006	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
13 F 1	0.000	0.006	0.002	0.018	0.002	0.000	0.002	0.000	0.000	0.002	0.008	0.002	0.014	0.004	0.016	0.022
14 F 2	0.001	0.000	0.000	0.003	0.001	0.001	0.000	0.001	0.000	0.000	0.001	0.004	0.000	0.000	0.001	0.004
15 F 3	0.007	0.000	0.000	0.007	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
16 F 4	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
17 R1 3	0.025	0.075	0.018	0.011	0.020	0.011	0.016	0.007	0.005	0.020	0.020	0.011	0.011	0.011	0.005	0.016
18 R1 6	0.000	0.016	0.032	0.000	0.000	0.016	0.000	0.032	0.032	0.000	0.048	0.032	0.016	0.000	0.032	0.016
19 R1 11	0.036	0.018	0.018	0.012	0.012	0.006	0.006	0.006	0.012	0.006	0.012	0.006	0.006	0.006	0.012	0.018
20 R1 21	0.017	0.034	0.013	0.017	0.004	0.008	0.004	0.008	0.004	0.004	0.013	0.004	0.013	0.008	0.025	0.030
21 R1 32	0.008	0.025	0.021	0.013	0.008	0.000	0.008	0.013	0.004	0.004	0.000	0.008	0.004	0.000	0.021	0.030
22 R2 3	0.000	0.017	0.035	0.026	0.026	0.017	0.017	0.026	0.000	0.017	0.009	0.009	0.009	0.017	0.043	0.052
23 R2 6	0.250	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.000	0.000	0.000	0.000
24 R2 11	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
25 R2 21	0.000	0.063	0.000	0.000	0.063	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
26 R2 32	0.000	0.040	0.000	0.040	0.040	0.040	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
27 R3 3	0.026	0.000	0.000	0.000	0.026	0.000	0.026	0.000	0.000	0.026	0.000	0.026	0.000	0.000	0.000	0.000
28 R3 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
29 R3 11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
30 R3 21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
31 R3 32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
32 R4 3	0.000	0.029	0.029	0.000	0.029	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
33 R4 6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	1.000	0.000	0.000	0.000	0.000	0.000
34 R4 11	0.000	0.000	0.143	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.143	0.000	0.143
35 R4 21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.111	0.111	0.000	0.000	0.000
36 R4 32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
37 ALL	0.012	0.019	0.017	0.013	0.009	0.006	0.005	0.003	0.002	0.006	0.006	0.005	0.007	0.005	0.008	0.011

METBIN	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	TOTAL	PER CENT
1 B 3	0.011	0.031	0.021	0.010	0.020	0.007	0.017	0.010	0.013	0.010	0.010	0.014	0.008	0.016	0.008	0.016	708	8.0822
2 B 4	0.000	0.000	0.000	0.000	0.000	0.002	0.002	0.000	0.000	0.002	0.000	0.005	0.005	0.019	0.036	0.026	419	4.7831
3 D 1	0.000	0.044	0.022	0.011	0.011	0.011	0.033	0.011	0.000	0.011	0.022	0.011	0.022	0.000	0.011	0.000	90	1.0274
4 D 2	0.017	0.006	0.002	0.010	0.010	0.011	0.017	0.015	0.011	0.011	0.017	0.019	0.019	0.015	0.013	0.019	524	5.9817

5	D	3	0.000	0.000	0.000	0.000	0.000	0.005	0.009	0.000	0.016	0.027	0.021	0.023	0.028	0.018	0.028	0.030	565	6.4498
6	D	4	0.000	0.000	0.000	0.000	0.000	0.001	0.001	0.003	0.000	0.008	0.001	0.001	0.009	0.020	0.015	0.020	743	8.4817
7	D	5	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.010	0.005	0.014	0.010	208	2.3744
8	D	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.031	0.000	0.000	32	0.3653
9	E	1	0.031	0.027	0.021	0.019	0.014	0.016	0.031	0.023	0.010	0.016	0.041	0.023	0.027	0.027	0.031	0.010	486	5.5479
10	E	2	0.005	0.017	0.011	0.011	0.010	0.012	0.016	0.021	0.020	0.020	0.029	0.025	0.031	0.028	0.031	0.033	1029	11.7466
11	E	3	0.000	0.003	0.000	0.000	0.003	0.003	0.001	0.001	0.010	0.018	0.019	0.019	0.019	0.025	0.025	0.016	773	8.8242
12	E	4	0.000	0.000	0.000	0.000	0.000	0.003	0.003	0.000	0.006	0.003	0.003	0.008	0.014	0.014	0.034	0.065	354	4.0411
13	F	1	0.045	0.035	0.043	0.006	0.061	0.012	0.025	0.004	0.000	0.008	0.012	0.024	0.006	0.006	0.012	0.000	510	5.8219
14	F	2	0.004	0.013	0.009	0.007	0.011	0.005	0.009	0.004	0.001	0.009	0.000	0.005	0.007	0.005	0.007	0.003	750	8.5616
15	F	3	0.000	0.000	0.000	0.000	0.007	0.000	0.000	0.000	0.000	0.015	0.000	0.000	0.015	0.007	0.007	0.007	137	1.5639
16	F	4	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	4	0.0457
17	R1	3	0.007	0.005	0.005	0.023	0.020	0.011	0.011	0.023	0.018	0.023	0.029	0.027	0.034	0.025	0.039	0.020	441	5.0342
18	R1	6	0.063	0.000	0.016	0.016	0.000	0.000	0.000	0.063	0.000	0.000	0.048	0.032	0.016	0.016	0.032	0.016	63	0.7192
19	R1	11	0.030	0.006	0.018	0.018	0.018	0.018	0.012	0.030	0.030	0.012	0.042	0.030	0.024	0.018	0.018	0.024	165	1.8836
20	R1	21	0.025	0.013	0.013	0.017	0.021	0.004	0.025	0.013	0.038	0.059	0.017	0.013	0.017	0.017	0.030	0.042	236	2.6941
21	R1	32	0.025	0.017	0.008	0.013	0.038	0.021	0.021	0.017	0.038	0.051	0.038	0.017	0.008	0.021	0.034	0.025	237	2.7055
22	R2	3	0.000	0.000	0.009	0.017	0.035	0.000	0.017	0.052	0.043	0.017	0.009	0.052	0.070	0.026	0.043	0.043	115	1.3128
23	R2	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.000	0.250	4	0.0457
24	R2	11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.125	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.250	8	0.0913
25	R2	21	0.000	0.000	0.000	0.000	0.063	0.063	0.000	0.000	0.000	0.000	0.000	0.188	0.125	0.063	0.063	0.000	16	0.1826
26	R2	32	0.000	0.000	0.040	0.000	0.000	0.000	0.000	0.080	0.000	0.000	0.000	0.080	0.120	0.040	0.000	0.080	25	0.2854
27	R3	3	0.000	0.000	0.051	0.000	0.000	0.026	0.026	0.051	0.000	0.077	0.000	0.000	0.000	0.051	0.026	0.026	39	0.4452
28	R3	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	1.000	1	0.0114
29	R3	11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.333	0.000	0.000	3	0.0342
30	R3	21	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.200	0.000	0.000	0.200	0.000	5	0.0571
31	R3	32	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.000	0.000	4	0.0457
32	R4	3	0.059	0.000	0.000	0.000	0.000	0.000	0.029	0.029	0.029	0.000	0.029	0.088	0.000	0.029	0.147	0.059	34	0.3881
33	R4	6	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	1	0.0114
34	R4	11	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	7	0.0799
35	R4	21	0.000	0.111	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	9	0.1027
36	R4	32	0.133	0.067	0.000	0.000	0.000	0.000	0.000	0.067	0.000	0.000	0.000	0.000	0.000	0.067	0.000	0.000	15	0.1712
37	ALL		0.010	0.012	0.009	0.007	0.013	0.008	0.012	0.011	0.011	0.016	0.016	0.017	0.018	0.019	0.023	0.021	8760	100.0000

* * * * METEOROLOGICAL BIN SUMMARY * * * *

BIN PRIORITIES

RI XX - RAIN INTENSITY I WITHIN THE INTERVAL ENDING AT XX

INTERVAL ENDPOINTS ARE IN KILOMETERS FROM THE ACCIDENT SITE, THE 5 INTERVAL ENDPOINTS ARE 3 6 11 21 32

RAIN INTENSITIES ARE IN MILLIMETERS OF RAIN PER HOUR, THE 3 INTENSITY BREAKPOINTS ARE 2.0 4.0 6.0

S V - INITIAL WEATHER CONDITIONS WITH STABILITY CLASS S AND WIND SPEED INTERVAL V

STABILITY CLASSES ARE B = A/B, D = C/D, E = E, AND F = F

WIND SPEED INTERVALS ARE IN METERS PER SECOND (M/S), 1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (5-7), 6 (GT 7)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1 B 3	12	7	3	8	3	0	2	6	3	7	7	5	3	5	8	11
2 B 4	7	6	5	7	2	3	6	7	4	5	2	7	4	6	11	12
3 D 1	0	0	0	0	0	1	1	1	0	2	1	2	2	0	0	3
4 D 2	9	11	7	5	6	5	5	4	3	4	4	6	6	11	10	7
5 D 3	7	12	6	7	6	4	5	3	6	3	5	3	7	4	7	15
6 D 4	3	4	2	3	6	2	2	4	8	3	6	8	10	8	10	11
7 D 5	0	0	0	0	0	0	0	0	0	0	1	0	2	1	8	15
8 D 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
9 E 1	11	14	15	2	4	18	6	3	8	5	9	7	2	4	5	5
10 E 2	29	31	22	11	14	11	12	14	10	14	18	26	23	15	23	31
11 E 3	8	10	6	7	4	3	3	4	7	17	21	23	24	18	26	31
12 E 4	3	4	1	2	2	0	3	2	4	3	4	3	10	6	12	20
13 F 1	9	6	21	11	7	12	16	5	14	16	13	12	10	11	14	13
14 F 2	4	3	12	4	7	14	9	20	22	43	45	58	72	41	50	42
15 F 3	0	1	0	0	1	0	1	0	3	8	11	9	16	19	13	14
16 F 4	0	0	0	0	0	0	0	0	0	1	1	0	1	0	1	0
17 R1 3	8	8	4	3	3	2	4	1	4	8	6	5	2	1	6	4
18 R1 6	0	0	1	1	2	2	1	1	0	2	2	0	3	1	0	1
19 R1 11	3	7	4	2	0	4	4	3	1	3	1	2	4	6	2	4
20 R1 21	5	8	1	1	1	2	1	6	0	5	5	2	0	3	0	5
21 R1 32	2	5	0	1	7	4	8	2	1	1	7	1	2	1	5	7
22 R2 3	3	0	2	4	0	0	1	1	1	0	1	1	1	0	1	0
23 R2 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
24 R2 11	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	0
25 R2 21	0	0	0	1	0	0	2	0	0	0	0	0	0	0	0	0
26 R2 32	2	1	1	0	0	1	0	0	0	5	0	0	0	0	0	0
27 R3 3	1	0	2	0	0	0	0	0	0	0	0	2	2	0	0	0
28 R3 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
29 R3 11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
30 R3 21	0	0	0	0	0	0	0	0	0	0	0	2	0	0	0	0
31 R3 32	0	1	0	0	0	0	0	0	0	0	0	0	0	1	0	1
32 R4 3	0	0	0	0	0	0	1	1	1	0	0	0	0	0	0	0
33 R4 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
34 R4 11	1	1	0	0	1	0	0	0	0	0	0	0	0	0	1	0
35 R4 21	0	2	1	2	0	0	0	0	0	0	1	0	0	0	0	0
36 R4 32	1	0	2	0	1	1	1	0	1	0	0	0	0	1	0	0

METBIN	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1 B 3	2	9	5	4	7	7	5	4	5	11	7	19	28	25	16	27
2 B 4	2	9	10	16	23	18	17	25	15	21	31	37	27	8	6	2
3 D 1	3	2	0	2	0	0	2	2	1	1	0	2	1	1	0	3
4 D 2	8	6	10	6	13	8	9	11	13	5	12	10	13	15	9	15
5 D 3	6	9	10	12	22	19	18	10	11	27	34	31	26	18	16	27
6 D 4	13	17	42	26	43	36	63	53	48	60	46	29	42	22	7	8
7 D 5	4	5	7	11	15	16	12	24	26	15	20	7	7	3	0	0
8 D 6	1	0	1	1	1	0	0	0	0	15	12	0	0	0	0	0
9 E 1	3	3	7	6	4	2	6	9	7	8	4	2	2	7	2	17
10 E 2	24	20	27	22	21	13	23	21	12	22	27	10	18	9	13	24
11 E 3	35	39	45	34	30	34	24	19	16	31	22	24	16	11	19	12
12 E 4	11	8	16	9	24	25	23	19	19	16	10	4	4	7	5	9
13 F 1	8	8	7	11	5	8	11	7	10	8	17	1	0	1	1	16
14 F 2	23	23	21	13	22	8	15	11	17	10	10	9	10	1	3	19
15 F 3	6	6	2	2	3	1	1	4	0	2	0	1	1	0	1	1
16 F 4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
17 R1 3	2	3	3	6	4	4	4	5	7	8	11	12	9	8	3	17
18 R1 6	0	0	1	0	1	1	0	0	0	0	1	0	2	0	1	2
19 R1 11	0	3	1	1	1	0	3	2	4	1	0	2	2	2	1	2
20 R1 21	1	2	0	4	3	1	5	3	5	4	8	3	5	4	2	6
21 R1 32	2	4	2	0	3	2	5	0	3	5	1	1	4	5	8	5
22 R2 3	0	0	0	0	1	0	0	1	0	1	4	1	1	1	0	2
23 R2 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
24 R2 11	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0

25 R2 21	0	0	0	0	0	0	0	0	0	1	0	0	1	0	0	0
26 R2 32	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
27 R3 3	0	0	0	0	0	0	1	0	0	4	1	1	1	2	0	4
28 R3 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
29 R3 11	0	0	0	0	1	0	0	0	0	1	0	0	0	0	0	0
30 R3 21	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0
31 R3 32	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
32 R4 3	1	0	0	0	0	1	0	0	0	2	3	1	0	1	2	0
33 R4 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
34 R4 11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
35 R4 21	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
36 R4 32	0	0	0	2	0	0	0	0	0	0	0	0	0	0	0	0

METBIN	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1 B 3	17	34	32	35	15	13	12	6	5	14	17	13	17	14	16	20
2 B 4	0	8	4	4	1	0	0	0	0	0	0	0	0	0	0	0
3 D 1	0	2	0	4	3	2	3	1	0	2	3	2	4	2	4	5
4 D 2	10	23	21	16	15	7	8	3	3	8	5	4	7	2	7	7
5 D 3	15	9	18	4	4	3	0	0	0	0	0	0	0	0	0	0
6 D 4	8	9	18	1	1	0	1	0	0	0	0	0	0	0	0	0
7 D 5	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0
8 D 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
9 E 1	6	4	5	11	3	2	3	3	4	6	4	2	13	9	14	12
10 E 2	11	11	17	6	9	8	3	2	0	5	2	2	2	1	2	8
11 E 3	7	3	6	5	3	1	0	0	0	0	0	0	0	0	0	0
12 E 4	6	4	2	0	0	0	0	0	0	0	0	0	0	0	0	0
13 F 1	0	3	1	9	1	0	1	0	0	1	4	1	7	2	8	11
14 F 2	1	0	0	2	1	1	0	1	0	0	1	3	0	0	1	3
15 F 3	1	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0
16 F 4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
17 R1 3	11	33	8	5	9	5	7	3	2	9	9	5	5	5	2	7
18 R1 6	0	1	2	0	0	1	0	2	2	0	3	2	1	0	2	1
19 R1 11	6	3	3	2	2	1	1	1	2	1	2	1	1	1	2	3
20 R1 21	4	8	3	4	1	2	1	2	1	1	3	1	3	2	6	7
21 R1 32	2	6	5	3	2	0	2	3	1	1	0	2	1	0	5	7
22 R2 3	0	2	4	3	3	2	2	3	0	2	1	1	1	2	5	6
23 R2 6	1	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0
24 R2 11	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0
25 R2 21	0	1	0	0	1	0	0	0	0	0	0	0	0	0	0	0
26 R2 32	0	1	0	1	1	1	0	0	0	0	0	0	0	0	0	0
27 R3 3	1	0	0	0	1	0	1	0	0	1	0	1	0	0	0	0
28 R3 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
29 R3 11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
30 R3 21	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
31 R3 32	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
32 R4 3	0	1	1	0	1	0	0	0	0	0	0	0	0	0	0	0
33 R4 6	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0
34 R4 11	0	0	1	0	0	0	0	0	0	0	0	0	0	1	0	1
35 R4 21	0	0	0	0	0	0	0	0	0	0	0	1	1	0	0	0
36 R4 32	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

METBIN	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	TOTAL	PER CENT
1 B 3	8	22	15	7	14	5	12	7	9	7	7	10	6	11	6	11	708	8.0822
2 B 4	0	0	0	0	0	1	1	0	0	1	0	2	2	8	15	11	419	4.7831
3 D 1	0	4	2	1	1	1	3	1	0	1	2	1	2	0	1	0	90	1.0274
4 D 2	9	3	1	5	5	6	9	8	6	6	9	10	10	8	7	10	524	5.9817
5 D 3	0	0	0	0	0	3	5	0	9	15	12	13	16	10	16	17	565	6.4498
6 D 4	0	0	0	0	0	1	1	2	0	6	1	1	7	15	11	15	743	8.4817
7 D 5	0	0	0	0	0	0	0	0	0	0	0	0	2	1	3	2	208	2.3744
8 D 6	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	32	0.3653
9 E 1	15	13	10	9	7	8	15	11	5	8	20	11	13	13	15	5	486	5.5479
10 E 2	5	18	11	11	10	12	16	22	21	21	30	26	32	29	32	34	1029	11.7466
11 E 3	0	2	0	0	2	2	1	1	8	14	15	15	15	19	19	12	773	8.8242
12 E 4	0	0	0	0	0	1	1	0	2	1	1	3	5	5	12	23	354	4.0411
13 F 1	23	18	22	3	31	6	13	2	0	4	6	12	3	3	6	0	510	5.8219
14 F 2	3	10	7	5	8	4	7	3	1	7	0	4	5	4	5	2	750	8.5616
15 F 3	0	0	0	0	1	0	0	0	0	2	0	0	2	1	1	1	137	1.5639
16 F 4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4	0.0457
17 R1 3	3	2	2	10	9	5	5	10	8	10	13	12	15	11	17	9	441	5.0342
18 R1 6	4	0	1	1	0	0	0	4	0	0	3	2	1	1	2	1	63	0.7192
19 R1 11	5	1	3	3	3	3	2	5	5	2	7	5	4	3	3	4	165	1.8836
20 R1 21	6	3	3	4	5	1	6	3	9	14	4	3	4	4	7	10	236	2.6941
21 R1 32	6	4	2	3	9	5	5	4	9	12	9	4	2	5	8	6	237	2.7055
22 R2 3	0	0	1	2	4	0	2	6	5	2	1	6	8	3	5	5	115	1.3128
23 R2 6	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	1	4	0.0457
24 R2 11	0	0	0	0	0	0	0	1	0	0	0	0	0	2	0	2	8	0.0913
25 R2 21	0	0	0	0	1	1	0	0	0	0	0	3	2	1	1	0	16	0.1826
26 R2 32	0	0	1	0	0	0	0	2	0	0	0	2	3	1	0	2	25	0.2854
27 R3 3	0	0	2	0	0	1	1	2	0	3	0	0	0	2	1	1	39	0.4452
28 R3 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	0.0114
29 R3 11	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	3	0.0342
30 R3 21	0	0	0	0	0	0	0	0	0	0	0	1	0	0	1	0	5	0.0571
31 R3 32	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	4	0.0457
32 R4 3	2	0	0	0	0	0	1	1	1	0	1	3	0	1	5	2	34	0.3881
33 R4 6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0.0114
34 R4 11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	7	0.0799
35 R4 21	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	9	0.1027
36 R4 32	2	1	0	0	0	0	0	1	0	0	0	0	0	1	0	0	15	0.1712

* * * * SUMMARIES * * * *

R	26	33	18	15	15	16	23	15	10	24	23	15	14	14	15	22
B	19	13	8	15	5	3	8	13	7	12	9	12	7	11	19	23
D	19	27	15	15	18	12	13	12	17	12	17	19	27	24	35	51
E	51	59	44	22	24	32	24	23	29	39	52	59	59	43	66	87
F	13	10	33	15	15	26	26	25	39	68	70	79	99	71	78	69
1	20	20	37	14	11	31	23	9	22	23	23	21	14	15	19	24
2	45	46	42	20	27	30	27	38	37	67	73	92	103	67	86	85
3	24	29	13	21	14	7	10	13	17	29	38	38	48	46	51	63
4	11	14	8	10	9	4	11	11	16	9	13	17	22	15	27	39
5	2	0	0	2	1	1	0	1	0	2	1	1	5	6	14	19
6	0	0	0	0	0	0	0	1	0	1	0	0	0	0	1	0

R	6	12	7	13	14	10	18	11	19	27	30	21	25	23	17	38
B	4	18	15	20	30	25	22	29	20	32	38	56	55	33	22	29
D	35	39	70	58	94	79	104	100	99	123	124	79	89	59	32	53
E	73	70	95	71	79	74	76	68	54	77	63	40	40	34	39	62
F	37	37	30	26	30	17	27	22	27	20	27	11	11	2	5	36
1	14	14	15	19	10	10	20	18	19	17	21	5	3	10	3	36

2	56	51	59	43	57	32	48	45	43	40	51	31	48	32	30	64
3	48	60	60	50	60	58	46	35	30	68	61	73	64	46	47	61
4	25	31	67	47	79	74	100	90	75	95	79	67	72	37	18	19
5	5	8	8	15	24	21	15	31	32	17	27	10	8	3	0	0
6	1	0	1	1	3	0	0	0	1	15	13	0	0	0	0	0

R	25	56	28	18	21	12	14	14	8	15	20	14	13	11	22	32
B	17	42	36	39	16	13	12	6	5	14	17	13	17	14	16	20
D	33	44	57	25	23	12	12	4	3	10	8	6	11	4	11	12
E	30	22	30	22	15	11	6	5	4	11	6	4	15	10	16	20
F	2	3	1	12	2	1	1	1	0	1	5	4	7	2	9	14
1	6	9	6	29	7	4	8	4	4	10	11	8	25	14	26	28
2	34	41	50	39	37	27	21	12	8	26	25	19	25	16	26	38
3	28	39	44	25	10	6	1	0	0	0	0	0	0	0	0	0
4	14	21	24	5	2	0	1	0	0	0	0	0	0	0	0	0
5	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

R	28	12	15	23	31	16	22	39	37	43	38	41	40	37	50	44	1428	16.3014
B	8	22	15	7	14	6	13	7	9	8	7	12	8	19	21	22	1127	12.8653
D	9	7	3	6	6	11	18	11	15	28	24	25	37	35	38	44	2162	24.6804
E	20	33	21	20	19	23	33	34	36	44	66	55	65	66	78	74	2642	30.1598
F	26	28	29	8	40	10	20	5	1	13	6	16	10	8	12	3	1401	15.9932
1	38	38	35	13	39	15	33	15	5	13	31	24	18	16	22	5	1119	12.7740
2	24	50	33	27	32	23	40	38	32	37	40	42	50	43	47	47	2664	30.4110
3	1	2	0	1	8	9	8	2	22	35	30	36	36	39	39	40	1789	20.4224
4	0	0	0	0	0	3	3	2	2	8	2	6	13	27	38	46	1428	16.3014
5	0	0	0	0	0	0	0	0	0	0	0	0	3	2	3	5	293	3.3447
6	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	39	0.4452

* * * * * BIN WINDROSE SUMMARY * * * * *

BIN	DIRECTION															
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	0.017	0.010	0.004	0.011	0.004	0.000	0.003	0.008	0.004	0.010	0.010	0.007	0.004	0.007	0.011	0.016
2	0.017	0.014	0.012	0.017	0.005	0.007	0.014	0.017	0.010	0.012	0.005	0.017	0.010	0.014	0.026	0.029
3	0.000	0.000	0.000	0.000	0.000	0.011	0.011	0.011	0.000	0.022	0.011	0.022	0.022	0.000	0.000	0.033
4	0.017	0.021	0.013	0.010	0.011	0.010	0.010	0.008	0.006	0.008	0.008	0.011	0.011	0.021	0.019	0.013
5	0.012	0.021	0.011	0.012	0.011	0.007	0.009	0.005	0.011	0.005	0.009	0.005	0.012	0.007	0.012	0.027
6	0.004	0.005	0.003	0.004	0.008	0.003	0.003	0.005	0.011	0.004	0.008	0.011	0.013	0.011	0.013	0.015
7	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.005	0.000	0.010	0.005	0.038	0.072
8	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
9	0.023	0.029	0.031	0.004	0.008	0.037	0.012	0.006	0.016	0.010	0.019	0.014	0.004	0.008	0.010	0.010
10	0.028	0.030	0.021	0.011	0.014	0.011	0.012	0.014	0.010	0.014	0.017	0.025	0.022	0.015	0.022	0.030
11	0.010	0.013	0.008	0.009	0.005	0.004	0.004	0.005	0.009	0.022	0.027	0.030	0.031	0.023	0.034	0.040
12	0.008	0.011	0.003	0.006	0.006	0.000	0.008	0.006	0.011	0.008	0.011	0.008	0.028	0.017	0.034	0.056
13	0.018	0.012	0.041	0.022	0.014	0.024	0.031	0.010	0.027	0.031	0.025	0.024	0.020	0.022	0.027	0.025
14	0.005	0.004	0.016	0.005	0.009	0.019	0.012	0.027	0.029	0.057	0.060	0.077	0.096	0.055	0.067	0.056
15	0.000	0.007	0.000	0.000	0.007	0.000	0.007	0.000	0.022	0.058	0.080	0.066	0.117	0.139	0.095	0.102
16	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.250	0.250	0.000	0.250	0.000	0.250	0.000
17	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
18	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
19	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
20	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
21	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
22	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
23	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
24	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
25	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
26	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
27	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
28	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
29	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
30	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
31	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
32	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
33	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
34	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
35	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
36	0.018	0.023	0.013	0.011	0.011	0.011	0.016	0.011	0.007	0.017	0.016	0.011	0.010	0.010	0.011	0.015
37	0.015	0.016	0.013	0.009	0.009	0.010	0.011	0.010	0.012	0.018	0.020	0.021	0.024	0.019	0.024	0.029
	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	0.003	0.013	0.007	0.006	0.010	0.010	0.007	0.006	0.007	0.016	0.010	0.027	0.040	0.035	0.023	0.038
2	0.005	0.021	0.024	0.038	0.055	0.043	0.041	0.060	0.036	0.050	0.074	0.088	0.064	0.019	0.014	0.005
3	0.033	0.022	0.000	0.022	0.000	0.000	0.022	0.022	0.011	0.011	0.000	0.022	0.011	0.011	0.000	0.033
4	0.015	0.011	0.019	0.011	0.025	0.015	0.017	0.021	0.025	0.010	0.023	0.019	0.025	0.029	0.017	0.029
5	0.011	0.016	0.018	0.021	0.039	0.034	0.032	0.018	0.019	0.048	0.060	0.055	0.046	0.032	0.028	0.048
6	0.017	0.023	0.057	0.035	0.058	0.048	0.085	0.071	0.065	0.081	0.062	0.039	0.057	0.030	0.009	0.011
7	0.019	0.024	0.034	0.053	0.072	0.077	0.058	0.115	0.125	0.072	0.096	0.034	0.034	0.014	0.000	0.000
8	0.031	0.000	0.031	0.031	0.031	0.000	0.000	0.000	0.000	0.469	0.375	0.000	0.000	0.000	0.000	0.000
9	0.006	0.006	0.014	0.012	0.008	0.004	0.012	0.019	0.014	0.016	0.008	0.004	0.004	0.014	0.004	0.035
10	0.023	0.019	0.026	0.021	0.020	0.013	0.022	0.020	0.012	0.021	0.026	0.010	0.017	0.009	0.013	0.023
11	0.045	0.050	0.058	0.044	0.039	0.044	0.031	0.025	0.021	0.040	0.028	0.031	0.021	0.014	0.025	0.016
12	0.031	0.023	0.045	0.025	0.068	0.071	0.065	0.054	0.054	0.045	0.028	0.011	0.011	0.020	0.014	0.025
13	0.016	0.016	0.014	0.022	0.010	0.016	0.022	0.014	0.020	0.016	0.033	0.002	0.000	0.002	0.002	0.031
14	0.031	0.031	0.028	0.017	0.029	0.011	0.020	0.015	0.023	0.013	0.013	0.012	0.013	0.001	0.004	0.025
15	0.044	0.044	0.015	0.015	0.022	0.007	0.007	0.029	0.000	0.015	0.000	0.007	0.007	0.000	0.007	0.007
16	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
17	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
18	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
19	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
20	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
21	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
22	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
23	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
24	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
25	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
26	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
27	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
28	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
29	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
30	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
31	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
32	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
33	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
34	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
35	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
36	0.004	0.008	0.005	0.009	0.010	0.007	0.013	0.008	0.013	0.019	0.021	0.015	0.018	0.016	0.012	0.027
37	0.018	0.020	0.025	0.021	0.028	0.023	0.028	0.026	0.025	0.032	0.032	0.024	0.025	0.017	0.013	0.025

	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	0.024	0.048	0.045	0.049	0.021	0.018	0.017	0.008	0.007	0.020	0.024	0.018	0.024	0.020	0.023	0.028
2	0.000	0.019	0.010	0.010	0.002	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
3	0.000	0.022	0.000	0.044	0.033	0.022	0.033	0.011	0.000	0.022	0.033	0.022	0.044	0.022	0.044	0.056
4	0.019	0.044	0.040	0.031	0.029	0.013	0.015	0.006	0.006	0.015	0.010	0.008	0.013	0.004	0.013	0.013
5	0.027	0.016	0.032	0.007	0.007	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
6	0.011	0.012	0.024	0.001	0.001	0.000	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
7	0.000	0.005	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
8	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
9	0.012	0.008	0.010	0.023	0.006	0.004	0.006	0.006	0.008	0.012	0.008	0.004	0.027	0.019	0.029	0.025
10	0.011	0.011	0.017	0.006	0.009	0.008	0.003	0.002	0.000	0.005	0.002	0.002	0.002	0.001	0.002	0.008
11	0.009	0.004	0.008	0.006	0.004	0.001	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
12	0.017	0.011	0.006	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
13	0.000	0.006	0.002	0.018	0.002	0.000	0.002	0.000	0.000	0.002	0.008	0.002	0.014	0.004	0.016	0.022
14	0.001	0.000	0.000	0.003	0.001	0.001	0.000	0.001	0.000	0.000	0.001	0.004	0.000	0.000	0.001	0.004
15	0.007	0.000	0.000	0.007	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
16	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
17	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
18	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
19	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
20	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
21	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
22	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
23	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
24	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
25	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
26	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
27	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
28	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
29	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
30	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
31	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
32	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
33	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
34	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
35	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
36	0.018	0.039	0.020	0.013	0.015	0.008	0.010	0.010	0.006	0.011	0.014	0.010	0.009	0.008	0.015	0.022
37	0.012	0.019	0.017	0.013	0.009	0.006	0.005	0.003	0.002	0.006	0.006	0.005	0.007	0.005	0.008	0.011

	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	TOTAL
1	0.011	0.031	0.021	0.010	0.020	0.007	0.017	0.010	0.013	0.010	0.010	0.014	0.008	0.016	0.008	0.016	1.000000
2	0.000	0.000	0.000	0.000	0.000	0.002	0.002	0.000	0.000	0.002	0.000	0.005	0.005	0.019	0.036	0.026	1.000000
3	0.000	0.044	0.022	0.011	0.011	0.011	0.033	0.011	0.000	0.011	0.022	0.011	0.022	0.000	0.011	0.000	1.000000
4	0.017	0.006	0.002	0.010	0.010	0.011	0.017	0.015	0.011	0.011	0.017	0.019	0.019	0.015	0.013	0.019	1.000000
5	0.000	0.000	0.000	0.000	0.000	0.005	0.009	0.000	0.016	0.027	0.021	0.023	0.028	0.018	0.028	0.030	1.000000
6	0.000	0.000	0.000	0.000	0.000	0.001	0.001	0.003	0.000	0.008	0.001	0.001	0.009	0.020	0.015	0.020	1.000000
7	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.010	0.005	0.014	0.010	1.000000
8	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.031	0.000	0.000	1.000000
9	0.031	0.027	0.021	0.019	0.014	0.016	0.031	0.023	0.010	0.016	0.041	0.023	0.027	0.027	0.031	0.010	1.000000
10	0.005	0.017	0.011	0.011	0.010	0.012	0.016	0.021	0.020	0.020	0.029	0.025	0.031	0.028	0.031	0.033	1.000000
11	0.000	0.003	0.000	0.000	0.003	0.003	0.001	0.001	0.010	0.018	0.019	0.019	0.019	0.025	0.025	0.016	1.000000
12	0.000	0.000	0.000	0.000	0.000	0.003	0.003	0.000	0.006	0.003	0.003	0.008	0.014	0.014	0.034	0.065	1.000000
13	0.045	0.035	0.043	0.006	0.061	0.012	0.025	0.004	0.000	0.008	0.012	0.024	0.006	0.006	0.012	0.000	1.000001
14	0.004	0.013	0.009	0.007	0.011	0.005	0.009	0.004	0.001	0.009	0.000	0.005	0.007	0.005	0.007	0.003	1.000000
15	0.000	0.000	0.000	0.000	0.007	0.000	0.000	0.000	0.000	0.015	0.000	0.000	0.015	0.007	0.007	0.007	1.000000
16	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000	1.000000
17	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
18	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
19	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
20	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
21	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
22	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
23	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
24	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
25	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
26	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
27	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
28	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
29	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
30	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
31	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
32	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
33	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
34	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
35	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
36	0.020	0.008	0.011	0.016	0.022	0.011	0.015	0.027	0.026	0.030	0.027	0.029	0.028	0.026	0.035	0.031	1.000000
37	0.010	0.012	0.009	0.007	0.013	0.008	0.012	0.011	0.011	0.016	0.016	0.017	0.018	0.019	0.023	0.021	1.000000

USER INPUT IS READ FROM UNIT 25
RECORD IDENTIFIER FIELDS 11 CHARACTERS LONG ARE EXPECTED.
THE FIRST 499 COLUMNS OF EACH INPUT RECORD ARE PROCESSED.

```
RECORD
NUMBER                RECORD

    * File created using WinMACCS version 3.7.0 11/13/2012 10:58:50 AM
    *
    * DCF_FILE_TH - Identifies the DCF file to be used for the MACCS calculation
  1 DCF_FILE001     'C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Data\FGR13GyEquivDCF.INP'
    *
    * EANAM1 - Identifies the EARLY calculation
  2 MIEANAM1001     'OCP3 high density no spray, EARLY input'
    *
    * ENDAT2 - control flag allowing execution of ATMOS and EARLY without CHRONC
  3 MIENDAT2001     .FALSE.
    *
    * IPLUME - dispersion code option
  4 MIIPLUME001     3
    *
    * Form 'Grid Subdivisions' Comment:
    * Value used in NUREG-1150.
    *
    * NUMFIN - number of fine-grid subdivisions used by model
  5 MINUMFIN001     7
    *
    * IPRINT - amount of output desired
  6 MIIPRINT001     0
    *
    * POPFLG - is population uniform or defined by Site Data File.
  7 PDPOPFLG001     FILE
```

```
  *
  * ORGNAM_FGR13, ORGFLG_FGR13 - list of organs to be included in the calculations using FGR13 DCF file
 8  MIORGDEF001      A-SKIN  .TRUE.
 9  MIORGDEF002      'A-RED MARR'    .TRUE.
10  MIORGDEF003      A-LUNGS .TRUE.
11  MIORGDEF004      A-THYROID       .TRUE.
12  MIORGDEF005      A-STOMACH       .TRUE.
13  MIORGDEF006      'A-LOWER LI'    .TRUE.
14  MIORGDEF007      L-ICRP60ED      .TRUE.
15  MIORGDEF008      'L-RED MARR'    .TRUE.
16  MIORGDEF009      'L-BONE SUR'    .TRUE.
17  MIORGDEF010      L-BREAST        .TRUE.
18  MIORGDEF011      L-LUNGS .TRUE.
19  MIORGDEF012      L-THYROID       .TRUE.
20  MIORGDEF013      'L-LOWER LI'    .TRUE.
21  MIORGDEF014      'L-BLAD WAL'    .TRUE.
22  MIORGDEF015      L-LIVER .TRUE.
  *
  * RISCAT - Output relative contribution of each weather category bins
23  MIRISCAT001      .FALSE.
  *
  * OVRRID - Flag indicating if Wind Rose defaults from ATMOS are to be overridden
24  MIOVRRID001      .FALSE.
  *
  * CSFACT - Cloudshine shielding factor
25  SECSFACT001      1.
26  SECSFACT002      0.6
27  SECSFACT003      0.5
  *
  * PROTIN - Inhalation protection factor
28  SEPROTIN001      0.98
29  SEPROTIN002      0.46
30  SEPROTIN003      0.33
  *
  * BRRATE - Breathing rates
31  SEBRRATE001    2.66E-04
32  SEBRRATE002      2.66E-04
33  SEBRRATE003      2.66E-04
  *
  * SKPFAC - skin protection factors
34  SESKPFAC001      0.98
35  SESKPFAC002      0.46
36  SESKPFAC003      0.33
  *
  * GSHFAC - groundshine shielding factors
37  SEGSHFAC001      0.5
38  SEGSHFAC002      0.18
39  SEGSHFAC003      0.1
  *
  * Form 'Emergency Phase Resuspension' Comment:
  * Values from NUREG-1150.
  *
  * RESCON - Initial value for emergency-phase resuspension concentration factor.
40  SERESCON001      1.E-04
  *
  * RESHAF - Emergency-phase resuspension concentration coefficient weathering half-life.
41  SERESHAF001      1.82000E+05
  *
  * EANAM2 - Name of emergency response cohort
42  EZEANAM2001      '0-10 Schools'
  *
  * WTNAME - type of weighting factor to be used in generating weighted sum of results
43  EZWTNAME001      SUMPOP
  *
  * WTFRAC - weighting fraction applied to results of emergency response cohort
44  EZWTFRAC001      0.172
  *
  * EVATYP - decides on radial or network evacuation option.
45  EZEVATYP001      NETWORK
  *
  * TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
46  TRAVELPOINT      CENTERPOINT
  *
  * ESPEED - evacuee travel speed during the three phases of evacuation
47  EZESPEED001      8.941
48  EZESPEED002      6.706
49  EZESPEED003      8.941
  *
  * ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
50  EZESPMUL001      0.7
51  EZESPMUL002      0.7
52  EZESPMUL003      0.7
  *
  * Form 'Phase Durations' Comment:
  * 0-10 Schools
  *
  * REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
53  EZREFPNT001      ALARM
  *
  * DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
54  EZDURBEG001      900.
  *
  * DURMID - duration of middle phase of evacuation, in seconds.
55  EZDURMID001      7200.
  *
  * NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
56  EZNUMEVA001      18
  *
  * DLTSHL - delay from reference time point to when individual takes shelter.  DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
57  EZDLTSHL001      900.
58  EZDLTSHL002      900.
59  EZDLTSHL003      900.
60  EZDLTSHL004      900.
61  EZDLTSHL005      900.
62  EZDLTSHL006      900.
63  EZDLTSHL007      900.
64  EZDLTSHL008      900.
65  EZDLTSHL009      900.
66  EZDLTSHL010      900.
67  EZDLTSHL011      900.
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 68 EZDLTSHL012     900.
 69 EZDLTSHL013     900.
 70 EZDLTSHL014     900.
 71 EZDLTSHL015     900.
 72 EZDLTSHL016     900.
 73 EZDLTSHL017     900.
 74 EZDLTSHL018     900.
    *
    * DLTEVA -Delay time to begin evacuation
 75 EZDLTEVA001     4500.
 76 EZDLTEVA002     4500.
 77 EZDLTEVA003     4500.
 78 EZDLTEVA004     4500.
 79 EZDLTEVA005     4500.
 80 EZDLTEVA006     4500.
 81 EZDLTEVA007     4500.
 82 EZDLTEVA008     4500.
 83 EZDLTEVA009     4500.
 84 EZDLTEVA010     4500.
 85 EZDLTEVA011     4500.
 86 EZDLTEVA012     4500.
 87 EZDLTEVA013     4500.
 88 EZDLTEVA014     4500.
 89 EZDLTEVA015     4500.
 90 EZDLTEVA016     4500.
 91 EZDLTEVA017     4500.
 92 EZDLTEVA018     4500.
    *
    * ESPGRD_NET - Evacuation speed multiplier for network evacuation
 93 EZESPGRD001   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 94 EZESPGRD002   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 95 EZESPGRD003   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 96 EZESPGRD004   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 97 EZESPGRD005   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 98 EZESPGRD006   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 99 EZESPGRD007   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 100 EZESPGRD008   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 101 EZESPGRD009   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 102 EZESPGRD010   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 103 EZESPGRD011   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 104 EZESPGRD012   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 105 EZESPGRD013   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  0.75  1.
1.  1.  1.  1.  1.  0.75  0.75  0.75
 106 EZESPGRD014   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.5  1.5  1.5  1.5  1.5  1.5  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  0.75  1.
1.  1.  1.  1.  1.  0.75  0.75  0.75
 107 EZESPGRD015   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.5  1.5  1.5  1.  1.  1.  1.  1.  1.5  1.5  1.5  1.5  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  0.75  1.
1.  1.  1.  1.  1.  0.75  0.75  0.75
 108 EZESPGRD016   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 109 EZESPGRD017   0.75  0.75  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  0.75  0.75  0.75
 110 EZESPGRD018   1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  1.  1.  1.
 111 EZESPGRD019   1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  1.  1.  1.
    *
    * IDIREC - destination direction of every spatial element in the evacuation and sheltering region
 112 EZIDIREC001   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1
 113 EZIDIREC002   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1
 114 EZIDIREC003   1  1  1  1  1  1  1  1  1  1  1  1  1  2  1  1  1  1  1  1  1  1  1  1  2  1  4  4  4  4  2  2  2  2  2  1  1  1  4  4  4  4  2  2  2  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  4  1  1
 115 EZIDIREC004   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  2  2  1  1  1  1  4  4  4  2  2  1  4  4  2  1  4  4  2  1  1  4  2  1
4  4  4  4  4  1  1
 116 EZIDIREC005   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  1  1  1  1  1  1  1  1  2  2  2  2  2  2  1  1  1  1  1  2  1  4  4  2  2  1  4  4  2  2  1  4  4  2  2  1  4  4  2  1
1  1  1  1  1  1  1
 117 EZIDIREC006   1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  1  1  1  1  1  1  1  1  1  1  2  2  2  2  2  2  2  2  1  1  1  1  2  1  1  4  2  2  1  2  1  4  4  2  2  1  4  4  4  4  4
4  4  1  1  1  1  1
 118 EZIDIREC007   2  2  1  2  2  1  2  2  1  4  2  1  4  2  2  1  1  1  1  1  1  1  1  1  1  1  2  2  1  4  4  2  1  1  1  1  4  4  2  2  1  1  4  4  1  1  4  1  4  4  4  1  4  4  4  4  4
4  1  1  1  1  1  1
 119 EZIDIREC008   1  4  1  1  4  2  1  4  2  1  1  4  2  2  1  1  2  1  1  1  4  4  4  1  1  1  1  1  1  4  1  4  2  1  4  4  2  1  2  1  2  1  4  2  1  4  2  1  1  4  4  1  4  4  4  2  2
1  1  1  1  1  2  2
 120 EZIDIREC009   1  1  4  2  1  1  2  1  1  4  1  4  1  4  1  1  1  2  1  1  1  4  4  4  1  1  4  1  2  2  2  1  1  2  1  4  2  1  2  1  2  1  1  4  1  4  1  4  4  4  4  1  4  2  1  4  1
4  1  1  1  2  2  2
 121 EZIDIREC010   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  4  4  2  2  2  2  1  2  1  4  4  1  1  1  4  2  1  4  4  2  2  1  1  1  1  1  4
4  4  2  2  1  1  1
 122 EZIDIREC011   1  1  4  2  1  4  2  1  4  4  4  4  2  2  1  4  4  2  2  2  2  2  2  1  4  4  1  1  4  2  2  1  1  1  4  2  1  4  1  4  2  1  4  1  1  2  1  1  4  4  4  2  2  2  1  4  4
4  2  2  1  4  2  2
 123 EZIDIREC012   2  1  1  4  1  1  4  1  4  4  2  1  4  2  1  1  4  4  2  2  2  1  2  1  2  1  2  2  1  4  4  4  4  2  2  2  1  2  1  1  1  1  4  1  1  1  1  4  4  4  2  2  2  2  1  4  2
1  1  2  2  1  4  2
 124 EZIDIREC013   1  1  4  1  4  2  1  1  2  1  4  4  2  2  1  1  1  1  1  1  1  2  1  4  4  4  1  2  1  4  4  4  4  2  2  1  4  4  1  1  4  2  1  1  2  1  2  1  4  4  2  2  2  1  1  4  2
1  4  2  1  4  1  1
 125 EZIDIREC014   1  1  4  1  1  1  2  1  2  1  2  1  1  2  1  1  1  1  1  1  1  4  1  1  1  1  2  1  4  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  2  1  1  4  4  2  2  1  1  4  1
4  1  1  1  1  1  1
 126 EZIDIREC015   1  1  4  2  2  2  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  4  4  2  2  2  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  2  1  1  1  4  2  1  2  1  1  2  1
1  2  2  2  1  2  1
 127 EZIDIREC016   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1
 128 EZIDIREC017   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1
 129 EZIDIREC018   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1
 130 EZIDIREC019   1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
1  1  1  1  1  1  1
    *
    * LASMOV - The outermost spatial interval of the evacuation movement zone.
 131 EZLASMOV001     19
```

```
     *
     * Form 'Duration of Early Phase' Comment:
     * 1 week.
     *
     * ENDEMP - duration of the emergency-phase period, seconds
 132  SRENDEMP001      6.04800E+05
     *
     * CRIORG - critical organ for relocation decisions during emergency-phase period
 133  SRCRIORG001      L-ICRP60ED
     *
     * Form 'Hot Spot Relocation' Comment:
     * Randy Sullivan recommended these values.  (4 hours for 10 mile evac)
     *
     * TIMHOT - hot-spot relocation action time, seconds after plume arrival
 134  SRTIMHOT001      14400.
     *
     * Form 'Normal Relocation' Comment:
     * Randy Sullivan recommended these values. (8 hours for 10 mile evac)
     *
     *
     * TIMNRM - Normal Relocation Time (Seconds from Plume Arrival)
 135  SRTIMNRM001      57600.
     *
     * DOSHOT - Hot-Spot Relocation Dose Threshold (Sieverts)
 136  SRDOSHOT001      0.05
     *
     * DOSNRM - Normal Relocation Dose Threshold (Sieverts)
 137  SRDOSNRM001      0.01
     *
     * NUMEFA - Number of Early Fatality Effects
 138  EFNUMEFA001      3
     *
     * ORGNAM2, EFFACA, EFFACB, EFFTHR Early Fatality Effects - target organ, alpha factor and beta factor for hazard function, and threshold dose (Sieverts)
 139  EFATAGRP001      'A-RED MARR'    5.6     6.1     2.32
 140  EFATAGRP002      A-LUNGS 23.5    9.6     13.6
 141  EFATAGRP003      A-STOMACH       12.1    9.3     6.5
     *
     * NUMEIN - Number of Early Injury Effects
 142  EINUMEIN001      7
     *
     * ORGNAM3, EINAME, EISUSC, EITHRE, EIFACA, EIFACB Early Injury Effects - name, target organ, affected population fract, threshold dose, alpha factor, beta factor.
 143  EINJUGRP001      'PRODROMAL VOMIT'       A-STOMACH       1.      0.5     2.      3.
 144  EINJUGRP002      DIARRHEA        A-STOMACH       1.      1.      3.      2.5
 145  EINJUGRP003      PNEUMONITIS     A-LUNGS 1.      9.2     16.6    7.3
 146  EINJUGRP004      'SKIN ERYTHEMA' A-SKIN  1.      3.      6.      5.
 147  EINJUGRP005      TRANSEPIDERMAL  A-SKIN  1.      10.     20.     5.
 148  EINJUGRP006      THYROIDITIS     A-THYROID       1.      40.     240.    2.
 149  EINJUGRP007      HYPOTHYROIDISM  A-THYROID       1.      2.      60.     1.3
     *
     * Form 'Latent Cancer Parameters' Comment:
     * Risk factors are those recommended by Keith Eckerman to use with a FGR-13 DCF file set modified as follows:
     * Red marrow DCFs have been modified to use a RBE of 1 for alpha radiation; breast DCFs have been modified to use an RBE of 10 for alpha radiation.
     * As a kluge, the organ named bladder wall contains data for the pancreas, which is used as a surrogate for the soft tissue for the purpose of evaluating residual cancers.
     *
     * NUMACA - number of latent cancer effects
 150  LCNUMACA001      8
     *
     * ACTHRE - dose threshold for linear dose response, Sieverts
 151  LCACTHRE001      0.E+00
     *
     * DDTHRE - dose threshold for applying dose-dependent reduction factor, DDREFA
 152  LCDDTHRE001      .2
     *
     * ACNAME, ORGNAM4, ACSUSC, DOSEFA, DOSEFB, CFRISK, CIRISK, DDREFA - Latent Cancer Effects Parameters
 153  LCANCERS001      LEUKEMIA        'L-RED MARR'    1.      1.      0.      0.0111  0.0113  2.
 154  LCANCERS002      BONE    'L-BONE SUR'    1.      1.      0.      1.9E-04 2.71E-04        2.
 155  LCANCERS003      BREAST  L-BREAST        1.      1.      0.      0.00506 0.0101  1.
 156  LCANCERS004      LUNG    L-LUNGS 1.      1.      0.      0.0198  0.0208  2.
 157  LCANCERS005      THYROID L-THYROID       1.      1.      0.      6.48E-04        0.00648 2.
 158  LCANCERS006      LIVER   L-LIVER 1.      1.      0.      0.003   0.00316 2.
 159  LCANCERS007      COLON   'L-LOWER LI'    1.      1.      0.      0.0208  0.0378  2.
 160  LCANCERS008      RESIDUAL        'L-BLAD WAL'    1.      1.      0.      0.0493  0.169   2.
     *
     * NUM1=0
 161  TYPE1NUMBER      0
     *
     * NUM1 - Number of results of type 1
 162  TYPE1NUMBER      38
 ********   RECORD NUMBER   162 REPLACES RECORD NUMBER   161   ********
     *
     * NAME1, I1DIS1, I2DIS1, CCDF1 - Health-Effect Cases
 163  TYPE1OUT001      'ERL FAT/TOTAL' 1       12      REPORT
 164  TYPE1OUT002      'ERL FAT/TOTAL' 1       19      REPORT
 165  TYPE1OUT003      'ERL FAT/TOTAL' 1       26      REPORT
 166  TYPE1OUT004      'CAN INJ/TOTAL' 1       12      REPORT
 167  TYPE1OUT005      'CAN INJ/TOTAL' 1       15      REPORT
 168  TYPE1OUT006      'CAN INJ/TOTAL' 1       17      REPORT
 169  TYPE1OUT007      'CAN INJ/TOTAL' 1       18      REPORT
 170  TYPE1OUT008      'CAN INJ/TOTAL' 1       19      REPORT
 171  TYPE1OUT009      'CAN INJ/TOTAL' 1       21      REPORT
 172  TYPE1OUT010      'CAN INJ/TOTAL' 1       23      REPORT
 173  TYPE1OUT011      'CAN INJ/TOTAL' 1       25      REPORT
 174  TYPE1OUT012      'CAN INJ/TOTAL' 1       26      REPORT
 175  TYPE1OUT013      'CAN FAT/TOTAL' 1       12      REPORT
 176  TYPE1OUT014      'CAN FAT/TOTAL' 1       15      REPORT
 177  TYPE1OUT015      'CAN FAT/TOTAL' 1       17      REPORT
 178  TYPE1OUT016      'CAN FAT/TOTAL' 1       18      REPORT
 179  TYPE1OUT017      'CAN FAT/TOTAL' 1       19      REPORT
 180  TYPE1OUT018      'CAN FAT/TOTAL' 1       21      REPORT
 181  TYPE1OUT019      'CAN FAT/TOTAL' 1       23      REPORT
 182  TYPE1OUT020      'CAN FAT/TOTAL' 1       25      REPORT
 183  TYPE1OUT021      'CAN FAT/TOTAL' 1       26      REPORT
 184  TYPE1OUT022      'CAN FAT/THYROID'       1       12      REPORT
 185  TYPE1OUT023      'CAN FAT/THYROID'       1       19      REPORT
 186  TYPE1OUT024      'CAN FAT/THYROID'       1       21      REPORT
 187  TYPE1OUT025      'CAN FAT/THYROID'       1       26      REPORT
 188  TYPE1OUT026      'CAN FAT/BREAST'        1       12      REPORT
 189  TYPE1OUT027      'CAN FAT/BREAST'        1       19      REPORT
 190  TYPE1OUT028      'CAN FAT/BREAST'        1       21      REPORT
 191  TYPE1OUT029      'CAN FAT/BREAST'        1       26      REPORT
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  192  TYPE1OUT030     'CAN FAT/LUNG'  1      12     REPORT
  193  TYPE1OUT031     'CAN FAT/LUNG'  1      19     REPORT
  194  TYPE1OUT032     'CAN FAT/LUNG'  1      21     REPORT
  195  TYPE1OUT033     'CAN FAT/LUNG'  1      26     REPORT
  196  TYPE1OUT034     'CAN FAT/LEUKEMIA'     1      26     REPORT
  197  TYPE1OUT035     'CAN FAT/BONE'  1      26     REPORT
  198  TYPE1OUT036     'CAN FAT/LIVER' 1      26     REPORT
  199  TYPE1OUT037     'CAN FAT/COLON' 1      26     REPORT
  200  TYPE1OUT038     'CAN FAT/RESIDUAL'     1      26     REPORT
     *
     * NUM2=0
  201  TYPE2NUMBER     0
     *
     * NUM2 - Number of results of type 2
  202  TYPE2NUMBER     1
********   RECORD NUMBER   202 REPLACES RECORD NUMBER   201   ********
     *
     * RISTHR, CCDF2  - Early-Fatality Radius
  203  TYPE2OUT001     0.      NONE
     *
     * NUM3=0
  204  TYPE3NUMBER     0
     *
     * NUM3 - Number of results of type 3
  205  TYPE3NUMBER     3
********   RECORD NUMBER   205 REPLACES RECORD NUMBER   204   ********
     *
     * NAME3, DOSTH3, CCDF3 - Population Exceeding a Dose Threshold
  206  TYPE3OUT001     'A-RED MARR'    2.32    NONE
  207  TYPE3OUT002     A-LUNGS 13.6    NONE
  208  TYPE3OUT003     A-STOMACH       6.5     NONE
     *
     * NUM4=0
  209  TYPE4NUMBER     0
     *
     * NUM5 =0
  210  TYPE5NUMBER     0
     *
     * NUM5 - Number of results of type 5
  211  TYPE5NUMBER     4
********   RECORD NUMBER   211 REPLACES RECORD NUMBER   210   ********
     *
     * NAME5, I1DIS5, CCDF5 - Population Dose
  212  TYPE5OUT001     L-ICRP60ED      1      12     REPORT
  213  TYPE5OUT002     L-ICRP60ED      1      19     REPORT
  214  TYPE5OUT003     L-ICRP60ED      1      21     REPORT
  215  TYPE5OUT004     L-ICRP60ED      1      26     REPORT
     *
     * NUM6 =0
  216  TYPE6NUMBER     0
     *
     * NUM7=0
  217  TYPE7NUMBER     0
     *
     * NUM8=0
  218  TYPE8NUMBER     0
     *
     * NUM8 - Number of results of type 8
  219  TYPE8NUMBER     17
********   RECORD NUMBER   219 REPLACES RECORD NUMBER   218   ********
     *
     * NAME8, I1DIS8, I2DIS8, CCDF8 - Population-Weighted Risk
  220  TYPE8OUT001     'CAN FAT/TOTAL' 1      12     REPORT
  221  TYPE8OUT002     'CAN FAT/TOTAL' 1      15     REPORT
  222  TYPE8OUT003     'CAN FAT/TOTAL' 1      17     REPORT
  223  TYPE8OUT004     'CAN FAT/TOTAL' 1      18     REPORT
  224  TYPE8OUT005     'CAN FAT/TOTAL' 1      19     REPORT
  225  TYPE8OUT006     'CAN FAT/TOTAL' 1      21     REPORT
  226  TYPE8OUT007     'CAN FAT/TOTAL' 1      23     REPORT
  227  TYPE8OUT008     'CAN FAT/TOTAL' 1      25     REPORT
  228  TYPE8OUT009     'CAN FAT/TOTAL' 1      26     REPORT
  229  TYPE8OUT010     'CAN FAT/TOTAL' 13     15     REPORT
  230  TYPE8OUT011     'CAN FAT/TOTAL' 16     17     REPORT
  231  TYPE8OUT012     'CAN FAT/TOTAL' 18     18     REPORT
  232  TYPE8OUT013     'CAN FAT/TOTAL' 19     19     REPORT
  233  TYPE8OUT014     'CAN FAT/TOTAL' 20     21     REPORT
  234  TYPE8OUT015     'CAN FAT/TOTAL' 22     23     REPORT
  235  TYPE8OUT016     'CAN FAT/TOTAL' 24     25     REPORT
  236  TYPE8OUT017     'CAN FAT/TOTAL' 26     26     REPORT
     *
     * NUMA=0
  237  TYPEANUMBER     0
     *
     * NUMA - Number of results of type A
  238  TYPEANUMBER     1
********   RECORD NUMBER   238 REPLACES RECORD NUMBER   237   ********
     *
     * NAMEA, I1DISA, I2DISA, CCDFA - Peak Dose vs Distance
  239  TYPEAOUT001     L-ICRP60ED      1      26     REPORT
     *
     * NUMB =0
  240  TYPEBNUMBER     0
     *
     * NUMC=0
  241  TYPECNUMBER     0
     *
     * Form 'Land Area Exceeding Dose' Comment:
     * Emergency Phase PAGs
     *
     * NUMC number of typeC output
  242  TYPECNUMBER     3
********   RECORD NUMBER   242 REPLACES RECORD NUMBER   241   ********
     *
     * ORGNAM8, ELEVDOSE, PRINT_FLAG_C - organs for typeC output
  243  TYPECOUT001     L-ICRP60ED      0.01    .FALSE.
  244  TYPECOUT002     L-ICRP60ED      0.05    .FALSE.
  245  TYPECOUT003     A-THYROID       0.05    .FALSE.
     *
     * NUMD = 0
  246  TYPEDNUMBER     0
```

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      *
      * NUMD number of typeD output
  247  TYPEDNUMBER      16
********   RECORD NUMBER   247 REPLACES RECORD NUMBER   246   ********
      *
      * I1DISD, NUCLIDED, ELEVCONC, PRINT_FLAG_D
  248  TYPEDOUT001      12        Cs-137  37000.  .FALSE.
  249  TYPEDOUT002      12      Cs-137  1.85000E+05     .FALSE.
  250  TYPEDOUT003      12      Cs-137  5.55000E+05     .FALSE.
  251  TYPEDOUT004      12      Cs-137  1.480000E+06    .FALSE.
  252  TYPEDOUT005      19      Cs-137  37000.      .FALSE.
  253  TYPEDOUT006      19      Cs-137  1.85000E+05     .FALSE.
  254  TYPEDOUT007      19      Cs-137  5.55000E+05     .FALSE.
  255  TYPEDOUT008      19      Cs-137  1.480000E+06    .FALSE.
  256  TYPEDOUT009      21      Cs-137  37000.  .FALSE.
  257  TYPEDOUT010      21      Cs-137  1.85000E+05     .FALSE.
  258  TYPEDOUT011      21      Cs-137  5.55000E+05     .FALSE.
  259  TYPEDOUT012      21      Cs-137  1.480000E+06    .FALSE.
  260  TYPEDOUT013      25      Cs-137  37000.  .FALSE.
  261  TYPEDOUT014      25      Cs-137  1.85000E+05     .FALSE.
  262  TYPEDOUT015      25      Cs-137  5.55000E+05     .FALSE.
  263  TYPEDOUT016      25      Cs-137  1.480000E+06    .FALSE.
      *
      * DOSMOD, dose model, LNT, AT or PL
  264  LCDOSMOD001      LNT
      *
      * Form 'Annual Threshold' Comment:
      * Threshold values are from Health Physics Society position statement PS010-1 (August 2004).
      *
      * DTHNUM, Number of annual dose threshold values
  265  LCDTHNUM001      1
      *
      * DTHANN, Annual threshold values
  266  LCDTHANN001      1.E-04
      *
      * DTHLIF, Lifetime dose restriction
  267  LCDTHLIF001      10000.
      *
      * KIMODL, KI model
  268  EZKIMODL001      KI
      *
      * EFFACY_TH, KI Ingestion
  269  EZEFFACY001      0.7
      *
      * POPFRAC_TH, KI Ingestion, SLT
  270  EZPOPFRC001      1.
      *
      * FRACLD_FILE - popflg=FILE, dummy variable
  271  STFRACLD001      1.0
      *
      * NUME=0
  272  TYPEENUMBER      0
      .
******** TERMINATOR RECORD ENCOUNTERED -- END OF BASE CASE USER INPUT ********


USER INPUT PROCESSING SUMMARY - BASE CASE

NUMBER OF RECORDS READ                         =  472
NUMBER OF BLANK OR COMMENT RECORDS READ        =  199
NUMBER OF TERMINATOR RECORDS                   =    1
NUMBER OF RECORDS PROCESSED                    =  272
    NUMBER OF PROCESSED RECORDS DUPLICATED     =    8
    NUMBER OF PROCESSED RECORDS SORTED         =  264
****************************************************************************************************


THE KI MODEL IS IN EFFECT
READING DCF FILE:C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Data\FGR13GyEquivDCF.INP
DCF FILE is of type :FGR13DF
Am using a FGR13DCF dose factor file

The list of defined organs is as follows (A- is ACUTE and L- is LIFETIME):
   A-SKIN
   A-RED MARR
   A-LUNGS
   A-THYROID
   A-STOMACH
   A-LOWER LI
   L-ICRP60ED
   L-RED MARR
   L-BONE SUR
   L-BREAST
   L-LUNGS
   L-THYROID
   L-LOWER LI
   L-BLAD WAL
   L-LIVER

READING FROM A DOSE CONVERSION FILE WITH THE FOLLOWING HEADER:
FGR13DF 5/13/2008 12:23:56   Version 1.03, Gy-Equivalent DCFs
Internal Dose Coefficients derived from FGR 13, EPA 402-R-99-001



With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:


IRAD     1  2  3  4  5  6  7  8  9 10 11 12 13 14 15 16
   1     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   2     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   3     1  1  1  1  1  1  1  1  1  1  1  1  1  2  1  1
   4     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   5     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4
   6     1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  1
   7     2  2  1  2  2  1  2  2  1  4  2  1  4  2  2  1
   8     1  4  1  1  4  2  1  4  2  1  1  4  2  2  1  1
   9     1  1  4  2  1  1  2  1  1  4  1  4  1  4  1  1
  10     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
```

```
  11     1  1  4  2  1  4  2  1  4  4  4  4  2  2  1  4
  12     2  1  1  4  1  1  4  1  4  4  2  1  4  2  1  1
  13     1  1  4  1  4  2  1  1  2  1  4  4  2  2  1  1
  14     1  1  4  1  1  1  2  1  2  1  2  1  1  2  1  1
  15     1  1  4  2  2  2  1  1  1  1  1  1  1  1  1  1
  16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4
  17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  18     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  19     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD     17  18  19  20  21  22  23  24  25  26  27  28  29  30  31  32
   1     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   2     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   3     1  1  1  1  1  1  1  1  2  1  4  4  4  4  2  2
   4     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   5     1  1  1  1  1  1  1  1  1  2  2  2  2  2  2  1
   6     1  1  1  1  1  1  1  1  1  1  2  2  2  2  2  2
   7     1  1  1  1  1  1  1  1  1  1  2  2  1  4  4  2
   8     2  1  1  1  4  4  4  1  1  1  1  1  1  4  1  4
   9     1  2  1  1  1  4  4  4  1  1  4  1  2  2  2  1
  10     1  1  1  1  1  1  1  1  1  1  1  1  1  4  4  4
  11     4  2  2  2  2  2  2  1  4  4  1  1  4  2  2  1
  12     4  4  2  2  2  1  2  1  2  1  2  2  1  4  4  4
  13     1  1  1  1  1  2  1  4  4  4  1  2  1  4  4  4
  14     1  1  1  1  1  4  1  1  1  1  2  1  4  1  1  1
  15     1  1  1  1  1  1  1  4  4  4  2  2  2  1  1  1
  16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  18     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  19     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD     33  34  35  36  37  38  39  40  41  42  43  44  45  46  47  48
   1     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   2     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   3     2  2  2  1  1  1  4  4  4  4  2  2  2  1  1  1
   4     4  2  2  1  1  1  1  4  4  4  2  2  1  4  4  2
   5     1  1  1  1  2  1  4  4  2  2  1  4  4  2  2  1
   6     2  2  1  1  1  1  2  1  1  4  2  2  1  2  1  4
   7     1  1  1  1  4  4  2  2  1  1  4  4  1  1  4  1
   8     2  1  4  4  2  1  2  1  2  1  4  2  1  4  2  1
   9     1  2  1  4  2  1  2  1  2  1  1  4  1  4  1  4
  10     2  2  2  2  1  2  1  4  4  1  1  1  4  2  1  4
  11     1  1  4  2  1  4  1  4  2  1  4  1  1  2  1  1
  12     4  2  2  2  1  2  1  1  1  1  4  1  1  1  1  4
  13     4  2  2  1  4  4  1  1  4  2  1  1  2  1  2  1
  14     1  1  1  1  1  1  1  1  1  1  1  1  1  1  2  1
  15     1  1  1  1  1  1  1  1  1  1  1  4  1  2  1  1
  16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  18     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  19     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD     49  50  51  52  53  54  55  56  57  58  59  60  61  62  63  64
   1     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   2     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
   3     1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  1
   4     1  4  4  2  1  1  4  2  1  4  4  4  4  4  1  1
   5     4  4  2  2  1  4  4  2  1  1  1  1  1  1  1  1
   6     4  2  2  1  4  4  4  4  4  4  4  1  1  1  1  1
   7     4  4  4  1  4  4  4  4  4  4  1  1  1  1  1  1
   8     1  4  4  1  4  4  4  2  2  1  1  1  1  1  2  2
   9     4  4  4  1  4  2  1  4  1  4  1  1  1  2  2  2
  10     4  2  2  1  1  1  1  1  4  4  4  2  2  1  1  1
  11     4  4  4  2  2  2  1  4  4  4  2  2  1  4  2  2
  12     4  4  2  2  2  2  1  4  2  1  1  2  2  1  4  2
  13     4  4  2  2  2  1  1  4  2  1  4  2  1  4  1  1
  14     1  4  4  2  2  1  1  4  1  4  1  1  1  1  1  1
  15     1  4  2  1  2  1  1  2  1  1  2  2  2  1  2  1
  16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  18     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  19     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1


USING THE FOLLOWING SITE DATA FILE:

 SECPOP2000 Version: 3.13.1 MACCS2 Formatted Site:  File for Peach Bottom Census: C:\Program Files\SecPOP_2000\Census\CENSUS00.DAT County: C:\Program Files\SecPOP_2000\Census\COUNTY2002RA.DAT* Created from C:\NBixler\WinMACCS Projects\SOARCA\PeachBottomLTSBO-SNL-
Jan2008\Data\PBSite2005_16.inp using PopMod 1.0.4 1/30/2008 11:29:59 AM
 Lat: 39d45'32" Long:  76d16' 9" Population multiplier: 1.0533 Economic multiplier: 1.0900 Run Time: 1/30/2008 11:19:40 AM from C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Data\PBSite2005_64.inp using WinMACCS 3.7.0 11/13/2012 11:02:12 AM
  26 SPATIAL INTERVALS
  64 WIND DIRECTIONS
   7 CROP CATEGORIES
   4 WATER PATHWAY ISOTOPES
   1 WATERSHEDS
  97 ECONOMIC REGIONS
 SPATIAL DISTANCES     KILOMETERS
    0.1600    0.5200    1.2100    1.6100    2.1300    3.2200    4.0200    4.8300
    5.6300    8.0500   11.2700   16.0900   20.9200   25.7500   32.1900   40.2300
   48.2800   64.3700   80.4700  112.6500  160.9300  241.1400  321.8700  563.2700
  804.6700 1609.3400
 POPULATION1
        0        0        0        0        0        0        0     1.72
        0     12.9   15.652  75.85201        0        0        0        0
        0        0        0        0        0        0        0        0
        0        0
        0        0        0        0        0        0        0     1.72
        0     12.9   15.652  75.85201        0        0        0        0
        0        0        0        0        0        0        0        0
        0        0
        0        0        0        0        0        0    0.516     0.86
    1.376    11.18   20.296   93.912        0        0        0        0
        0        0        0        0        0        0        0        0
        0        0
        0        0        0        0        0        0    1.032        0
     2.58    9.632    24.94  111.972        0        0        0        0
        0        0        0        0        0        0        0        0
        0        0
        0        0        0        0        0        0    1.032        0
     2.58    9.632    24.94  111.972        0        0        0        0
        0        0        0        0        0        0        0        0
```

0	0						
0	0	0	0	0	0	1.032	0
2.58	9.632	24.94	111.972	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.344	1.892	2.924
1.376	13.932	22.704	77.57201	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.86	2.58	5.676
0	18.232	20.64	43	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.86	2.58	5.676
0	18.232	20.64	43	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.86	2.58	5.676
0	18.232	20.64	43	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.344	2.236	2.924
1.204	18.748	21.156	40.936	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	1.892	0.172
2.408	19.264	21.844	38.7	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	1.892	0.172
2.408	19.264	21.844	38.7	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	1.892	0.172
2.408	19.264	21.844	38.7	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	1.032	1.032
2.924	16.34	25.456	61.92	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.172	0.172	2.064
3.44	13.416	29.24	84.968	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.172	0.172	2.064
3.44	13.416	29.24	84.968	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.172	0.172	2.064
3.44	13.416	29.24	84.968	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	1.032
3.784	14.104	26.316	87.03201	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
3.956	14.792	23.564	88.924	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
3.956	14.792	23.564	88.924	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
3.956	14.792	23.564	88.924	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
1.892	9.632	26.66	82.56001	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
0	4.472	29.928	76.368	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
0	4.472	29.928	76.368	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
0	4.472	29.928	76.368	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.86	0.344	0
0.516	3.956	32.852	70.86401	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	1.72	0.516	0
1.032	3.44	35.776	65.53201	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	1.72	0.516	0
1.032	3.44	35.776	65.53201	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	1.72	0.516	0
1.032	3.44	35.776	65.53201	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.86	0.516	0.516
1.204	4.988	24.08	86.172	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0.344	0.86
1.376	6.536	12.556	106.984	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0.344	0.86

1.376 6.536 12.556 106.984 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0.344 0.86
1.376 6.536 12.556 106.984 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1.548 1.376 0.516
1.892 16.856 24.08 82.216 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 3.268 2.408 0
2.408 27.004 35.604 57.62 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 3.268 2.408 0
2.408 27.004 35.604 57.62 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 3.268 2.408 0
2.408 27.004 35.604 57.62 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1.548 2.064 0
7.74 41.796 33.024 60.372 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 1.72 0
13.244 56.416 30.616 63.124 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 1.72 0
13.244 56.416 30.616 63.124 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 1.72 0
13.244 56.416 30.616 63.124 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 1.376 0 0 0.516 1.72 2.924
9.804 37.152 25.112 60.028 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 2.58 0 0 1.204 1.72 5.848
6.364 17.888 19.436 57.104 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 2.58 0 0 1.204 1.72 5.848
6.364 17.888 19.436 57.104 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 2.58 0 0 1.204 1.72 5.848
6.364 17.888 19.436 57.104 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 1.376 0 0 2.752 1.032 3.44
3.268 23.908 53.148 60.544 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 4.472 0.516 1.032
0.344 30.1 86.688 64.15601 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 4.472 0.516 1.032
0.344 30.1 86.688 64.15601 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 4.472 0.516 1.032
0.344 30.1 86.688 64.15601 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0.86 0.688 0 2.236 0.344 0.516
4.472 20.296 50.568 47.644 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 1.892 1.204 0 0 0 0
8.6 10.664 14.62 31.304 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 1.892 1.204 0 0 0 0
8.6 10.664 14.62 31.304 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 1.892 1.204 0 0 0 0
8.6 10.664 14.62 31.304 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0.86 0.688 0 1.032 0 0.86
4.3 15.136 12.9 21.672 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1.892 0 1.72
0 19.608 11.352 12.04 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1.892 0 1.72
0 19.608 11.352 12.04 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1.892 0 1.72
0 19.608 11.352 12.04 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0.172 1.376 0 0.86
0 9.804 25.112 44.376 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0.516 0.86 0 0
0 0 38.872 76.54 0 0 0 0
0 0 0 0 0 0 0 0

0 0
0 0 0 0 0.516 0.86 0 0
0 0 38.872 76.54 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0.516 0.86 0 0
0 0 38.872 76.54 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0.172 0.344 0 0.86
0 6.364 27.348 76.196 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0 1.72
0 12.9 15.652 75.85201 0 0 0 0
0 0 0 0 0 0 0 0
0 0
POPULATION2
0 0 0 0 0 0 0 2
0 15 18.2 88.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0 2
0 15 18.2 88.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0.6 1
1.6 13 23.6 109.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 1.2 0
3 11.2 29 130.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 1.2 0
3 11.2 29 130.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 1.2 0
3 11.2 29 130.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0.4 2.2 3.4
1.6 16.2 26.4 90.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1 3 6.6
0 21.2 24 50 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1 3 6.6
0 21.2 24 50 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1 3 6.6
0 21.2 24 50 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0.4 2.6 3.4
1.4 21.8 24.6 47.6 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 2.2 0.2
2.8 22.4 25.4 45 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 2.2 0.2
2.8 22.4 25.4 45 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 2.2 0.2
2.8 22.4 25.4 45 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 1.2 1.2
3.4 19 29.6 72 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0.2 0.2 2.4
4 15.6 34 98.8 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0.2 0.2 2.4
4 15.6 34 98.8 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0.2 0.2 2.4
4 15.6 34 98.8 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0 1.2
4.4 16.4 30.6 101.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0 0
4.6 17.2 27.4 103.4 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0 0
4.6 17.2 27.4 103.4 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0 0
4.6 17.2 27.4 103.4 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0 0
2.2 11.2 31 96 0 0 0 0
0 0 0 0 0 0 0 0
0 0

0 0 0 0 0 0 0 0
0 5.2 34.8 88.8 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0 0
0 5.2 34.8 88.8 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0 0
0 5.2 34.8 88.8 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1 0.4 0
0.6 4.6 38.2 82.4 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 2 0.6 0
1.2 4 41.6 76.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 2 0.6 0
1.2 4 41.6 76.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 2 0.6 0
1.2 4 41.6 76.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 1 0.6 0.6
1.4 5.8 28 100.2 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0.4 1
1.6 7.6 14.6 124.4 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0.4 1
1.6 7.6 14.6 124.4 0 0 0 0
0 0 0 0 0 0 0 0
0 0
0 0 0 0 0 0 0.4 1
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7.4 20.8 22.6 66.4 0 0 0 0
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0 0 0 0 0 5.2 0.6 1.2
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POPULATION3
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POPULATION4
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0	0	0	0	174.6	412.2	666.9	0	
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0	0	0	0	85.5	107.1	627.9	0	
0	0	0	0	0	0	0	0	
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POPULATION5
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POPULATION6
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0 0 0 0 0 0 1.1 0.1
1.4 11.2 12.7 22.5 0 0 0 0

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0	0						
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2	7.8	17	49.4	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.1	0.1	1.2
2	7.8	17	49.4	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.1	0.1	1.2
2	7.8	17	49.4	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0.6
2.2	8.2	15.3	50.6	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
2.3	8.6	13.7	51.7	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
2.3	8.6	13.7	51.7	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0	0
2.3	8.6	13.7	51.7	0	0	0	0
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0	0						
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1.1	5.6	15.5	48	0	0	0	0
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0	2.6	17.4	44.4	0	0	0	0
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0	0						
0	0	0	0	0	0	0	0
0	2.6	17.4	44.4	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
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0	2.6	17.4	44.4	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
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0.3	2.3	19.1	41.2	0	0	0	0
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0	0						
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0.6	2	20.8	38.1	0	0	0	0
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0	0						
0	0	0	0	0	1	0.3	0
0.6	2	20.8	38.1	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.5	0.3	0.3
0.7	2.9	14	50.1	0	0	0	0
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0	0						
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0.8	3.8	7.3	62.2	0	0	0	0
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0	0						
0	0	0	0	0	0	0.2	0.5
0.8	3.8	7.3	62.2	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	0.2	0.5
0.8	3.8	7.3	62.2	0	0	0	0
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0	0						
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1.1	9.8	14	47.8	0	0	0	0
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0	0						
0	0	0	0	0	1.9	1.4	0
1.4	15.7	20.7	33.5	0	0	0	0
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0	0						
0	0	0	0	0	1.9	1.4	0
1.4	15.7	20.7	33.5	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	1.9	1.4	0
1.4	15.7	20.7	33.5	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0.9	1.2	0
4.5	24.3	19.2	35.1	0	0	0	0
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0	0						
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7.7	32.8	17.8	36.7	0	0	0	0
0	0	0	0	0	0	0	0
0	0						
0	0	0	0	0	0	1	0
7.7	32.8	17.8	36.7	0	0	0	0
0	0	0	0	0	0	0	0
0	0						

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7.7 32.8 17.8 36.7 0 0 0 0
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0 7.5 9.1 44.1 0 0 0 0
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POPULATION7
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26.435 66.455 55569 27311 58059 122309 44825 380230
121371 16
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26.435 66.455 55569 27311 58059 122309 44825 380230
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357994 74505
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7420 0
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32.435 181.255 50793 610856 308735 3178496 948471 1117110
7420 0
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69.88 347.05 56283 209653 79672 103844 0 0
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0 0
0 0 0 0 0 0.05 0.015 0
0.03 1.E-01 1.04 1.905 1.765 6.585 36.035 1.55
1.53 15.05 2516 9605 28016 43469 175 0
0 0
0 0 0 0 0 0.05 0.015 0
0.03 1.E-01 1.04 1.905 1.765 6.585 36.035 1.55
1.53 15.05 2516 9605 28016 43469 175 0
0 0
0 0 0 0 0 0.025 0.015 0.015
0.035 0.145 0.7 2.505 2.76 11.48 40.96 27.1
4.66 9.809999 2144 12626 24286 34494 120104 151270
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7.785 4.575 1772 15648 20556 25519 240033 302539
22481 0
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0.04 0.19 0.365 3.11 3.755 16.375 45.885 52.645
7.785 4.575 1772 15648 20556 25519 240033 302539
22481 0
0 0 0 0 0 0 0.01 0.025
0.04 0.19 0.365 3.11 3.755 16.375 45.885 52.645
7.785 4.575 1772 15648 20556 25519 240033 302539
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0.055 0.49 0.7 2.39 3.325 26.675 37.46 38.215
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264151 2067123
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505820 4134247
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0.07 0.785 1.035 1.675 2.895 36.98 29.03 23.79
144.96 839.215 81037 167231 202011 49525 266674 621551
505820 4134247
0 0 0 0 0 0.095 0.07 0
0.07 0.785 1.035 1.675 2.895 36.98 29.03 23.79
144.96 839.215 81037 167231 202011 49525 266674 621551
505820 4134247
0 0 0 0 0 0.045 0.06 0
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111.425 688.085 73311 236004 352160 60457 178128 626154
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0.385 1.64 0.89 1.835 2.91 6.87 11.115 16.1
77.89 536.955 65586 304777 502310 71390 89583 630757
1290932 3511814
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0.385 1.64 0.89 1.835 2.91 6.87 11.115 16.1

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 77.89   536.955    65586    304777    502310    71390    89583    630757
1290932   3511814
     0       0       0       0       0       0    0.05       0
 0.385    1.64    0.89    1.835    2.91    6.87   11.115    16.1
 77.89   536.955    65586    304777    502310    71390    89583    630757
1290932   3511814
     0       0    0.04       0       0   0.015    0.05   0.085
 0.285    1.08    0.73    1.745    2.49   4.635    7.38   13.52
 42.07   294.52    41336    174749    276692    62277    62795    452745
 968879   3353601
     0       0   0.075       0       0   0.035    0.05    0.17
 0.185    0.52   0.565     1.66    2.07     2.4    3.65   10.94
 6.255   52.085    17086    44721    51075    53165    36007    274733
 646826   3195388
     0       0   0.075       0       0   0.035    0.05    0.17
 0.185    0.52   0.565     1.66    2.07     2.4    3.65   10.94
 6.255   52.085    17086    44721    51075    53165    36007    274733
 646826   3195388
     0       0   0.075       0       0   0.035    0.05    0.17
 0.185    0.52   0.565     1.66    2.07     2.4    3.65   10.94
 6.255   52.085    17086    44721    51075    53165    36007    274733
 646826   3195388
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  7.03    62.74    14097    33118    54364    51482    83732    385030
1111245   3472321
     0       0       0       0       0    0.13   0.015    0.03
  0.01   0.875    2.52    1.865   1.425   1.785  10.465   20.985
  7.81     73.4    11108    21515    57654    49799   131457    495327
1575665   3749254
     0       0       0       0       0    0.13   0.015    0.03
  0.01   0.875    2.52    1.865   1.425   1.785  10.465   20.985
  7.81     73.4    11108    21515    57654    49799   131457    495327
1575665   3749254
     0       0       0       0       0    0.13   0.015    0.03
  0.01   0.875    2.52    1.865   1.425   1.785  10.465   20.985
  7.81     73.4    11108    21515    57654    49799   131457    495327
1575665   3749254
     0       0   0.025    0.02       0   0.065    0.01   0.015
  0.13    0.59    1.47    1.385    1.23    2.17  11.395   33.615
 58.19    77.62     9453    22538    35246    72651   208798    992232
1855143   4794612
     0       0   0.055   0.035       0       0       0       0
  0.25    0.31   0.425     0.91   1.035    2.56  12.325    46.24
108.57    81.84     7799    23562    12839    95503   286139   1489138
2134621   5839970
     0       0   0.055   0.035       0       0       0       0
  0.25    0.31   0.425     0.91   1.035    2.56  12.325    46.24
108.57    81.84     7799    23562    12839    95503   286139   1489138
2134621   5839970
     0       0   0.055   0.035       0       0       0       0
  0.25    0.31   0.425     0.91   1.035    2.56  12.325    46.24
108.57    81.84     7799    23562    12839    95503   286139   1489138
2134621   5839970
     0       0   0.025    0.02       0    0.03       0   0.025
 0.125    0.44   0.375     0.63    1.24   2.085   10.53   34.725
73.475   77.265    37123    25951    16476    73091   165329    822067
1128816   3043298
     0       0       0       0       0   0.055       0    0.05
     0    0.57    0.33     0.35    1.44   1.615   8.735    23.21
38.375  72.68999    66446    28340    20114    50680    44519    154997
 123012    246625
     0       0       0       0       0   0.055       0    0.05
     0    0.57    0.33     0.35    1.44   1.615   8.735    23.21
38.375  72.68999    66446    28340    20114    50680    44519    154997
 123012    246625
     0       0       0       0       0   0.055       0    0.05
     0    0.57    0.33     0.35    1.44   1.615   8.735    23.21
38.375  72.68999    66446    28340    20114    50680    44519    154997
 123012    246625
     0       0       0       0   0.005    0.04       0   0.025
     0   0.285    0.73     1.29   1.885   3.455  18.885    44.84
34.305   65.395    44684    18611    31024    48774    45737    372300
  61506    123313
     0       0       0       0   0.015   0.025       0       0
     0       0    1.13    2.225    2.33   5.295   29.04   66.465
 30.23     58.1    22922     8881    41934    46868    46955    589604
     0       0
     0       0       0       0   0.015   0.025       0       0
     0       0    1.13    2.225    2.33   5.295   29.04   66.465
 30.23     58.1    22922     8881    41934    46868    46955    589604
     0       0
     0       0       0       0   0.015   0.025       0       0
     0       0    1.13    2.225    2.33   5.295   29.04   66.465
 30.23     58.1    22922     8881    41934    46868    46955    589604
     0       0
     0       0       0       0   0.005    0.01       0   0.025
     0   0.185   0.795    2.215     3.6   8.325  65.965   69.165
 48.53   48.415    18750    16074    47862    52523    95481    448756
  2995         0
     0       0       0       0       0       0       0    0.05
     0   0.375   0.455    2.205    4.87   11.35 102.895    71.87
66.825    38.73    14578    23266    53790    58178   144007    307909
  5990         0
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CROP SEASON AND SHARE
  1 PASTURE              90. 270.  0.4100
  2 STORED FORAGE       150. 240.  0.1300
  3 GRAINS              150. 240.  0.2100
  4 GRN LEAFY VEGETABLES 150. 240.  0.0020
  5 OTHER FOOD CROPS     150. 240.  0.0040
  6 LEGUMES AND SEEDS    150. 240.  0.1500
  7 ROOTS AND TUBERS     150. 240.  0.0030
WATERSHED DEFINITION  -- INITIAL AND ANNUAL WASHOFF AND INGESTION FACTORS
  1 Sr-89                     5.00E-06      0.0
  2 Sr-90                     5.00E-06      0.0
  3 Cs-134                    5.00E-06      0.0
  4 Cs-137                    5.00E-06      0.0
REGIONAL ECONOMIC DATA
  1 EXCLUSION      .493 .164    1393.0   12941.6  233218.6
  2 REGION_02      .493 .164    1393.0   12941.6  233218.6
  3 REGION_03      .678 .333    5221.1   21426.1  234994.2
  4 REGION_04      .678 .333    5221.1   21426.1  234994.0
  5 REGION_05      .327 .186    1943.4   10750.6  213214.4
  6 REGION_06      .322 .627     899.5    4085.8  193086.4
  7 REGION_07      .000 .000       0.0       0.0       0.0
  8 REGION_08      .000 .000       0.0       0.0       0.0
  9 REGION_09      .678 .333    5221.1   21426.1  234994.2
 10 REGION_10      .678 .333    5221.1   21426.1  234994.3
 11 REGION_11      .276 .142    2095.8   13396.2  229678.7
 12 REGION_12      .183 .521     830.5    4691.9  202840.3
 13 REGION_13      .000 .000       0.0       0.0       0.0
 14 REGION_14      .678 .333    5221.1   21426.1  234994.2
 15 REGION_15      .678 .333    5221.1   21426.1  234994.2
 16 REGION_16      .602 .286    5408.1   22914.7  269246.8
 17 REGION_17      .314 .127    2237.8   23715.1  334669.0
 18 REGION_18      .091 .237    1229.2   11059.4  235490.6
 19 REGION_19      .000 .000       0.0       0.0       0.0
 20 REGION_20      .678 .333    5221.1   21426.1  234994.2
 21 REGION_21      .678 .333    5221.1   21426.1  234994.3
 22 REGION_22      .443 .186    5799.5   26029.9  340928.5
 23 REGION_23      .191 .070    2361.9   36713.0  346735.1
 24 REGION_24      .086 .046    8196.1   43486.7  311650.0
 25 REGION_25      .000 .000       0.0       0.0       0.0
 26 REGION_26      .678 .333    5221.1   21426.1  234994.2
 27 REGION_27      .678 .333    5221.1   21426.1  234994.2
 28 REGION_28      .382 .132    5495.9   25692.3  350421.2
 29 REGION_29      .196 .040    2967.0   22984.9  277378.8
 30 REGION_30      .033 .000    2442.6   38497.5  262022.8
 31 REGION_31      .000 .000       0.0       0.0       0.0
 32 REGION_32      .000 .000       0.0       0.0       0.0
 33 REGION_33      .618 .273    4711.6   20378.6  235866.5
 34 REGION_34      .350 .008    2542.2   16061.5  244061.7
 35 REGION_35      .209 .027    4061.8   14682.1  247490.6
 36 REGION_36      .084 .000    6970.5   15611.0  256084.4
 37 REGION_37      .000 .000       0.0       0.0       0.0
 38 REGION_38      .666 .322    4970.1   20869.8  234877.8
 39 REGION_39      .346 .000    2396.9   15619.7  239829.4
 40 REGION_40      .346 .001    2390.0   15608.8  239997.2
 41 REGION_41      .444 .060    2595.9   11246.9  230213.9
 42 REGION_42      .467 .013    4158.9   10106.0  217217.5
 43 REGION_43      .000 .000       0.0       0.0       0.0
 44 REGION_44      .493 .164    1393.0   12941.6  233218.6
 45 REGION_45      .303 .315     900.8   13187.7  273921.2
 46 REGION_46      .289 .327     863.3   13206.5  277027.4
 47 REGION_47      .515 .040    2730.9    9172.1  227936.3
 48 REGION_48      .361 .001    3979.4    6895.0  200594.0
 49 REGION_49      .000 .000       0.0       0.0       0.0
 50 REGION_50      .493 .164    1393.0   12941.6  233218.6
 51 REGION_51      .301 .317     895.6   13190.3  274356.8
 52 REGION_52      .289 .327     863.3   13206.5  277026.9
 53 REGION_53      .455 .043    1230.4    9361.0  267918.8
 54 REGION_54      .277 .002    1363.4    7100.2  199006.0
 55 REGION_55      .000 .000       0.0       0.0       0.0
 56 REGION_56      .493 .164    1393.0   12941.6  233218.6
 57 REGION_57      .309 .310     916.4   13179.9  272631.7
 58 REGION_58      .257 .249    1320.9   14798.8  289775.8
 59 REGION_59      .157 .022     726.8   14565.3  285095.8
 60 REGION_60      .311 .043    1605.9    7828.6  198615.7
 61 REGION_61      .000 .000       0.0       0.0       0.0
 62 REGION_62      .000 .000       0.0       0.0       0.0
 63 REGION_63      .329 .295     966.3   13154.9  268506.3
 64 REGION_64      .246 .223    1476.6   15340.4  294111.0
 65 REGION_65      .202 .075    1278.1   18214.2  364175.0
 66 REGION_66      .272 .049    1082.0    6868.9  186989.3
 67 REGION_67      .000 .000       0.0       0.0       0.0
 68 REGION_68      .493 .164    1393.0   12941.6  233218.6
 69 REGION_69      .455 .194    1295.2   12990.5  241305.4
 70 REGION_70      .236 .177    1734.2   16172.0  299488.0
 71 REGION_71      .453 .304    1115.2   15936.2  283157.3
 72 REGION_72      .463 .046     596.2    4628.9  173031.0
 73 REGION_73      .000 .000       0.0       0.0       0.0
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 74 REGION_74      .493 .164   1393.0   12941.6  233218.6
 75 REGION_75      .493 .164   1393.0   12941.6  233218.6
 76 REGION_76      .462 .155   1486.6   13484.1  241853.6
 77 REGION_77      .464 .393   1631.5   10633.0  225960.7
 78 REGION_78      .699 .059    633.3    5669.1  198117.1
 79 REGION_79      .000 .000      0.0       0.0       0.0
 80 REGION_80      .493 .164   1393.0   12941.6  233218.6
 81 REGION_81      .493 .164   1393.0   12941.6  233218.6
 82 REGION_82      .493 .164   1393.0   12941.6  233218.8
 83 REGION_83      .409 .423   1596.2    9559.0  216875.6
 84 REGION_84      .542 .249   1032.8    6850.8  215564.2
 85 REGION_85      .000 .000      0.0       0.0       0.0
 86 REGION_86      .493 .164   1393.0   12941.6  233218.6
 87 REGION_87      .500 .170   1541.8   13271.4  233287.6
 88 REGION_88      .498 .168   1487.8   13151.6  233262.7
 89 REGION_89      .369 .357   1697.9   10004.5  211758.9
 90 REGION_90      .132 .268    446.8    4939.7  194466.8
 91 REGION_91      .000 .000      0.0       0.0       0.0
 92 REGION_92      .493 .164   1393.0   12941.6  233218.6
 93 REGION_93      .678 .333   5221.1   21426.1  234994.1
 94 REGION_94      .678 .333   5217.0   21417.1  234992.2
 95 REGION_95      .406 .260   2375.4   11626.5  216363.5
 96 REGION_96      .342 .453   1050.6    4558.4  200112.7
 97 REGION_97      .000 .000      0.0       0.0       0.0


******** BEGINNING OF CHANGE CASE   1 USER INPUT ********
     *
     * CSFACT - Cloudshine shielding factor
  273  SECSFACT001      1.
********   RECORD NUMBER   273 REPLACES RECORD NUMBER    25   ********
  274  SECSFACT002      0.6
********   RECORD NUMBER   274 REPLACES RECORD NUMBER    26   ********
  275  SECSFACT003      0.5
********   RECORD NUMBER   275 REPLACES RECORD NUMBER    27   ********
     *
     * PROTIN - Inhalation protection factor
  276  SEPROTIN001      0.98
********   RECORD NUMBER   276 REPLACES RECORD NUMBER    28   ********
  277  SEPROTIN002      0.46
********   RECORD NUMBER   277 REPLACES RECORD NUMBER    29   ********
  278  SEPROTIN003      0.33
********   RECORD NUMBER   278 REPLACES RECORD NUMBER    30   ********
     *
     * BRRATE - Breathing rates
  279  SEBRRATE001      2.66E-04
********   RECORD NUMBER   279 REPLACES RECORD NUMBER    31   ********
  280  SEBRRATE002      2.66E-04
********   RECORD NUMBER   280 REPLACES RECORD NUMBER    32   ********
  281  SEBRRATE003      2.66E-04
********   RECORD NUMBER   281 REPLACES RECORD NUMBER    33   ********
     *
     * SKPFAC - skin protection factors
  282  SESKPFAC001      0.98
********   RECORD NUMBER   282 REPLACES RECORD NUMBER    34   ********
  283  SESKPFAC002      0.46
********   RECORD NUMBER   283 REPLACES RECORD NUMBER    35   ********
  284  SESKPFAC003      0.33
********   RECORD NUMBER   284 REPLACES RECORD NUMBER    36   ********
     *
     * GSHFAC - groundshine shielding factors
  285  SEGSHFAC001      0.5
********   RECORD NUMBER   285 REPLACES RECORD NUMBER    37   ********
  286  SEGSHFAC002      0.18
********   RECORD NUMBER   286 REPLACES RECORD NUMBER    38   ********
  287  SEGSHFAC003      0.1
********   RECORD NUMBER   287 REPLACES RECORD NUMBER    39   ********
     *
     * EANAM2 - Name of emergency response cohort
  288  EZEANAM2001      '0-10 Early Evacuees'
********   RECORD NUMBER   288 REPLACES RECORD NUMBER    42   ********
     *
     * WTFRAC - weighting fraction applied to results of emergency response cohort
  289  EZWTFRAC001      0.2
********   RECORD NUMBER   289 REPLACES RECORD NUMBER    44   ********
     *
     * TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
  290  TRAVELPOINT      CENTERPOINT
********   RECORD NUMBER   290 REPLACES RECORD NUMBER    46   ********
     *
     * ESPEED - evacuee travel speed during the three phases of evacuation
  291  EZESPEED001      8.941
********   RECORD NUMBER   291 REPLACES RECORD NUMBER    47   ********
  292  EZESPEED002      4.47
********   RECORD NUMBER   292 REPLACES RECORD NUMBER    48   ********
  293  EZESPEED003      8.941
********   RECORD NUMBER   293 REPLACES RECORD NUMBER    49   ********
     *
     * ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
  294  EZESPMUL001      0.7
********   RECORD NUMBER   294 REPLACES RECORD NUMBER    50   ********
  295  EZESPMUL002      0.7
********   RECORD NUMBER   295 REPLACES RECORD NUMBER    51   ********
  296  EZESPMUL003      0.7
********   RECORD NUMBER   296 REPLACES RECORD NUMBER    52   ********
     *
     * REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
  297  EZREFPNT001      ALARM
********   RECORD NUMBER   297 REPLACES RECORD NUMBER    53   ********
     *
     * DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
  298  EZDURBEG001      3600.
********   RECORD NUMBER   298 REPLACES RECORD NUMBER    54   ********
     *
     * DURMID - duration of middle phase of evacuation, in seconds.
  299  EZDURMID001      7200.
********   RECORD NUMBER   299 REPLACES RECORD NUMBER    55   ********
     *
     * NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
  300  EZNUMEVA001      18
```

******** RECORD NUMBER 300 REPLACES RECORD NUMBER 56 ********
*
* DLTSHL - delay from reference time point to when individual takes shelter. DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
301 EZDLTSHL001 1800.
******** RECORD NUMBER 301 REPLACES RECORD NUMBER 57 ********
302 EZDLTSHL002 1800.
******** RECORD NUMBER 302 REPLACES RECORD NUMBER 58 ********
303 EZDLTSHL003 1800.
******** RECORD NUMBER 303 REPLACES RECORD NUMBER 59 ********
304 EZDLTSHL004 1800.
******** RECORD NUMBER 304 REPLACES RECORD NUMBER 60 ********
305 EZDLTSHL005 1800.
******** RECORD NUMBER 305 REPLACES RECORD NUMBER 61 ********
306 EZDLTSHL006 1800.
******** RECORD NUMBER 306 REPLACES RECORD NUMBER 62 ********
307 EZDLTSHL007 1800.
******** RECORD NUMBER 307 REPLACES RECORD NUMBER 63 ********
308 EZDLTSHL008 1800.
******** RECORD NUMBER 308 REPLACES RECORD NUMBER 64 ********
309 EZDLTSHL009 1800.
******** RECORD NUMBER 309 REPLACES RECORD NUMBER 65 ********
310 EZDLTSHL010 1800.
******** RECORD NUMBER 310 REPLACES RECORD NUMBER 66 ********
311 EZDLTSHL011 1800.
******** RECORD NUMBER 311 REPLACES RECORD NUMBER 67 ********
312 EZDLTSHL012 1800.
******** RECORD NUMBER 312 REPLACES RECORD NUMBER 68 ********
313 EZDLTSHL013 1800.
******** RECORD NUMBER 313 REPLACES RECORD NUMBER 69 ********
314 EZDLTSHL014 1800.
******** RECORD NUMBER 314 REPLACES RECORD NUMBER 70 ********
315 EZDLTSHL015 1800.
******** RECORD NUMBER 315 REPLACES RECORD NUMBER 71 ********
316 EZDLTSHL016 1800.
******** RECORD NUMBER 316 REPLACES RECORD NUMBER 72 ********
317 EZDLTSHL017 1800.
******** RECORD NUMBER 317 REPLACES RECORD NUMBER 73 ********
318 EZDLTSHL018 1800.
******** RECORD NUMBER 318 REPLACES RECORD NUMBER 74 ********
*
* DLTEVA -Delay time to begin evacuation
319 EZDLTEVA001 1800.
******** RECORD NUMBER 319 REPLACES RECORD NUMBER 75 ********
320 EZDLTEVA002 1800.
******** RECORD NUMBER 320 REPLACES RECORD NUMBER 76 ********
321 EZDLTEVA003 1800.
******** RECORD NUMBER 321 REPLACES RECORD NUMBER 77 ********
322 EZDLTEVA004 1800.
******** RECORD NUMBER 322 REPLACES RECORD NUMBER 78 ********
323 EZDLTEVA005 1800.
******** RECORD NUMBER 323 REPLACES RECORD NUMBER 79 ********
324 EZDLTEVA006 1800.
******** RECORD NUMBER 324 REPLACES RECORD NUMBER 80 ********
325 EZDLTEVA007 1800.
******** RECORD NUMBER 325 REPLACES RECORD NUMBER 81 ********
326 EZDLTEVA008 1800.
******** RECORD NUMBER 326 REPLACES RECORD NUMBER 82 ********
327 EZDLTEVA009 1800.
******** RECORD NUMBER 327 REPLACES RECORD NUMBER 83 ********
328 EZDLTEVA010 1800.
******** RECORD NUMBER 328 REPLACES RECORD NUMBER 84 ********
329 EZDLTEVA011 1800.
******** RECORD NUMBER 329 REPLACES RECORD NUMBER 85 ********
330 EZDLTEVA012 1800.
******** RECORD NUMBER 330 REPLACES RECORD NUMBER 86 ********
331 EZDLTEVA013 1800.
******** RECORD NUMBER 331 REPLACES RECORD NUMBER 87 ********
332 EZDLTEVA014 1800.
******** RECORD NUMBER 332 REPLACES RECORD NUMBER 88 ********
333 EZDLTEVA015 1800.
******** RECORD NUMBER 333 REPLACES RECORD NUMBER 89 ********
334 EZDLTEVA016 1800.
******** RECORD NUMBER 334 REPLACES RECORD NUMBER 90 ********
335 EZDLTEVA017 1800.
******** RECORD NUMBER 335 REPLACES RECORD NUMBER 91 ********
336 EZDLTEVA018 1800.
******** RECORD NUMBER 336 REPLACES RECORD NUMBER 92 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
337 EZESPGRD001 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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******** RECORD NUMBER 337 REPLACES RECORD NUMBER 93 ********
338 EZESPGRD002 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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******** RECORD NUMBER 338 REPLACES RECORD NUMBER 94 ********
339 EZESPGRD003 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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******** RECORD NUMBER 339 REPLACES RECORD NUMBER 95 ********
340 EZESPGRD004 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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******** RECORD NUMBER 340 REPLACES RECORD NUMBER 96 ********
341 EZESPGRD005 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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******** RECORD NUMBER 341 REPLACES RECORD NUMBER 97 ********
342 EZESPGRD006 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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******** RECORD NUMBER 342 REPLACES RECORD NUMBER 98 ********
343 EZESPGRD007 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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******** RECORD NUMBER 343 REPLACES RECORD NUMBER 99 ********
344 EZESPGRD008 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 344 REPLACES RECORD NUMBER 100 ********
345 EZESPGRD009 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 345 REPLACES RECORD NUMBER 101 ********
346 EZESPGRD010 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 346 REPLACES RECORD NUMBER 102 ********
347 EZESPGRD011 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.

1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 347 REPLACES RECORD NUMBER 103 ********
348 EZESPGRD012 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 348 REPLACES RECORD NUMBER 104 ********
349 EZESPGRD013 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 349 REPLACES RECORD NUMBER 105 ********
350 EZESPGRD014 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 350 REPLACES RECORD NUMBER 106 ********
351 EZESPGRD015 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 351 REPLACES RECORD NUMBER 107 ********
352 EZESPGRD016 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 352 REPLACES RECORD NUMBER 108 ********
353 EZESPGRD017 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 353 REPLACES RECORD NUMBER 109 ********
354 EZESPGRD018 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 354 REPLACES RECORD NUMBER 110 ********
355 EZESPGRD019 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 355 REPLACES RECORD NUMBER 111 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
356 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 356 REPLACES RECORD NUMBER 112 ********
357 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 357 REPLACES RECORD NUMBER 113 ********
358 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 4 1 1
******** RECORD NUMBER 358 REPLACES RECORD NUMBER 114 ********
359 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2 1
4 4 4 4 4 1 1
******** RECORD NUMBER 359 REPLACES RECORD NUMBER 115 ********
360 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 1
1 1 1 1 1 1 1
******** RECORD NUMBER 360 REPLACES RECORD NUMBER 116 ********
361 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4 4
4 4 1 1 1 1 1
******** RECORD NUMBER 361 REPLACES RECORD NUMBER 117 ********
362 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 4 4 1 4 4 4 4 4
4 1 1 1 1 1 1
******** RECORD NUMBER 362 REPLACES RECORD NUMBER 118 ********
363 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 4 1 4 4 4 2 2
1 1 1 1 1 2 2
******** RECORD NUMBER 363 REPLACES RECORD NUMBER 119 ********
364 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 4 4 1 4 2 1 4 1
4 1 1 1 2 2 2
******** RECORD NUMBER 364 REPLACES RECORD NUMBER 120 ********
365 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 4 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1 4
4 4 2 2 1 1 1
******** RECORD NUMBER 365 REPLACES RECORD NUMBER 121 ********
366 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 4 4 2 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4 4
4 2 2 1 4 2 2
******** RECORD NUMBER 366 REPLACES RECORD NUMBER 122 ********
367 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 2 1 2 1 2 2 1 4 4 4 4 2 2 2 1 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4 2
1 1 2 2 1 4 2
******** RECORD NUMBER 367 REPLACES RECORD NUMBER 123 ********
368 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 4 4 4 1 2 1 4 4 4 4 2 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4 2
1 4 2 1 4 1 1
******** RECORD NUMBER 368 REPLACES RECORD NUMBER 124 ********
369 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4 1
4 1 1 1 1 1 1
******** RECORD NUMBER 369 REPLACES RECORD NUMBER 125 ********
370 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2 1
1 2 2 2 1 2 1
******** RECORD NUMBER 370 REPLACES RECORD NUMBER 126 ********
371 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 371 REPLACES RECORD NUMBER 127 ********
372 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 372 REPLACES RECORD NUMBER 128 ********
373 EZIDIREC018 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 373 REPLACES RECORD NUMBER 129 ********
374 EZIDIREC019 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 374 REPLACES RECORD NUMBER 130 ********
*
* LASMOV - The outermost spatial interval of the evacuation movement zone.
375 EZLASMOV001 19
******** RECORD NUMBER 375 REPLACES RECORD NUMBER 131 ********
*
* EFFACY, KI Ingestion
376 EZEFFACY001 0.7
******** RECORD NUMBER 376 REPLACES RECORD NUMBER 269 ********
*
* POPFRAC, KI Ingestion
377 EZPOPFRC001 0.
******** RECORD NUMBER 377 REPLACES RECORD NUMBER 270 ********
.
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE 1 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE 1
NUMBER OF RECORDS CHANGED = 105
NUMBER OF RECORDS ADDED = 0
****

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	4	4	2	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	2	1	2	1	2	2	1	4	4	4
13	1	1	1	1	1	2	1	4	4	4	1	2	1	4	4	4
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1
4	4	2	2	1	1	1	1	4	4	4	2	2	1	4	4	2
5	1	1	1	1	2	1	4	4	2	2	1	4	4	2	2	1
6	2	2	1	1	1	1	2	1	1	4	2	2	1	2	1	4
7	1	1	1	1	4	4	2	2	1	1	4	4	1	1	4	1
8	2	1	4	4	2	1	2	1	2	1	4	2	1	4	2	1
9	1	2	1	4	2	1	2	1	2	1	1	4	1	4	1	4
10	2	2	2	2	1	2	1	4	4	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	4	2	2	2	1	2	1	1	1	1	4	1	1	1	1	4
13	4	2	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1	1
4	1	4	4	2	1	1	4	2	1	4	4	4	4	4	1	1
5	4	4	2	2	1	4	4	2	1	1	1	1	1	1	1	1
6	4	2	2	1	4	4	4	4	4	4	4	1	1	1	1	1
7	4	4	4	1	4	4	4	4	4	4	1	1	1	1	1	1
8	1	4	4	1	4	4	4	2	2	1	1	1	1	1	2	2
9	4	4	4	1	4	2	1	4	1	4	1	1	1	2	2	2
10	4	2	2	1	1	1	1	1	4	4	4	2	2	1	1	1
11	4	4	4	2	2	2	1	4	4	4	2	2	1	4	2	2
12	4	4	2	2	2	2	1	4	2	1	1	2	2	1	4	2
13	4	4	2	2	2	1	1	4	2	1	4	2	1	4	1	1
14	1	4	4	2	2	1	1	4	1	4	1	1	1	1	1	1
15	1	4	2	1	2	1	1	2	1	1	2	2	2	1	2	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

THE KI MODEL IS IN EFFECT

```
******** BEGINNING OF CHANGE CASE   2 USER INPUT ********
     *
     * CSFACT - Cloudshine shielding factor
  378  SECSFACT001      1.
********   RECORD NUMBER   378 REPLACES RECORD NUMBER    25   ********
  379  SECSFACT002      0.6
********   RECORD NUMBER   379 REPLACES RECORD NUMBER    26   ********
  380  SECSFACT003      0.5
********   RECORD NUMBER   380 REPLACES RECORD NUMBER    27   ********
     *
     * PROTIN - Inhalation protection factor
  381  SEPROTIN001      0.98
********   RECORD NUMBER   381 REPLACES RECORD NUMBER    28   ********
  382  SEPROTIN002      0.46
********   RECORD NUMBER   382 REPLACES RECORD NUMBER    29   ********
  383  SEPROTIN003      0.33
********   RECORD NUMBER   383 REPLACES RECORD NUMBER    30   ********
     *
     * BRRATE - Breathing rates
  384  SEBRRATE001      2.66E-04
********   RECORD NUMBER   384 REPLACES RECORD NUMBER    31   ********
  385  SEBRRATE002      2.66E-04
```

******** RECORD NUMBER 385 REPLACES RECORD NUMBER 32 ********
386 SEBRRATE003 2.66E-04
******** RECORD NUMBER 386 REPLACES RECORD NUMBER 33 ********
*
* SKPFAC - skin protection factors
387 SESKPFAC001 0.98
******** RECORD NUMBER 387 REPLACES RECORD NUMBER 34 ********
388 SESKPFAC002 0.46
******** RECORD NUMBER 388 REPLACES RECORD NUMBER 35 ********
389 SESKPFAC003 0.33
******** RECORD NUMBER 389 REPLACES RECORD NUMBER 36 ********
*
* GSHFAC - groundshine shielding factors
390 SEGSHFAC001 0.5
******** RECORD NUMBER 390 REPLACES RECORD NUMBER 37 ********
391 SEGSHFAC002 0.18
******** RECORD NUMBER 391 REPLACES RECORD NUMBER 38 ********
392 SEGSHFAC003 0.1
******** RECORD NUMBER 392 REPLACES RECORD NUMBER 39 ********
*
* EANAM2 - Name of emergency response cohort
393 EZEANAM2001 '0-10 Public'
******** RECORD NUMBER 393 REPLACES RECORD NUMBER 42 ********
*
* WTFRAC - weighting fraction applied to results of emergency response cohort
394 EZWTFRAC001 0.517
******** RECORD NUMBER 394 REPLACES RECORD NUMBER 44 ********
*
* TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
395 TRAVELPOINT CENTERPOINT
******** RECORD NUMBER 395 REPLACES RECORD NUMBER 46 ********
*
* ESPEED - evacuee travel speed during the three phases of evacuation
396 EZESPEED001 2.235
******** RECORD NUMBER 396 REPLACES RECORD NUMBER 47 ********
397 EZESPEED002 0.894
******** RECORD NUMBER 397 REPLACES RECORD NUMBER 48 ********
398 EZESPEED003 8.941
******** RECORD NUMBER 398 REPLACES RECORD NUMBER 49 ********
*
* ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
399 EZESPMUL001 0.7
******** RECORD NUMBER 399 REPLACES RECORD NUMBER 50 ********
400 EZESPMUL002 0.7
******** RECORD NUMBER 400 REPLACES RECORD NUMBER 51 ********
401 EZESPMUL003 0.7
******** RECORD NUMBER 401 REPLACES RECORD NUMBER 52 ********
*
* REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
402 EZREFPNT001 ALARM
******** RECORD NUMBER 402 REPLACES RECORD NUMBER 53 ********
*
* DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
403 EZDURBEG001 900.
******** RECORD NUMBER 403 REPLACES RECORD NUMBER 54 ********
*
* DURMID - duration of middle phase of evacuation, in seconds.
404 EZDURMID001 10800.
******** RECORD NUMBER 404 REPLACES RECORD NUMBER 55 ********
*
* NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
405 EZNUMEVA001 18
******** RECORD NUMBER 405 REPLACES RECORD NUMBER 56 ********
*
* DLTSHL - delay from reference time point to when individual takes shelter. DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
406 EZDLTSHL001 3600.
******** RECORD NUMBER 406 REPLACES RECORD NUMBER 57 ********
407 EZDLTSHL002 3600.
******** RECORD NUMBER 407 REPLACES RECORD NUMBER 58 ********
408 EZDLTSHL003 3600.
******** RECORD NUMBER 408 REPLACES RECORD NUMBER 59 ********
409 EZDLTSHL004 3600.
******** RECORD NUMBER 409 REPLACES RECORD NUMBER 60 ********
410 EZDLTSHL005 3600.
******** RECORD NUMBER 410 REPLACES RECORD NUMBER 61 ********
411 EZDLTSHL006 3600.
******** RECORD NUMBER 411 REPLACES RECORD NUMBER 62 ********
412 EZDLTSHL007 3600.
******** RECORD NUMBER 412 REPLACES RECORD NUMBER 63 ********
413 EZDLTSHL008 3600.
******** RECORD NUMBER 413 REPLACES RECORD NUMBER 64 ********
414 EZDLTSHL009 3600.
******** RECORD NUMBER 414 REPLACES RECORD NUMBER 65 ********
415 EZDLTSHL010 3600.
******** RECORD NUMBER 415 REPLACES RECORD NUMBER 66 ********
416 EZDLTSHL011 3600.
******** RECORD NUMBER 416 REPLACES RECORD NUMBER 67 ********
417 EZDLTSHL012 3600.
******** RECORD NUMBER 417 REPLACES RECORD NUMBER 68 ********
418 EZDLTSHL013 3600.
******** RECORD NUMBER 418 REPLACES RECORD NUMBER 69 ********
419 EZDLTSHL014 3600.
******** RECORD NUMBER 419 REPLACES RECORD NUMBER 70 ********
420 EZDLTSHL015 3600.
******** RECORD NUMBER 420 REPLACES RECORD NUMBER 71 ********
421 EZDLTSHL016 3600.
******** RECORD NUMBER 421 REPLACES RECORD NUMBER 72 ********
422 EZDLTSHL017 3600.
******** RECORD NUMBER 422 REPLACES RECORD NUMBER 73 ********
423 EZDLTSHL018 3600.
******** RECORD NUMBER 423 REPLACES RECORD NUMBER 74 ********
*
* DLTEVA -Delay time to begin evacuation
424 EZDLTEVA001 7200.
******** RECORD NUMBER 424 REPLACES RECORD NUMBER 75 ********
425 EZDLTEVA002 7200.
******** RECORD NUMBER 425 REPLACES RECORD NUMBER 76 ********
426 EZDLTEVA003 7200.
******** RECORD NUMBER 426 REPLACES RECORD NUMBER 77 ********
427 EZDLTEVA004 7200.

******** RECORD NUMBER 427 REPLACES RECORD NUMBER 78 ********
428 EZDLTEVA005 7200.
******** RECORD NUMBER 428 REPLACES RECORD NUMBER 79 ********
429 EZDLTEVA006 7200.
******** RECORD NUMBER 429 REPLACES RECORD NUMBER 80 ********
430 EZDLTEVA007 7200.
******** RECORD NUMBER 430 REPLACES RECORD NUMBER 81 ********
431 EZDLTEVA008 7200.
******** RECORD NUMBER 431 REPLACES RECORD NUMBER 82 ********
432 EZDLTEVA009 7200.
******** RECORD NUMBER 432 REPLACES RECORD NUMBER 83 ********
433 EZDLTEVA010 7200.
******** RECORD NUMBER 433 REPLACES RECORD NUMBER 84 ********
434 EZDLTEVA011 7200.
******** RECORD NUMBER 434 REPLACES RECORD NUMBER 85 ********
435 EZDLTEVA012 7200.
******** RECORD NUMBER 435 REPLACES RECORD NUMBER 86 ********
436 EZDLTEVA013 7200.
******** RECORD NUMBER 436 REPLACES RECORD NUMBER 87 ********
437 EZDLTEVA014 7200.
******** RECORD NUMBER 437 REPLACES RECORD NUMBER 88 ********
438 EZDLTEVA015 7200.
******** RECORD NUMBER 438 REPLACES RECORD NUMBER 89 ********
439 EZDLTEVA016 7200.
******** RECORD NUMBER 439 REPLACES RECORD NUMBER 90 ********
440 EZDLTEVA017 7200.
******** RECORD NUMBER 440 REPLACES RECORD NUMBER 91 ********
441 EZDLTEVA018 7200.
******** RECORD NUMBER 441 REPLACES RECORD NUMBER 92 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
442 EZESPGRD001 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 442 REPLACES RECORD NUMBER 93 ********
443 EZESPGRD002 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 443 REPLACES RECORD NUMBER 94 ********
444 EZESPGRD003 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 444 REPLACES RECORD NUMBER 95 ********
445 EZESPGRD004 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 445 REPLACES RECORD NUMBER 96 ********
446 EZESPGRD005 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 446 REPLACES RECORD NUMBER 97 ********
447 EZESPGRD006 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 447 REPLACES RECORD NUMBER 98 ********
448 EZESPGRD007 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 448 REPLACES RECORD NUMBER 99 ********
449 EZESPGRD008 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 449 REPLACES RECORD NUMBER 100 ********
450 EZESPGRD009 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 450 REPLACES RECORD NUMBER 101 ********
451 EZESPGRD010 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 451 REPLACES RECORD NUMBER 102 ********
452 EZESPGRD011 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 452 REPLACES RECORD NUMBER 103 ********
453 EZESPGRD012 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 453 REPLACES RECORD NUMBER 104 ********
454 EZESPGRD013 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 454 REPLACES RECORD NUMBER 105 ********
455 EZESPGRD014 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 455 REPLACES RECORD NUMBER 106 ********
456 EZESPGRD015 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 456 REPLACES RECORD NUMBER 107 ********
457 EZESPGRD016 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 457 REPLACES RECORD NUMBER 108 ********
458 EZESPGRD017 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 458 REPLACES RECORD NUMBER 109 ********
459 EZESPGRD018 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 459 REPLACES RECORD NUMBER 110 ********
460 EZESPGRD019 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 460 REPLACES RECORD NUMBER 111 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
461 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 461 REPLACES RECORD NUMBER 112 ********
462 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 462 REPLACES RECORD NUMBER 113 ********
463 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 4 1 1
******** RECORD NUMBER 463 REPLACES RECORD NUMBER 114 ********
464 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2 1
4 4 4 4 4 1 1
******** RECORD NUMBER 464 REPLACES RECORD NUMBER 115 ********
465 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 1
1 1 1 1 1 1 1
******** RECORD NUMBER 465 REPLACES RECORD NUMBER 116 ********
466 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4 4
4 4 1 1 1 1 1
******** RECORD NUMBER 466 REPLACES RECORD NUMBER 117 ********
467 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 4 4 1 4 4 4 4 4
4 1 1 1 1 1 1

```
******** RECORD NUMBER 467 REPLACES RECORD NUMBER 118 ********
 468 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 4 1 4 4 4 2 2
1 1 1 1 1 2 2
******** RECORD NUMBER 468 REPLACES RECORD NUMBER 119 ********
 469 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 4 4 1 4 2 1 4 1
4 1 1 1 2 2 2
******** RECORD NUMBER 469 REPLACES RECORD NUMBER 120 ********
 470 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 4 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1 4
4 4 2 2 1 1 1
******** RECORD NUMBER 470 REPLACES RECORD NUMBER 121 ********
 471 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 4 4 2 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4 4
4 2 2 1 4 2 2
******** RECORD NUMBER 471 REPLACES RECORD NUMBER 122 ********
 472 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 2 1 2 1 2 2 1 4 4 4 4 2 2 2 1 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4 2
1 1 2 2 1 4 2
******** RECORD NUMBER 472 REPLACES RECORD NUMBER 123 ********
 473 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 4 4 4 1 2 1 4 4 4 4 2 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4 2
1 4 2 1 4 1 1
******** RECORD NUMBER 473 REPLACES RECORD NUMBER 124 ********
 474 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4 1
4 1 1 1 1 1 1
******** RECORD NUMBER 474 REPLACES RECORD NUMBER 125 ********
 475 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2 1
1 2 2 2 1 2 1
******** RECORD NUMBER 475 REPLACES RECORD NUMBER 126 ********
 476 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 476 REPLACES RECORD NUMBER 127 ********
 477 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 477 REPLACES RECORD NUMBER 128 ********
 478 EZIDIREC018 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 478 REPLACES RECORD NUMBER 129 ********
 479 EZIDIREC019 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 479 REPLACES RECORD NUMBER 130 ********
    *
    * LASMOV - The outermost spatial interval of the evacuation movement zone.
 480 EZLASMOV001 19
******** RECORD NUMBER 480 REPLACES RECORD NUMBER 131 ********
    *
    * EFFACY, KI Ingestion
 481 EZEFFACY001 0.7
******** RECORD NUMBER 481 REPLACES RECORD NUMBER 269 ********
    *
    * POPFRAC, KI Ingestion
 482 EZPOPFRC001 1.
******** RECORD NUMBER 482 REPLACES RECORD NUMBER 270 ********
    .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE 2 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE 2
NUMBER OF RECORDS CHANGED = 105
NUMBER OF RECORDS ADDED = 0
************************************************************************************************************
```

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	4	4	2	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	2	1	2	1	2	2	1	4	4	4
13	1	1	1	1	1	2	1	4	4	4	1	2	1	4	4	4
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1

```
  4     4  2  2  1  1  1  1  4  4  4  2  2  1  4  4  2
  5     1  1  1  1  2  1  4  4  2  2  1  4  4  2  2  1
  6     2  2  1  1  1  1  2  1  1  4  2  2  1  2  1  4
  7     1  1  1  1  4  4  2  2  1  1  4  4  1  1  4  1
  8     2  1  4  4  2  1  2  1  2  1  4  2  1  4  2  1
  9     1  2  1  4  2  1  2  1  2  1  1  4  1  4  1  4
 10     2  2  2  2  1  2  1  4  4  1  1  1  4  2  1  4
 11     1  1  4  2  1  4  1  4  2  1  4  1  1  2  1  1
 12     4  2  2  2  1  2  1  1  1  1  4  1  1  1  1  4
 13     4  2  2  1  4  4  1  1  4  2  1  1  2  1  2  1
 14     1  1  1  1  1  1  1  1  1  1  1  1  1  1  2  1
 15     1  1  1  1  1  1  1  1  1  1  1  4  1  2  1  1
 16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD    49 50 51 52 53 54 55 56 57 58 59 60 61 62 63 64
  1     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3     1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  1
  4     1  4  4  2  1  1  4  2  1  4  4  4  4  4  1  1
  5     4  4  2  2  1  4  4  2  1  1  1  1  1  1  1  1
  6     4  2  2  1  4  4  4  4  4  4  4  1  1  1  1  1
  7     4  4  4  1  4  4  4  4  4  4  1  1  1  1  1  1
  8     1  4  4  1  4  4  4  2  2  1  1  1  1  1  2  2
  9     4  4  4  1  4  2  1  4  1  4  1  1  1  2  2  2
 10     4  2  2  1  1  1  1  1  4  4  4  2  2  1  1  1
 11     4  4  4  2  2  2  1  4  4  4  2  2  1  4  2  2
 12     4  4  2  2  2  2  1  4  2  1  1  2  2  1  4  2
 13     4  4  2  2  2  1  1  4  2  1  4  2  1  4  1  1
 14     1  4  4  2  2  1  1  4  1  4  1  1  1  1  1  1
 15     1  4  2  1  2  1  1  2  1  1  2  2  2  1  2  1
 16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE   3 USER INPUT ********
      *
      * CSFACT - Cloudshine shielding factor
  483  SECSFACT001      1.
********  RECORD NUMBER   483 REPLACES RECORD NUMBER    25   ********
  484  SECSFACT002      0.6
********  RECORD NUMBER   484 REPLACES RECORD NUMBER    26   ********
  485  SECSFACT003      0.5
********  RECORD NUMBER   485 REPLACES RECORD NUMBER    27   ********
      *
      * PROTIN - Inhalation protection factor
  486  SEPROTIN001      0.98
********  RECORD NUMBER   486 REPLACES RECORD NUMBER    28   ********
  487  SEPROTIN002      0.46
********  RECORD NUMBER   487 REPLACES RECORD NUMBER    29   ********
  488  SEPROTIN003      0.33
********  RECORD NUMBER   488 REPLACES RECORD NUMBER    30   ********
      *
      * BRRATE - Breathing rates
  489  SEBRRATE001      2.66E-04
********  RECORD NUMBER   489 REPLACES RECORD NUMBER    31   ********
  490  SEBRRATE002      2.66E-04
********  RECORD NUMBER   490 REPLACES RECORD NUMBER    32   ********
  491  SEBRRATE003      2.66E-04
********  RECORD NUMBER   491 REPLACES RECORD NUMBER    33   ********
      *
      * SKPFAC - skin protection factors
  492  SESKPFAC001      0.98
********  RECORD NUMBER   492 REPLACES RECORD NUMBER    34   ********
  493  SESKPFAC002      0.46
********  RECORD NUMBER   493 REPLACES RECORD NUMBER    35   ********
  494  SESKPFAC003      0.33
********  RECORD NUMBER   494 REPLACES RECORD NUMBER    36   ********
      *
      * GSHFAC - groundshine shielding factors
  495  SEGSHFAC001      0.5
********  RECORD NUMBER   495 REPLACES RECORD NUMBER    37   ********
  496  SEGSHFAC002      0.18
********  RECORD NUMBER   496 REPLACES RECORD NUMBER    38   ********
  497  SEGSHFAC003      0.1
********  RECORD NUMBER   497 REPLACES RECORD NUMBER    39   ********
      *
      * EANAM2 - Name of emergency response cohort
  498  EZEANAM2001      '10-20 Shadow'
********  RECORD NUMBER   498 REPLACES RECORD NUMBER    42   ********
      *
      * WTFRAC - weighting fraction applied to results of emergency response cohort
  499  EZWTFRAC001      0.
********  RECORD NUMBER   499 REPLACES RECORD NUMBER    44   ********
      *
      * TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
  500  TRAVELPOINT      CENTERPOINT
********  RECORD NUMBER   500 REPLACES RECORD NUMBER    46   ********
      *
      * ESPEED - evacuee travel speed during the three phases of evacuation
  501  EZESPEED001      8.941
********  RECORD NUMBER   501 REPLACES RECORD NUMBER    47   ********
  502  EZESPEED002      6.706
********  RECORD NUMBER   502 REPLACES RECORD NUMBER    48   ********
  503  EZESPEED003      8.941
********  RECORD NUMBER   503 REPLACES RECORD NUMBER    49   ********
      *
      * ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
  504  EZESPMUL001      0.7
********  RECORD NUMBER   504 REPLACES RECORD NUMBER    50   ********
  505  EZESPMUL002      0.7
********  RECORD NUMBER   505 REPLACES RECORD NUMBER    51   ********
  506  EZESPMUL003      0.7
********  RECORD NUMBER   506 REPLACES RECORD NUMBER    52   ********
      *
```

* REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
507 EZREFPNT001 ALARM
******** RECORD NUMBER 507 REPLACES RECORD NUMBER 53 ********
*
* DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
508 EZDURBEG001 900.
******** RECORD NUMBER 508 REPLACES RECORD NUMBER 54 ********
*
* DURMID - duration of middle phase of evacuation, in seconds.
509 EZDURMID001 21600.
******** RECORD NUMBER 509 REPLACES RECORD NUMBER 55 ********
*
* NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
510 EZNUMEVA001 18
******** RECORD NUMBER 510 REPLACES RECORD NUMBER 56 ********
*
* DLTSHL - delay from reference time point to when individual takes shelter. DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
511 EZDLTSHL001 7200.
******** RECORD NUMBER 511 REPLACES RECORD NUMBER 57 ********
512 EZDLTSHL002 7200.
******** RECORD NUMBER 512 REPLACES RECORD NUMBER 58 ********
513 EZDLTSHL003 7200.
******** RECORD NUMBER 513 REPLACES RECORD NUMBER 59 ********
514 EZDLTSHL004 7200.
******** RECORD NUMBER 514 REPLACES RECORD NUMBER 60 ********
515 EZDLTSHL005 7200.
******** RECORD NUMBER 515 REPLACES RECORD NUMBER 61 ********
516 EZDLTSHL006 7200.
******** RECORD NUMBER 516 REPLACES RECORD NUMBER 62 ********
517 EZDLTSHL007 7200.
******** RECORD NUMBER 517 REPLACES RECORD NUMBER 63 ********
518 EZDLTSHL008 7200.
******** RECORD NUMBER 518 REPLACES RECORD NUMBER 64 ********
519 EZDLTSHL009 7200.
******** RECORD NUMBER 519 REPLACES RECORD NUMBER 65 ********
520 EZDLTSHL010 7200.
******** RECORD NUMBER 520 REPLACES RECORD NUMBER 66 ********
521 EZDLTSHL011 7200.
******** RECORD NUMBER 521 REPLACES RECORD NUMBER 67 ********
522 EZDLTSHL012 7200.
******** RECORD NUMBER 522 REPLACES RECORD NUMBER 68 ********
523 EZDLTSHL013 7200.
******** RECORD NUMBER 523 REPLACES RECORD NUMBER 69 ********
524 EZDLTSHL014 7200.
******** RECORD NUMBER 524 REPLACES RECORD NUMBER 70 ********
525 EZDLTSHL015 7200.
******** RECORD NUMBER 525 REPLACES RECORD NUMBER 71 ********
526 EZDLTSHL016 7200.
******** RECORD NUMBER 526 REPLACES RECORD NUMBER 72 ********
527 EZDLTSHL017 7200.
******** RECORD NUMBER 527 REPLACES RECORD NUMBER 73 ********
528 EZDLTSHL018 7200.
******** RECORD NUMBER 528 REPLACES RECORD NUMBER 74 ********
*
* DLTEVA -Delay time to begin evacuation
529 EZDLTEVA001 7200.
******** RECORD NUMBER 529 REPLACES RECORD NUMBER 75 ********
530 EZDLTEVA002 7200.
******** RECORD NUMBER 530 REPLACES RECORD NUMBER 76 ********
531 EZDLTEVA003 7200.
******** RECORD NUMBER 531 REPLACES RECORD NUMBER 77 ********
532 EZDLTEVA004 7200.
******** RECORD NUMBER 532 REPLACES RECORD NUMBER 78 ********
533 EZDLTEVA005 7200.
******** RECORD NUMBER 533 REPLACES RECORD NUMBER 79 ********
534 EZDLTEVA006 7200.
******** RECORD NUMBER 534 REPLACES RECORD NUMBER 80 ********
535 EZDLTEVA007 7200.
******** RECORD NUMBER 535 REPLACES RECORD NUMBER 81 ********
536 EZDLTEVA008 7200.
******** RECORD NUMBER 536 REPLACES RECORD NUMBER 82 ********
537 EZDLTEVA009 7200.
******** RECORD NUMBER 537 REPLACES RECORD NUMBER 83 ********
538 EZDLTEVA010 7200.
******** RECORD NUMBER 538 REPLACES RECORD NUMBER 84 ********
539 EZDLTEVA011 7200.
******** RECORD NUMBER 539 REPLACES RECORD NUMBER 85 ********
540 EZDLTEVA012 7200.
******** RECORD NUMBER 540 REPLACES RECORD NUMBER 86 ********
541 EZDLTEVA013 7200.
******** RECORD NUMBER 541 REPLACES RECORD NUMBER 87 ********
542 EZDLTEVA014 7200.
******** RECORD NUMBER 542 REPLACES RECORD NUMBER 88 ********
543 EZDLTEVA015 7200.
******** RECORD NUMBER 543 REPLACES RECORD NUMBER 89 ********
544 EZDLTEVA016 7200.
******** RECORD NUMBER 544 REPLACES RECORD NUMBER 90 ********
545 EZDLTEVA017 7200.
******** RECORD NUMBER 545 REPLACES RECORD NUMBER 91 ********
546 EZDLTEVA018 7200.
******** RECORD NUMBER 546 REPLACES RECORD NUMBER 92 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
547 EZESPGRD001 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 547 REPLACES RECORD NUMBER 93 ********
548 EZESPGRD002 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 548 REPLACES RECORD NUMBER 94 ********
549 EZESPGRD003 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 549 REPLACES RECORD NUMBER 95 ********
550 EZESPGRD004 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 550 REPLACES RECORD NUMBER 96 ********
551 EZESPGRD005 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 551 REPLACES RECORD NUMBER 97 ********
552 EZESPGRD006 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75

******** RECORD NUMBER 552 REPLACES RECORD NUMBER 98 ********
553 EZESPGRD007 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 553 REPLACES RECORD NUMBER 99 ********
554 EZESPGRD008 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 554 REPLACES RECORD NUMBER 100 ********
555 EZESPGRD009 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 555 REPLACES RECORD NUMBER 101 ********
556 EZESPGRD010 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 556 REPLACES RECORD NUMBER 102 ********
557 EZESPGRD011 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 557 REPLACES RECORD NUMBER 103 ********
558 EZESPGRD012 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 558 REPLACES RECORD NUMBER 104 ********
559 EZESPGRD013 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 559 REPLACES RECORD NUMBER 105 ********
560 EZESPGRD014 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 560 REPLACES RECORD NUMBER 106 ********
561 EZESPGRD015 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 561 REPLACES RECORD NUMBER 107 ********
562 EZESPGRD016 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 562 REPLACES RECORD NUMBER 108 ********
563 EZESPGRD017 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 563 REPLACES RECORD NUMBER 109 ********
564 EZESPGRD018 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 564 REPLACES RECORD NUMBER 110 ********
565 EZESPGRD019 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 565 REPLACES RECORD NUMBER 111 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
566 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 566 REPLACES RECORD NUMBER 112 ********
567 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 567 REPLACES RECORD NUMBER 113 ********
568 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 4 1 1
******** RECORD NUMBER 568 REPLACES RECORD NUMBER 114 ********
569 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2 1
4 4 4 4 4 1 1
******** RECORD NUMBER 569 REPLACES RECORD NUMBER 115 ********
570 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 1
1 1 1 1 1 1 1
******** RECORD NUMBER 570 REPLACES RECORD NUMBER 116 ********
571 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4 4
4 4 1 1 1 1 1
******** RECORD NUMBER 571 REPLACES RECORD NUMBER 117 ********
572 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 4 4 1 4 4 4 4 4
4 1 1 1 1 1 1
******** RECORD NUMBER 572 REPLACES RECORD NUMBER 118 ********
573 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 4 1 4 4 4 2 2
1 1 1 1 1 2 2
******** RECORD NUMBER 573 REPLACES RECORD NUMBER 119 ********
574 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 4 4 1 4 2 1 4 1
4 1 1 1 2 2 2
******** RECORD NUMBER 574 REPLACES RECORD NUMBER 120 ********
575 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 4 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1 4
4 4 2 2 1 1 1
******** RECORD NUMBER 575 REPLACES RECORD NUMBER 121 ********
576 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 4 4 2 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4 4
4 2 2 1 4 2 2
******** RECORD NUMBER 576 REPLACES RECORD NUMBER 122 ********
577 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 2 1 2 1 2 2 1 4 4 4 4 2 2 2 1 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4 2
1 1 2 2 1 4 2
******** RECORD NUMBER 577 REPLACES RECORD NUMBER 123 ********
578 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 4 4 4 1 2 1 4 4 4 4 2 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4 2
1 4 2 1 4 1 1
******** RECORD NUMBER 578 REPLACES RECORD NUMBER 124 ********
579 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4 1
4 1 1 1 1 1 1
******** RECORD NUMBER 579 REPLACES RECORD NUMBER 125 ********
580 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2 1
1 2 2 2 1 2 1
******** RECORD NUMBER 580 REPLACES RECORD NUMBER 126 ********
581 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 581 REPLACES RECORD NUMBER 127 ********
582 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 582 REPLACES RECORD NUMBER 128 ********
583 EZIDIREC018 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 583 REPLACES RECORD NUMBER 129 ********
584 EZIDIREC019 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 584 REPLACES RECORD NUMBER 130 ********
*
* LASMOV - The outermost spatial interval of the evacuation movement zone.
585 EZLASMOV001 19
******** RECORD NUMBER 585 REPLACES RECORD NUMBER 131 ********
*
* EFFACY, KI Ingestion
586 EZEFFACY001 0.7
******** RECORD NUMBER 586 REPLACES RECORD NUMBER 269 ********
*
* POPFRAC, KI Ingestion
587 EZPOPFRC001 0.

******** RECORD NUMBER 587 REPLACES RECORD NUMBER 270 ********
.
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE 3 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE 3
NUMBER OF RECORDS CHANGED = 105
NUMBER OF RECORDS ADDED = 0
****

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	4	4	2	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	2	1	2	1	2	2	1	4	4	4
13	1	1	1	1	1	2	1	4	4	4	1	2	1	4	4	4
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1
4	4	2	2	1	1	1	1	4	4	4	2	2	1	4	4	2
5	1	1	1	1	2	1	4	4	2	2	1	4	4	2	2	1
6	2	2	1	1	1	1	2	1	1	4	2	2	1	2	1	4
7	1	1	1	1	4	4	2	2	1	1	4	4	1	1	4	1
8	2	1	4	4	2	1	2	1	2	1	4	2	1	4	2	1
9	1	2	1	4	2	1	2	1	2	1	1	4	1	4	1	4
10	2	2	2	2	1	2	1	4	4	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	4	2	2	2	1	2	1	1	1	1	4	1	1	1	1	4
13	4	2	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1	1
4	1	4	4	2	1	1	4	2	1	4	4	4	4	4	1	1
5	4	4	2	2	1	4	4	2	1	1	1	1	1	1	1	1
6	4	2	2	1	4	4	4	4	4	4	4	1	1	1	1	1
7	4	4	4	1	4	4	4	4	4	4	1	1	1	1	1	1
8	1	4	4	1	4	4	4	2	2	1	1	1	1	1	2	2
9	4	4	4	1	4	2	1	4	1	4	1	1	1	2	2	2
10	4	2	2	1	1	1	1	1	4	4	4	2	2	1	1	1
11	4	4	4	2	2	2	1	4	4	4	2	2	1	4	2	2
12	4	4	2	2	2	2	1	4	2	1	1	2	2	1	4	2
13	4	4	2	2	2	1	1	4	2	1	4	2	1	4	1	1
14	1	4	4	2	2	1	1	4	1	4	1	1	1	1	1	1
15	1	4	2	1	2	1	1	2	1	1	2	2	2	1	2	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

THE KI MODEL IS IN EFFECT

******** BEGINNING OF CHANGE CASE 4 USER INPUT ********
*
* CSFACT - Cloudshine shielding factor
588 SECSFACT001 1.
******** RECORD NUMBER 588 REPLACES RECORD NUMBER 25 ********
589 SECSFACT002 0.31
******** RECORD NUMBER 589 REPLACES RECORD NUMBER 26 ********
590 SECSFACT003 0.31

******** RECORD NUMBER 590 REPLACES RECORD NUMBER 27 ********
*
* PROTIN - Inhalation protection factor
591 SEPROTIN001 0.98
******** RECORD NUMBER 591 REPLACES RECORD NUMBER 28 ********
592 SEPROTIN002 0.33
******** RECORD NUMBER 592 REPLACES RECORD NUMBER 29 ********
593 SEPROTIN003 0.33
******** RECORD NUMBER 593 REPLACES RECORD NUMBER 30 ********
*
* BRRATE - Breathing rates
594 SEBRRATE001 2.66E-04
******** RECORD NUMBER 594 REPLACES RECORD NUMBER 31 ********
595 SEBRRATE002 2.66E-04
******** RECORD NUMBER 595 REPLACES RECORD NUMBER 32 ********
596 SEBRRATE003 2.66E-04
******** RECORD NUMBER 596 REPLACES RECORD NUMBER 33 ********
*
* SKPFAC - skin protection factors
597 SESKPFAC001 0.98
******** RECORD NUMBER 597 REPLACES RECORD NUMBER 34 ********
598 SESKPFAC002 0.33
******** RECORD NUMBER 598 REPLACES RECORD NUMBER 35 ********
599 SESKPFAC003 0.33
******** RECORD NUMBER 599 REPLACES RECORD NUMBER 36 ********
*
* GSHFAC - groundshine shielding factors
600 SEGSHFAC001 0.5
******** RECORD NUMBER 600 REPLACES RECORD NUMBER 37 ********
601 SEGSHFAC002 0.05
******** RECORD NUMBER 601 REPLACES RECORD NUMBER 38 ********
602 SEGSHFAC003 0.05
******** RECORD NUMBER 602 REPLACES RECORD NUMBER 39 ********
*
* EANAM2 - Name of emergency response cohort
603 EZEANAM2001 '0-10 Special Facilities'
******** RECORD NUMBER 603 REPLACES RECORD NUMBER 42 ********
*
* WTFRAC - weighting fraction applied to results of emergency response cohort
604 EZWTFRAC001 0.006
******** RECORD NUMBER 604 REPLACES RECORD NUMBER 44 ********
*
* TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
605 TRAVELPOINT CENTERPOINT
******** RECORD NUMBER 605 REPLACES RECORD NUMBER 46 ********
*
* ESPEED - evacuee travel speed during the three phases of evacuation
606 EZESPEED001 0.894
******** RECORD NUMBER 606 REPLACES RECORD NUMBER 47 ********
607 EZESPEED002 2.235
******** RECORD NUMBER 607 REPLACES RECORD NUMBER 48 ********
608 EZESPEED003 8.941
******** RECORD NUMBER 608 REPLACES RECORD NUMBER 49 ********
*
* ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
609 EZESPMUL001 0.7
******** RECORD NUMBER 609 REPLACES RECORD NUMBER 50 ********
610 EZESPMUL002 0.7
******** RECORD NUMBER 610 REPLACES RECORD NUMBER 51 ********
611 EZESPMUL003 0.7
******** RECORD NUMBER 611 REPLACES RECORD NUMBER 52 ********
*
* REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
612 EZREFPNT001 ALARM
******** RECORD NUMBER 612 REPLACES RECORD NUMBER 53 ********
*
* DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
613 EZDURBEG001 10800.
******** RECORD NUMBER 613 REPLACES RECORD NUMBER 54 ********
*
* DURMID - duration of middle phase of evacuation, in seconds.
614 EZDURMID001 7200.
******** RECORD NUMBER 614 REPLACES RECORD NUMBER 55 ********
*
* NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
615 EZNUMEVA001 18
******** RECORD NUMBER 615 REPLACES RECORD NUMBER 56 ********
*
* DLTSHL - delay from reference time point to when individual takes shelter. DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
616 EZDLTSHL001 0.
******** RECORD NUMBER 616 REPLACES RECORD NUMBER 57 ********
617 EZDLTSHL002 0.
******** RECORD NUMBER 617 REPLACES RECORD NUMBER 58 ********
618 EZDLTSHL003 0.
******** RECORD NUMBER 618 REPLACES RECORD NUMBER 59 ********
619 EZDLTSHL004 0.
******** RECORD NUMBER 619 REPLACES RECORD NUMBER 60 ********
620 EZDLTSHL005 0.
******** RECORD NUMBER 620 REPLACES RECORD NUMBER 61 ********
621 EZDLTSHL006 0.
******** RECORD NUMBER 621 REPLACES RECORD NUMBER 62 ********
622 EZDLTSHL007 0.
******** RECORD NUMBER 622 REPLACES RECORD NUMBER 63 ********
623 EZDLTSHL008 0.
******** RECORD NUMBER 623 REPLACES RECORD NUMBER 64 ********
624 EZDLTSHL009 0.
******** RECORD NUMBER 624 REPLACES RECORD NUMBER 65 ********
625 EZDLTSHL010 0.
******** RECORD NUMBER 625 REPLACES RECORD NUMBER 66 ********
626 EZDLTSHL011 0.
******** RECORD NUMBER 626 REPLACES RECORD NUMBER 67 ********
627 EZDLTSHL012 0.
******** RECORD NUMBER 627 REPLACES RECORD NUMBER 68 ********
628 EZDLTSHL013 0.
******** RECORD NUMBER 628 REPLACES RECORD NUMBER 69 ********
629 EZDLTSHL014 0.
******** RECORD NUMBER 629 REPLACES RECORD NUMBER 70 ********
630 EZDLTSHL015 0.
******** RECORD NUMBER 630 REPLACES RECORD NUMBER 71 ********
631 EZDLTSHL016 0.

******** RECORD NUMBER 631 REPLACES RECORD NUMBER 72 ********
632 EZDLTSHL017 0.
******** RECORD NUMBER 632 REPLACES RECORD NUMBER 73 ********
633 EZDLTSHL018 0.
******** RECORD NUMBER 633 REPLACES RECORD NUMBER 74 ********
*
* DLTEVA -Delay time to begin evacuation
634 EZDLTEVA001 18000.
******** RECORD NUMBER 634 REPLACES RECORD NUMBER 75 ********
635 EZDLTEVA002 18000.
******** RECORD NUMBER 635 REPLACES RECORD NUMBER 76 ********
636 EZDLTEVA003 18000.
******** RECORD NUMBER 636 REPLACES RECORD NUMBER 77 ********
637 EZDLTEVA004 18000.
******** RECORD NUMBER 637 REPLACES RECORD NUMBER 78 ********
638 EZDLTEVA005 18000.
******** RECORD NUMBER 638 REPLACES RECORD NUMBER 79 ********
639 EZDLTEVA006 18000.
******** RECORD NUMBER 639 REPLACES RECORD NUMBER 80 ********
640 EZDLTEVA007 18000.
******** RECORD NUMBER 640 REPLACES RECORD NUMBER 81 ********
641 EZDLTEVA008 18000.
******** RECORD NUMBER 641 REPLACES RECORD NUMBER 82 ********
642 EZDLTEVA009 18000.
******** RECORD NUMBER 642 REPLACES RECORD NUMBER 83 ********
643 EZDLTEVA010 18000.
******** RECORD NUMBER 643 REPLACES RECORD NUMBER 84 ********
644 EZDLTEVA011 18000.
******** RECORD NUMBER 644 REPLACES RECORD NUMBER 85 ********
645 EZDLTEVA012 18000.
******** RECORD NUMBER 645 REPLACES RECORD NUMBER 86 ********
646 EZDLTEVA013 18000.
******** RECORD NUMBER 646 REPLACES RECORD NUMBER 87 ********
647 EZDLTEVA014 18000.
******** RECORD NUMBER 647 REPLACES RECORD NUMBER 88 ********
648 EZDLTEVA015 18000.
******** RECORD NUMBER 648 REPLACES RECORD NUMBER 89 ********
649 EZDLTEVA016 18000.
******** RECORD NUMBER 649 REPLACES RECORD NUMBER 90 ********
650 EZDLTEVA017 18000.
******** RECORD NUMBER 650 REPLACES RECORD NUMBER 91 ********
651 EZDLTEVA018 18000.
******** RECORD NUMBER 651 REPLACES RECORD NUMBER 92 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
652 EZESPGRD001 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 652 REPLACES RECORD NUMBER 93 ********
653 EZESPGRD002 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 653 REPLACES RECORD NUMBER 94 ********
654 EZESPGRD003 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 654 REPLACES RECORD NUMBER 95 ********
655 EZESPGRD004 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 655 REPLACES RECORD NUMBER 96 ********
656 EZESPGRD005 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 656 REPLACES RECORD NUMBER 97 ********
657 EZESPGRD006 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 657 REPLACES RECORD NUMBER 98 ********
658 EZESPGRD007 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 658 REPLACES RECORD NUMBER 99 ********
659 EZESPGRD008 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 659 REPLACES RECORD NUMBER 100 ********
660 EZESPGRD009 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 660 REPLACES RECORD NUMBER 101 ********
661 EZESPGRD010 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 661 REPLACES RECORD NUMBER 102 ********
662 EZESPGRD011 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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******** RECORD NUMBER 662 REPLACES RECORD NUMBER 103 ********
663 EZESPGRD012 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
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******** RECORD NUMBER 663 REPLACES RECORD NUMBER 104 ********
664 EZESPGRD013 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 664 REPLACES RECORD NUMBER 105 ********
665 EZESPGRD014 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 665 REPLACES RECORD NUMBER 106 ********
666 EZESPGRD015 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 666 REPLACES RECORD NUMBER 107 ********
667 EZESPGRD016 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 667 REPLACES RECORD NUMBER 108 ********
668 EZESPGRD017 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 668 REPLACES RECORD NUMBER 109 ********
669 EZESPGRD018 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 669 REPLACES RECORD NUMBER 110 ********
670 EZESPGRD019 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 670 REPLACES RECORD NUMBER 111 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
671 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 671 REPLACES RECORD NUMBER 112 ********
672 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 672 REPLACES RECORD NUMBER 113 ********

```
 673  EZIDIREC003    1    1    1    1    1    1    1    1    1    1    1    1    1    2    1    1    1    1    1    1    1    1    1    1    2    1    4    4    4    4    2    2    2    2    2    1    1    1    4    4    4    4    2    2    2    1    1    1    1    1    1    1    1    1    1    1    1
1    1    1    1    4    1    1
********  RECORD NUMBER  673 REPLACES RECORD NUMBER  114  ********
 674  EZIDIREC004    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    2    2    1    1    1    1    4    4    4    2    2    1    4    4    2    1    4    4    2    1    1    4    2    1
4    4    4    4    4    1    1
********  RECORD NUMBER  674 REPLACES RECORD NUMBER  115  ********
 675  EZIDIREC005    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    1    1    1    1    1    1    1    1    1    2    2    2    2    2    2    1    1    1    1    1    2    1    4    4    2    2    1    4    4    2    2    1    4    4    2    2    1    4    4    2    1
1    1    1    1    1    1    1
********  RECORD NUMBER  675 REPLACES RECORD NUMBER  116  ********
 676  EZIDIREC006    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    1    1    1    1    1    1    1    1    1    1    1    2    2    2    2    2    2    2    2    1    1    1    1    2    1    1    4    2    2    1    2    1    4    4    2    2    1    4    4    4    4    4
4    4    1    1    1    1    1
********  RECORD NUMBER  676 REPLACES RECORD NUMBER  117  ********
 677  EZIDIREC007    2    2    1    2    2    1    2    2    1    4    2    1    4    2    2    1    1    1    1    1    1    1    1    1    1    1    2    2    1    4    4    2    1    1    1    1    4    4    2    2    1    1    4    4    1    1    4    1    4    4    4    1    4    4    4    4    4
4    1    1    1    1    1    1
********  RECORD NUMBER  677 REPLACES RECORD NUMBER  118  ********
 678  EZIDIREC008    1    4    1    1    4    2    1    4    2    1    1    4    2    2    1    1    2    1    1    1    4    4    4    1    1    1    1    1    1    4    1    4    2    1    4    4    2    1    2    1    2    1    4    2    1    4    2    1    1    4    4    1    4    4    4    2    2
1    1    1    1    1    2    2
********  RECORD NUMBER  678 REPLACES RECORD NUMBER  119  ********
 679  EZIDIREC009    1    1    4    2    1    1    2    1    1    4    1    4    1    4    1    1    1    2    1    1    1    4    4    4    1    1    4    1    2    2    2    1    1    2    1    4    2    1    2    1    2    1    1    4    1    4    1    4    4    4    4    1    4    2    1    4    1
4    1    1    1    2    2    2
********  RECORD NUMBER  679 REPLACES RECORD NUMBER  120  ********
 680  EZIDIREC010    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    4    4    2    2    2    2    1    2    1    4    4    1    1    1    4    2    1    4    4    2    2    1    1    1    1    1    4
4    4    2    2    1    1    1
********  RECORD NUMBER  680 REPLACES RECORD NUMBER  121  ********
 681  EZIDIREC011    1    1    4    2    1    4    2    1    4    4    4    4    2    2    1    4    4    2    2    2    2    2    2    1    4    4    1    1    4    2    2    1    1    1    4    2    1    4    1    4    2    1    4    1    1    2    1    1    4    4    4    2    2    2    1    4    4
4    2    2    1    4    2    2
********  RECORD NUMBER  681 REPLACES RECORD NUMBER  122  ********
 682  EZIDIREC012    2    1    1    4    1    1    4    1    4    4    2    1    4    2    1    1    4    4    2    2    2    1    2    1    2    1    2    2    1    4    4    4    4    2    2    2    1    2    1    1    1    1    4    1    1    1    1    4    4    4    2    2    2    2    1    4    2
1    1    2    2    1    4    2
********  RECORD NUMBER  682 REPLACES RECORD NUMBER  123  ********
 683  EZIDIREC013    1    1    4    1    4    2    1    1    2    1    4    4    2    2    1    1    1    1    1    1    1    2    1    4    4    4    1    2    1    4    4    4    4    2    2    1    4    4    1    1    4    2    1    1    2    1    2    1    4    4    2    2    2    1    1    4    2
1    4    2    1    4    1    1
********  RECORD NUMBER  683 REPLACES RECORD NUMBER  124  ********
 684  EZIDIREC014    1    1    4    1    1    1    2    1    2    1    2    1    1    2    1    1    1    1    1    1    1    4    1    1    1    1    2    1    4    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    2    1    1    4    4    2    2    1    1    4    1
4    1    1    1    1    1    1
********  RECORD NUMBER  684 REPLACES RECORD NUMBER  125  ********
 685  EZIDIREC015    1    1    4    2    2    2    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    4    4    2    2    2    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    1    2    1    1    1    4    2    1    2    1    1    2    1
1    2    2    2    1    2    1
********  RECORD NUMBER  685 REPLACES RECORD NUMBER  126  ********
 686  EZIDIREC016    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    4    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1
1    1    1    1    1    1    1
********  RECORD NUMBER  686 REPLACES RECORD NUMBER  127  ********
 687  EZIDIREC017    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1
1    1    1    1    1    1    1
********  RECORD NUMBER  687 REPLACES RECORD NUMBER  128  ********
 688  EZIDIREC018    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1
1    1    1    1    1    1    1
********  RECORD NUMBER  688 REPLACES RECORD NUMBER  129  ********
 689  EZIDIREC019    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1    1
1    1    1    1    1    1    1
********  RECORD NUMBER  689 REPLACES RECORD NUMBER  130  ********
     *
     * LASMOV - The outermost spatial interval of the evacuation movement zone.
 690  EZLASMOV001     19
********  RECORD NUMBER  690 REPLACES RECORD NUMBER  131  ********
     *
     * EFFACY,  KI Ingestion
 691  EZEFFACY001     0.7
********  RECORD NUMBER  691 REPLACES RECORD NUMBER  269  ********
     *
     * POPFRAC, KI Ingestion
 692  EZPOPFRC001     0.
********  RECORD NUMBER  692 REPLACES RECORD NUMBER  270  ********
     .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE   4 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE   4
NUMBER OF RECORDS CHANGED          =  105
NUMBER OF RECORDS ADDED            =    0
****************************************************************************************************
```

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	4	4	2	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4

```
 11    4  2  2  2  2  2  2  1  4  4  1  1  4  2  2  1
 12    4  4  2  2  2  1  2  1  2  1  2  2  1  4  4  4
 13    1  1  1  1  1  2  1  4  4  4  1  2  1  4  4  4
 14    1  1  1  1  1  4  1  1  1  1  2  1  4  1  1  1
 15    1  1  1  1  1  1  1  4  4  4  2  2  2  1  1  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD    33  34  35  36  37  38  39  40  41  42  43  44  45  46  47  48
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    2  2  2  1  1  1  4  4  4  4  2  2  2  1  1  1
  4    4  2  2  1  1  1  1  4  4  4  2  2  1  4  4  2
  5    1  1  1  1  2  1  4  4  2  2  1  4  4  2  2  1
  6    2  2  1  1  1  1  2  1  1  4  2  2  1  2  1  4
  7    1  1  1  1  4  4  2  2  1  1  4  4  1  1  4  1
  8    2  1  4  4  2  1  2  1  2  1  4  2  1  4  2  1
  9    1  2  1  4  2  1  2  1  2  1  1  4  1  4  1  4
 10    2  2  2  2  1  2  1  4  4  1  1  1  4  2  1  4
 11    1  1  4  2  1  4  1  4  2  1  4  1  1  2  1  1
 12    4  2  2  2  1  2  1  1  1  1  4  1  1  1  1  4
 13    4  2  2  1  4  4  1  1  4  2  1  1  2  1  2  1
 14    1  1  1  1  1  1  1  1  1  1  1  1  1  1  2  1
 15    1  1  1  1  1  1  1  1  1  1  1  4  1  2  1  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD    49  50  51  52  53  54  55  56  57  58  59  60  61  62  63  64
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  1
  4    1  4  4  2  1  1  4  2  1  4  4  4  4  4  1  1
  5    4  4  2  2  1  4  4  2  1  1  1  1  1  1  1  1
  6    4  2  2  1  4  4  4  4  4  4  4  1  1  1  1  1
  7    4  4  4  1  4  4  4  4  4  4  1  1  1  1  1  1
  8    1  4  4  1  4  4  4  2  2  1  1  1  1  1  2  2
  9    4  4  4  1  4  2  1  4  1  4  1  1  1  2  2  2
 10    4  2  2  1  1  1  1  1  4  4  4  2  2  1  1  1
 11    4  4  4  2  2  2  1  4  4  4  2  2  1  4  2  2
 12    4  4  2  2  2  2  1  4  2  1  1  2  2  1  4  2
 13    4  4  2  2  2  1  1  4  2  1  4  2  1  4  1  1
 14    1  4  4  2  2  1  1  4  1  4  1  1  1  1  1  1
 15    1  4  2  1  2  1  1  2  1  1  2  2  2  1  2  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE   5 USER INPUT ********
      *
      * CSFACT - Cloudshine shielding factor
  693  SECSFACT001      1.
********   RECORD NUMBER   693 REPLACES RECORD NUMBER    25   ********
  694  SECSFACT002      0.6
********   RECORD NUMBER   694 REPLACES RECORD NUMBER    26   ********
  695  SECSFACT003      0.5
********   RECORD NUMBER   695 REPLACES RECORD NUMBER    27   ********
      *
      * PROTIN - Inhalation protection factor
  696  SEPROTIN001      0.98
********   RECORD NUMBER   696 REPLACES RECORD NUMBER    28   ********
  697  SEPROTIN002      0.46
********   RECORD NUMBER   697 REPLACES RECORD NUMBER    29   ********
  698  SEPROTIN003      0.33
********   RECORD NUMBER   698 REPLACES RECORD NUMBER    30   ********
      *
      * BRRATE - Breathing rates
  699  SEBRRATE001      2.66E-04
********   RECORD NUMBER   699 REPLACES RECORD NUMBER    31   ********
  700  SEBRRATE002      2.66E-04
********   RECORD NUMBER   700 REPLACES RECORD NUMBER    32   ********
  701  SEBRRATE003      2.66E-04
********   RECORD NUMBER   701 REPLACES RECORD NUMBER    33   ********
      *
      * SKPFAC - skin protection factors
  702  SESKPFAC001      0.98
********   RECORD NUMBER   702 REPLACES RECORD NUMBER    34   ********
  703  SESKPFAC002      0.46
********   RECORD NUMBER   703 REPLACES RECORD NUMBER    35   ********
  704  SESKPFAC003      0.33
********   RECORD NUMBER   704 REPLACES RECORD NUMBER    36   ********
      *
      * GSHFAC - groundshine shielding factors
  705  SEGSHFAC001      0.5
********   RECORD NUMBER   705 REPLACES RECORD NUMBER    37   ********
  706  SEGSHFAC002      0.18
********   RECORD NUMBER   706 REPLACES RECORD NUMBER    38   ********
  707  SEGSHFAC003      0.1
********   RECORD NUMBER   707 REPLACES RECORD NUMBER    39   ********
      *
      * EANAM2 - Name of emergency response cohort
  708  EZEANAM2001      '0-10 Evacuation Tail'
********   RECORD NUMBER   708 REPLACES RECORD NUMBER    42   ********
      *
      * WTFRAC - weighting fraction applied to results of emergency response cohort
  709  EZWTFRAC001      0.1
********   RECORD NUMBER   709 REPLACES RECORD NUMBER    44   ********
      *
      * TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
  710  TRAVELPOINT      CENTERPOINT
********   RECORD NUMBER   710 REPLACES RECORD NUMBER    46   ********
      *
      * ESPEED - evacuee travel speed during the three phases of evacuation
  711  EZESPEED001      0.894
```

******** RECORD NUMBER 711 REPLACES RECORD NUMBER 47 ********
712 EZESPEED002 2.235
******** RECORD NUMBER 712 REPLACES RECORD NUMBER 48 ********
713 EZESPEED003 8.941
******** RECORD NUMBER 713 REPLACES RECORD NUMBER 49 ********
*
* ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
714 EZESPMUL001 0.7
******** RECORD NUMBER 714 REPLACES RECORD NUMBER 50 ********
715 EZESPMUL002 0.7
******** RECORD NUMBER 715 REPLACES RECORD NUMBER 51 ********
716 EZESPMUL003 0.7
******** RECORD NUMBER 716 REPLACES RECORD NUMBER 52 ********
*
* REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
717 EZREFPNT001 ALARM
******** RECORD NUMBER 717 REPLACES RECORD NUMBER 53 ********
*
* DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
718 EZDURBEG001 7200.
******** RECORD NUMBER 718 REPLACES RECORD NUMBER 54 ********
*
* DURMID - duration of middle phase of evacuation, in seconds.
719 EZDURMID001 7200.
******** RECORD NUMBER 719 REPLACES RECORD NUMBER 55 ********
*
* NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
720 EZNUMEVA001 18
******** RECORD NUMBER 720 REPLACES RECORD NUMBER 56 ********
*
* DLTSHL - delay from reference time point to when individual takes shelter. DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
721 EZDLTSHL001 10800.
******** RECORD NUMBER 721 REPLACES RECORD NUMBER 57 ********
722 EZDLTSHL002 10800.
******** RECORD NUMBER 722 REPLACES RECORD NUMBER 58 ********
723 EZDLTSHL003 10800.
******** RECORD NUMBER 723 REPLACES RECORD NUMBER 59 ********
724 EZDLTSHL004 10800.
******** RECORD NUMBER 724 REPLACES RECORD NUMBER 60 ********
725 EZDLTSHL005 10800.
******** RECORD NUMBER 725 REPLACES RECORD NUMBER 61 ********
726 EZDLTSHL006 10800.
******** RECORD NUMBER 726 REPLACES RECORD NUMBER 62 ********
727 EZDLTSHL007 10800.
******** RECORD NUMBER 727 REPLACES RECORD NUMBER 63 ********
728 EZDLTSHL008 10800.
******** RECORD NUMBER 728 REPLACES RECORD NUMBER 64 ********
729 EZDLTSHL009 10800.
******** RECORD NUMBER 729 REPLACES RECORD NUMBER 65 ********
730 EZDLTSHL010 10800.
******** RECORD NUMBER 730 REPLACES RECORD NUMBER 66 ********
731 EZDLTSHL011 10800.
******** RECORD NUMBER 731 REPLACES RECORD NUMBER 67 ********
732 EZDLTSHL012 10800.
******** RECORD NUMBER 732 REPLACES RECORD NUMBER 68 ********
733 EZDLTSHL013 10800.
******** RECORD NUMBER 733 REPLACES RECORD NUMBER 69 ********
734 EZDLTSHL014 10800.
******** RECORD NUMBER 734 REPLACES RECORD NUMBER 70 ********
735 EZDLTSHL015 10800.
******** RECORD NUMBER 735 REPLACES RECORD NUMBER 71 ********
736 EZDLTSHL016 10800.
******** RECORD NUMBER 736 REPLACES RECORD NUMBER 72 ********
737 EZDLTSHL017 10800.
******** RECORD NUMBER 737 REPLACES RECORD NUMBER 73 ********
738 EZDLTSHL018 10800.
******** RECORD NUMBER 738 REPLACES RECORD NUMBER 74 ********
*
* DLTEVA -Delay time to begin evacuation
739 EZDLTEVA001 10800.
******** RECORD NUMBER 739 REPLACES RECORD NUMBER 75 ********
740 EZDLTEVA002 10800.
******** RECORD NUMBER 740 REPLACES RECORD NUMBER 76 ********
741 EZDLTEVA003 10800.
******** RECORD NUMBER 741 REPLACES RECORD NUMBER 77 ********
742 EZDLTEVA004 10800.
******** RECORD NUMBER 742 REPLACES RECORD NUMBER 78 ********
743 EZDLTEVA005 10800.
******** RECORD NUMBER 743 REPLACES RECORD NUMBER 79 ********
744 EZDLTEVA006 10800.
******** RECORD NUMBER 744 REPLACES RECORD NUMBER 80 ********
745 EZDLTEVA007 10800.
******** RECORD NUMBER 745 REPLACES RECORD NUMBER 81 ********
746 EZDLTEVA008 10800.
******** RECORD NUMBER 746 REPLACES RECORD NUMBER 82 ********
747 EZDLTEVA009 10800.
******** RECORD NUMBER 747 REPLACES RECORD NUMBER 83 ********
748 EZDLTEVA010 10800.
******** RECORD NUMBER 748 REPLACES RECORD NUMBER 84 ********
749 EZDLTEVA011 10800.
******** RECORD NUMBER 749 REPLACES RECORD NUMBER 85 ********
750 EZDLTEVA012 10800.
******** RECORD NUMBER 750 REPLACES RECORD NUMBER 86 ********
751 EZDLTEVA013 10800.
******** RECORD NUMBER 751 REPLACES RECORD NUMBER 87 ********
752 EZDLTEVA014 10800.
******** RECORD NUMBER 752 REPLACES RECORD NUMBER 88 ********
753 EZDLTEVA015 10800.
******** RECORD NUMBER 753 REPLACES RECORD NUMBER 89 ********
754 EZDLTEVA016 10800.
******** RECORD NUMBER 754 REPLACES RECORD NUMBER 90 ********
755 EZDLTEVA017 10800.
******** RECORD NUMBER 755 REPLACES RECORD NUMBER 91 ********
756 EZDLTEVA018 10800.
******** RECORD NUMBER 756 REPLACES RECORD NUMBER 92 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
757 EZESPGRD001 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 757 REPLACES RECORD NUMBER 93 ********

758 EZESPGRD002 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 758 REPLACES RECORD NUMBER 94 ********
759 EZESPGRD003 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 759 REPLACES RECORD NUMBER 95 ********
760 EZESPGRD004 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 760 REPLACES RECORD NUMBER 96 ********
761 EZESPGRD005 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 761 REPLACES RECORD NUMBER 97 ********
762 EZESPGRD006 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 762 REPLACES RECORD NUMBER 98 ********
763 EZESPGRD007 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 763 REPLACES RECORD NUMBER 99 ********
764 EZESPGRD008 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 764 REPLACES RECORD NUMBER 100 ********
765 EZESPGRD009 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 765 REPLACES RECORD NUMBER 101 ********
766 EZESPGRD010 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 766 REPLACES RECORD NUMBER 102 ********
767 EZESPGRD011 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 767 REPLACES RECORD NUMBER 103 ********
768 EZESPGRD012 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 768 REPLACES RECORD NUMBER 104 ********
769 EZESPGRD013 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 769 REPLACES RECORD NUMBER 105 ********
770 EZESPGRD014 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 770 REPLACES RECORD NUMBER 106 ********
771 EZESPGRD015 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 771 REPLACES RECORD NUMBER 107 ********
772 EZESPGRD016 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 772 REPLACES RECORD NUMBER 108 ********
773 EZESPGRD017 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 773 REPLACES RECORD NUMBER 109 ********
774 EZESPGRD018 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 774 REPLACES RECORD NUMBER 110 ********
775 EZESPGRD019 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 775 REPLACES RECORD NUMBER 111 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
776 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 776 REPLACES RECORD NUMBER 112 ********
777 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 777 REPLACES RECORD NUMBER 113 ********
778 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 4 1 1
******** RECORD NUMBER 778 REPLACES RECORD NUMBER 114 ********
779 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2 1
4 4 4 4 4 1 1
******** RECORD NUMBER 779 REPLACES RECORD NUMBER 115 ********
780 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 1
1 1 1 1 1 1 1
******** RECORD NUMBER 780 REPLACES RECORD NUMBER 116 ********
781 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4 4
4 4 1 1 1 1 1
******** RECORD NUMBER 781 REPLACES RECORD NUMBER 117 ********
782 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 4 4 1 4 4 4 4 4
4 1 1 1 1 1 1
******** RECORD NUMBER 782 REPLACES RECORD NUMBER 118 ********
783 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 4 1 4 4 4 2 2
1 1 1 1 1 2 2
******** RECORD NUMBER 783 REPLACES RECORD NUMBER 119 ********
784 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 4 4 1 4 2 1 4 1
4 1 1 1 2 2 2
******** RECORD NUMBER 784 REPLACES RECORD NUMBER 120 ********
785 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 4 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1 4
4 4 2 2 1 1 1
******** RECORD NUMBER 785 REPLACES RECORD NUMBER 121 ********
786 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 4 4 2 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4 4
4 2 2 1 4 2 2
******** RECORD NUMBER 786 REPLACES RECORD NUMBER 122 ********
787 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 2 1 2 1 2 2 1 4 4 4 4 2 2 2 1 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4 2
1 1 2 2 1 4 2
******** RECORD NUMBER 787 REPLACES RECORD NUMBER 123 ********
788 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 4 4 4 1 2 1 4 4 4 4 2 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4 2
1 4 2 1 4 1 1
******** RECORD NUMBER 788 REPLACES RECORD NUMBER 124 ********
789 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4 1
4 1 1 1 1 1 1
******** RECORD NUMBER 789 REPLACES RECORD NUMBER 125 ********
790 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2 1
1 2 2 2 1 2 1
******** RECORD NUMBER 790 REPLACES RECORD NUMBER 126 ********
791 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 791 REPLACES RECORD NUMBER 127 ********
792 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 792 REPLACES RECORD NUMBER 128 ********
793 EZIDIREC018 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 793 REPLACES RECORD NUMBER 129 ********

794 EZIDIREC019 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 794 REPLACES RECORD NUMBER 130 ********
*
* LASMOV - The outermost spatial interval of the evacuation movement zone.
795 EZLASMOV001 19
******** RECORD NUMBER 795 REPLACES RECORD NUMBER 131 ********
*
* EFFACY, KI Ingestion
796 EZEFFACY001 0.7
******** RECORD NUMBER 796 REPLACES RECORD NUMBER 269 ********
*
* POPFRAC, KI Ingestion
797 EZPOPFRC001 0.
******** RECORD NUMBER 797 REPLACES RECORD NUMBER 270 ********
.
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE 5 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE 5
NUMBER OF RECORDS CHANGED = 105
NUMBER OF RECORDS ADDED = 0
****

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	4	4	2	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	2	1	2	1	2	2	1	4	4	4
13	1	1	1	1	1	2	1	4	4	4	1	2	1	4	4	4
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1
4	4	2	2	1	1	1	1	4	4	4	2	2	1	4	4	2
5	1	1	1	1	2	1	4	4	2	2	1	4	4	2	2	1
6	2	2	1	1	1	1	2	1	1	4	2	2	1	2	1	4
7	1	1	1	1	4	4	2	2	1	1	4	4	1	1	4	1
8	2	1	4	4	2	1	2	1	2	1	4	2	1	4	2	1
9	1	2	1	4	2	1	2	1	2	1	1	4	1	4	1	4
10	2	2	2	2	1	2	1	4	4	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	4	2	2	2	1	2	1	1	1	1	4	1	1	1	1	4
13	4	2	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1	1
4	1	4	4	2	1	1	4	2	1	4	4	4	4	4	1	1
5	4	4	2	2	1	4	4	2	1	1	1	1	1	1	1	1
6	4	2	2	1	4	4	4	4	4	4	4	1	1	1	1	1
7	4	4	4	1	4	4	4	4	4	4	1	1	1	1	1	1
8	1	4	4	1	4	4	4	2	2	1	1	1	1	1	2	2
9	4	4	4	1	4	2	1	4	1	4	1	1	1	2	2	2
10	4	2	2	1	1	1	1	1	4	4	4	2	2	1	1	1
11	4	4	4	2	2	2	1	4	4	4	2	2	1	4	2	2
12	4	4	2	2	2	2	1	4	2	1	1	2	2	1	4	2
13	4	4	2	2	2	1	1	4	2	1	4	2	1	4	1	1
14	1	4	4	2	2	1	1	4	1	4	1	1	1	1	1	1
15	1	4	2	1	2	1	1	2	1	1	2	2	2	1	2	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

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 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE   6 USER INPUT ********
      *
      * CSFACT - Cloudshine shielding factor
  798  SECSFACT001      1.
********   RECORD NUMBER   798 REPLACES RECORD NUMBER    25   ********
  799  SECSFACT002      0.6
********   RECORD NUMBER   799 REPLACES RECORD NUMBER    26   ********
  800  SECSFACT003      0.5
********   RECORD NUMBER   800 REPLACES RECORD NUMBER    27   ********
      *
      * PROTIN - Inhalation protection factor
  801  SEPROTIN001      0.98
********   RECORD NUMBER   801 REPLACES RECORD NUMBER    28   ********
  802  SEPROTIN002      0.46
********   RECORD NUMBER   802 REPLACES RECORD NUMBER    29   ********
  803  SEPROTIN003      0.33
********   RECORD NUMBER   803 REPLACES RECORD NUMBER    30   ********
      *
      * BRRATE - Breathing rates
  804  SEBRRATE001      2.66E-04
********   RECORD NUMBER   804 REPLACES RECORD NUMBER    31   ********
  805  SEBRRATE002      2.66E-04
********   RECORD NUMBER   805 REPLACES RECORD NUMBER    32   ********
  806  SEBRRATE003      2.66E-04
********   RECORD NUMBER   806 REPLACES RECORD NUMBER    33   ********
      *
      * SKPFAC - skin protection factors
  807  SESKPFAC001      0.98
********   RECORD NUMBER   807 REPLACES RECORD NUMBER    34   ********
  808  SESKPFAC002      0.46
********   RECORD NUMBER   808 REPLACES RECORD NUMBER    35   ********
  809  SESKPFAC003      0.33
********   RECORD NUMBER   809 REPLACES RECORD NUMBER    36   ********
      *
      * GSHFAC - groundshine shielding factors
  810  SEGSHFAC001      0.5
********   RECORD NUMBER   810 REPLACES RECORD NUMBER    37   ********
  811  SEGSHFAC002      0.18
********   RECORD NUMBER   811 REPLACES RECORD NUMBER    38   ********
  812  SEGSHFAC003      0.1
********   RECORD NUMBER   812 REPLACES RECORD NUMBER    39   ********
      *
      * EANAM2 - Name of emergency response cohort
  813  EZEANAM2001      '10-30 Public'
********   RECORD NUMBER   813 REPLACES RECORD NUMBER    42   ********
      *
      * WTFRAC - weighting fraction applied to results of emergency response cohort
  814  EZWTFRAC001      0.
********   RECORD NUMBER   814 REPLACES RECORD NUMBER    44   ********
      *
      * TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
  815  TRAVELPOINT      CENTERPOINT
********   RECORD NUMBER   815 REPLACES RECORD NUMBER    46   ********
      *
      * ESPEED - evacuee travel speed during the three phases of evacuation
  816  EZESPEED001      0.894
********   RECORD NUMBER   816 REPLACES RECORD NUMBER    47   ********
  817  EZESPEED002      0.447
********   RECORD NUMBER   817 REPLACES RECORD NUMBER    48   ********
  818  EZESPEED003      8.941
********   RECORD NUMBER   818 REPLACES RECORD NUMBER    49   ********
      *
      * ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
  819  EZESPMUL001      0.7
********   RECORD NUMBER   819 REPLACES RECORD NUMBER    50   ********
  820  EZESPMUL002      0.7
********   RECORD NUMBER   820 REPLACES RECORD NUMBER    51   ********
  821  EZESPMUL003      0.7
********   RECORD NUMBER   821 REPLACES RECORD NUMBER    52   ********
      *
      * REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
  822  EZREFPNT001      ALARM
********   RECORD NUMBER   822 REPLACES RECORD NUMBER    53   ********
      *
      * DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
  823  EZDURBEG001      7200.
********   RECORD NUMBER   823 REPLACES RECORD NUMBER    54   ********
      *
      * DURMID - duration of middle phase of evacuation, in seconds.
  824  EZDURMID001      64800.
********   RECORD NUMBER   824 REPLACES RECORD NUMBER    55   ********
      *
      * NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
  825  EZNUMEVA001      18
********   RECORD NUMBER   825 REPLACES RECORD NUMBER    56   ********
      *
      * DLTSHL - delay from reference time point to when individual takes shelter.  DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
  826  EZDLTSHL001      86400.
********   RECORD NUMBER   826 REPLACES RECORD NUMBER    57   ********
  827  EZDLTSHL002      86400.
********   RECORD NUMBER   827 REPLACES RECORD NUMBER    58   ********
  828  EZDLTSHL003      86400.
********   RECORD NUMBER   828 REPLACES RECORD NUMBER    59   ********
  829  EZDLTSHL004      86400.
********   RECORD NUMBER   829 REPLACES RECORD NUMBER    60   ********
  830  EZDLTSHL005      86400.
********   RECORD NUMBER   830 REPLACES RECORD NUMBER    61   ********
  831  EZDLTSHL006      86400.
********   RECORD NUMBER   831 REPLACES RECORD NUMBER    62   ********
  832  EZDLTSHL007      86400.
********   RECORD NUMBER   832 REPLACES RECORD NUMBER    63   ********
  833  EZDLTSHL008      86400.
********   RECORD NUMBER   833 REPLACES RECORD NUMBER    64   ********
  834  EZDLTSHL009      86400.
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******** RECORD NUMBER 834 REPLACES RECORD NUMBER 65 ********
835 EZDLTSHL010 86400.
******** RECORD NUMBER 835 REPLACES RECORD NUMBER 66 ********
836 EZDLTSHL011 86400.
******** RECORD NUMBER 836 REPLACES RECORD NUMBER 67 ********
837 EZDLTSHL012 86400.
******** RECORD NUMBER 837 REPLACES RECORD NUMBER 68 ********
838 EZDLTSHL013 86400.
******** RECORD NUMBER 838 REPLACES RECORD NUMBER 69 ********
839 EZDLTSHL014 86400.
******** RECORD NUMBER 839 REPLACES RECORD NUMBER 70 ********
840 EZDLTSHL015 86400.
******** RECORD NUMBER 840 REPLACES RECORD NUMBER 71 ********
841 EZDLTSHL016 86400.
******** RECORD NUMBER 841 REPLACES RECORD NUMBER 72 ********
842 EZDLTSHL017 86400.
******** RECORD NUMBER 842 REPLACES RECORD NUMBER 73 ********
843 EZDLTSHL018 86400.
******** RECORD NUMBER 843 REPLACES RECORD NUMBER 74 ********
*
* DLTEVA -Delay time to begin evacuation
844 EZDLTEVA001 14400.
******** RECORD NUMBER 844 REPLACES RECORD NUMBER 75 ********
845 EZDLTEVA002 14400.
******** RECORD NUMBER 845 REPLACES RECORD NUMBER 76 ********
846 EZDLTEVA003 14400.
******** RECORD NUMBER 846 REPLACES RECORD NUMBER 77 ********
847 EZDLTEVA004 14400.
******** RECORD NUMBER 847 REPLACES RECORD NUMBER 78 ********
848 EZDLTEVA005 14400.
******** RECORD NUMBER 848 REPLACES RECORD NUMBER 79 ********
849 EZDLTEVA006 14400.
******** RECORD NUMBER 849 REPLACES RECORD NUMBER 80 ********
850 EZDLTEVA007 14400.
******** RECORD NUMBER 850 REPLACES RECORD NUMBER 81 ********
851 EZDLTEVA008 14400.
******** RECORD NUMBER 851 REPLACES RECORD NUMBER 82 ********
852 EZDLTEVA009 14400.
******** RECORD NUMBER 852 REPLACES RECORD NUMBER 83 ********
853 EZDLTEVA010 14400.
******** RECORD NUMBER 853 REPLACES RECORD NUMBER 84 ********
854 EZDLTEVA011 14400.
******** RECORD NUMBER 854 REPLACES RECORD NUMBER 85 ********
855 EZDLTEVA012 14400.
******** RECORD NUMBER 855 REPLACES RECORD NUMBER 86 ********
856 EZDLTEVA013 14400.
******** RECORD NUMBER 856 REPLACES RECORD NUMBER 87 ********
857 EZDLTEVA014 14400.
******** RECORD NUMBER 857 REPLACES RECORD NUMBER 88 ********
858 EZDLTEVA015 14400.
******** RECORD NUMBER 858 REPLACES RECORD NUMBER 89 ********
859 EZDLTEVA016 14400.
******** RECORD NUMBER 859 REPLACES RECORD NUMBER 90 ********
860 EZDLTEVA017 14400.
******** RECORD NUMBER 860 REPLACES RECORD NUMBER 91 ********
861 EZDLTEVA018 14400.
******** RECORD NUMBER 861 REPLACES RECORD NUMBER 92 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
862 EZESPGRD001 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 862 REPLACES RECORD NUMBER 93 ********
863 EZESPGRD002 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 863 REPLACES RECORD NUMBER 94 ********
864 EZESPGRD003 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 864 REPLACES RECORD NUMBER 95 ********
865 EZESPGRD004 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 865 REPLACES RECORD NUMBER 96 ********
866 EZESPGRD005 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 866 REPLACES RECORD NUMBER 97 ********
867 EZESPGRD006 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 867 REPLACES RECORD NUMBER 98 ********
868 EZESPGRD007 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 868 REPLACES RECORD NUMBER 99 ********
869 EZESPGRD008 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 869 REPLACES RECORD NUMBER 100 ********
870 EZESPGRD009 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 870 REPLACES RECORD NUMBER 101 ********
871 EZESPGRD010 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 871 REPLACES RECORD NUMBER 102 ********
872 EZESPGRD011 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 872 REPLACES RECORD NUMBER 103 ********
873 EZESPGRD012 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 873 REPLACES RECORD NUMBER 104 ********
874 EZESPGRD013 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 874 REPLACES RECORD NUMBER 105 ********
875 EZESPGRD014 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 875 REPLACES RECORD NUMBER 106 ********
876 EZESPGRD015 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 876 REPLACES RECORD NUMBER 107 ********
877 EZESPGRD016 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 877 REPLACES RECORD NUMBER 108 ********
878 EZESPGRD017 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 878 REPLACES RECORD NUMBER 109 ********

```
 879  EZESPGRD018    1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  1.  1.  1.
********  RECORD NUMBER  879 REPLACES RECORD NUMBER  110  ********
 880  EZESPGRD019    1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.  1.
1.  1.  1.  1.  1.  1.  1.
********  RECORD NUMBER  880 REPLACES RECORD NUMBER  111  ********
     *
     * IDIREC - destination direction of every spatial element in the evacuation and sheltering region
 881  EZIDIREC001    1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
1   1   1   1   1   1   1
********  RECORD NUMBER  881 REPLACES RECORD NUMBER  112  ********
 882  EZIDIREC002    1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
1   1   1   1   1   1   1
********  RECORD NUMBER  882 REPLACES RECORD NUMBER  113  ********
 883  EZIDIREC003    1   1   1   1   1   1   1   1   1   1   1   1   1   2   1   1   1   1   1   1   1   1   1   1   2   1   4   4   4   4   2   2   2   2   2   1   1   1   4   4   4   4   2   2   2   1   1   1   1   1   1   1   1   1   1   1   1
1   1   1   1   4   1   1
********  RECORD NUMBER  883 REPLACES RECORD NUMBER  114  ********
 884  EZIDIREC004    1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4   2   2   1   1   1   1   4   4   4   2   2   1   4   4   2   1   4   4   2   1   1   4   2   1
4   4   4   4   4   1   1
********  RECORD NUMBER  884 REPLACES RECORD NUMBER  115  ********
 885  EZIDIREC005    1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4   1   1   1   1   1   1   1   1   1   2   2   2   2   2   2   1   1   1   1   1   2   1   4   4   2   2   1   4   4   2   2   1   4   4   2   2   1   4   4   2   1
1   1   1   1   1   1   1
********  RECORD NUMBER  885 REPLACES RECORD NUMBER  116  ********
 886  EZIDIREC006    1   1   1   1   1   1   1   1   1   1   1   1   1   1   4   1   1   1   1   1   1   1   1   1   1   1   2   2   2   2   2   2   2   2   1   1   1   1   2   1   1   4   2   2   1   2   1   4   4   2   2   1   4   4   4   4   4
4   4   1   1   1   1   1
********  RECORD NUMBER  886 REPLACES RECORD NUMBER  117  ********
 887  EZIDIREC007    2   2   1   2   2   1   2   2   1   4   2   1   4   2   2   1   1   1   1   1   1   1   1   1   1   1   2   2   1   4   4   2   1   1   1   1   4   4   2   2   1   1   4   4   1   1   4   1   4   4   4   1   4   4   4   4   4
4   1   1   1   1   1   1
********  RECORD NUMBER  887 REPLACES RECORD NUMBER  118  ********
 888  EZIDIREC008    1   4   1   1   4   2   1   4   2   1   1   4   2   2   1   1   2   1   1   1   4   4   4   1   1   1   1   1   1   4   1   4   2   1   4   4   2   1   2   1   2   1   4   2   1   4   2   1   1   4   4   1   4   4   4   2   2
1   1   1   1   1   2   2
********  RECORD NUMBER  888 REPLACES RECORD NUMBER  119  ********
 889  EZIDIREC009    1   1   4   2   1   1   2   1   1   4   1   4   1   4   1   1   1   2   1   1   1   4   4   4   1   1   4   1   2   2   2   1   1   2   1   4   2   1   2   1   2   1   1   4   1   4   1   4   4   4   4   1   4   2   1   4   1
4   1   1   1   2   2   2
********  RECORD NUMBER  889 REPLACES RECORD NUMBER  120  ********
 890  EZIDIREC010    1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4   4   4   2   2   2   2   1   2   1   4   4   1   1   1   4   2   1   4   4   2   2   1   1   1   1   1   4
4   4   2   2   1   1   1
********  RECORD NUMBER  890 REPLACES RECORD NUMBER  121  ********
 891  EZIDIREC011    1   1   4   2   1   4   2   1   4   4   4   4   2   2   1   4   4   2   2   2   2   2   2   1   4   4   1   1   4   2   2   1   1   1   4   2   1   4   1   4   2   1   4   1   1   2   1   1   4   4   4   2   2   2   1   4   4
4   2   2   1   4   2   2
********  RECORD NUMBER  891 REPLACES RECORD NUMBER  122  ********
 892  EZIDIREC012    2   1   1   4   1   1   4   1   4   4   2   1   4   2   1   1   4   4   2   2   2   1   2   1   2   1   2   2   1   4   4   4   4   2   2   2   1   2   1   1   1   1   4   1   1   1   1   4   4   4   2   2   2   2   1   4   2
1   1   2   2   1   4   2
********  RECORD NUMBER  892 REPLACES RECORD NUMBER  123  ********
 893  EZIDIREC013    1   1   4   1   4   2   1   1   2   1   4   4   2   2   1   1   1   1   1   1   1   2   1   4   4   4   1   2   1   4   4   4   4   2   2   1   4   4   1   1   4   2   1   1   2   1   2   1   4   4   2   2   2   1   1   4   2
1   4   2   1   4   1   1
********  RECORD NUMBER  893 REPLACES RECORD NUMBER  124  ********
 894  EZIDIREC014    1   1   4   1   1   1   2   1   2   1   2   1   1   2   1   1   1   1   1   1   1   4   1   1   1   1   2   1   4   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   2   1   1   4   4   2   2   1   1   4   1
4   1   1   1   1   1   1
********  RECORD NUMBER  894 REPLACES RECORD NUMBER  125  ********
 895  EZIDIREC015    1   1   4   2   2   2   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4   4   4   2   2   2   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4   1   2   1   1   1   4   2   1   2   1   1   2   1
1   2   2   2   1   2   1
********  RECORD NUMBER  895 REPLACES RECORD NUMBER  126  ********
 896  EZIDIREC016    1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   4   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
1   1   1   1   1   1   1
********  RECORD NUMBER  896 REPLACES RECORD NUMBER  127  ********
 897  EZIDIREC017    1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
1   1   1   1   1   1   1
********  RECORD NUMBER  897 REPLACES RECORD NUMBER  128  ********
 898  EZIDIREC018    1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
1   1   1   1   1   1   1
********  RECORD NUMBER  898 REPLACES RECORD NUMBER  129  ********
 899  EZIDIREC019    1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
1   1   1   1   1   1   1
********  RECORD NUMBER  899 REPLACES RECORD NUMBER  130  ********
     *
     * LASMOV - The outermost spatial interval of the evacuation movement zone.
 900  EZLASMOV001     19
********  RECORD NUMBER  900 REPLACES RECORD NUMBER  131  ********
     *
     * EFFACY,  KI Ingestion
 901  EZEFFACY001     0.7
********  RECORD NUMBER  901 REPLACES RECORD NUMBER  269  ********
     *
     * POPFRAC, KI Ingestion
 902  EZPOPFRC001     0.
********  RECORD NUMBER  902 REPLACES RECORD NUMBER  270  ********
    .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE   6 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE   6
NUMBER OF RECORDS CHANGED          =  105
NUMBER OF RECORDS ADDED            =    0
****************************************************************************************************



With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:


IRAD     1  2  3  4  5  6  7  8  9 10 11 12 13 14 15 16
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    1  1  1  1  1  1  1  1  1  1  1  1  1  2  1  1
  4    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  5    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4
  6    1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  1
  7    2  2  1  2  2  1  2  2  1  4  2  1  4  2  2  1
  8    1  4  1  1  4  2  1  4  2  1  1  4  2  2  1  1
  9    1  1  4  2  1  1  2  1  1  4  1  4  1  4  1  1
 10    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 11    1  1  4  2  1  4  2  1  4  4  4  4  2  2  1  4
 12    2  1  1  4  1  1  4  1  4  4  2  1  4  2  1  1
 13    1  1  4  1  4  2  1  1  2  1  4  4  2  2  1  1
 14    1  1  4  1  1  1  2  1  2  1  2  1  1  2  1  1
 15    1  1  4  2  2  2  1  1  1  1  1  1  1  1  1  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
```

```
 18    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD     17  18  19  20  21  22  23  24  25  26  27  28  29  30  31  32
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    1  1  1  1  1  1  1  1  2  1  4  4  4  4  2  2
  4    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  5    1  1  1  1  1  1  1  1  1  2  2  2  2  2  2  1
  6    1  1  1  1  1  1  1  1  1  1  2  2  2  2  2  2
  7    1  1  1  1  1  1  1  1  1  1  2  2  1  4  4  2
  8    2  1  1  1  4  4  4  1  1  1  1  1  1  4  1  4
  9    1  2  1  1  1  4  4  4  1  1  4  1  2  2  2  1
 10    1  1  1  1  1  1  1  1  1  1  1  1  1  4  4  4
 11    4  2  2  2  2  2  2  1  4  4  1  1  4  2  2  1
 12    4  4  2  2  2  1  2  1  2  1  2  2  1  4  4  4
 13    1  1  1  1  1  2  1  4  4  4  1  2  1  4  4  4
 14    1  1  1  1  1  4  1  1  1  1  2  1  4  1  1  1
 15    1  1  1  1  1  1  1  4  4  4  2  2  2  1  1  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD     33  34  35  36  37  38  39  40  41  42  43  44  45  46  47  48
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    2  2  2  1  1  1  4  4  4  4  2  2  2  1  1  1
  4    4  2  2  1  1  1  1  4  4  4  2  2  1  4  4  2
  5    1  1  1  1  2  1  4  4  2  2  1  4  4  2  2  1
  6    2  2  1  1  1  1  2  1  1  4  2  2  1  2  1  4
  7    1  1  1  1  4  4  2  2  1  1  4  4  1  1  4  1
  8    2  1  4  4  2  1  2  1  2  1  4  2  1  4  2  1
  9    1  2  1  4  2  1  2  1  2  1  1  4  1  4  1  4
 10    2  2  2  2  1  2  1  4  4  1  1  1  4  2  1  4
 11    1  1  4  2  1  4  1  4  2  1  4  1  1  2  1  1
 12    4  2  2  2  1  2  1  1  1  1  4  1  1  1  1  4
 13    4  2  2  1  4  4  1  1  4  2  1  1  2  1  2  1
 14    1  1  1  1  1  1  1  1  1  1  1  1  1  1  2  1
 15    1  1  1  1  1  1  1  1  1  1  1  4  1  2  1  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD     49  50  51  52  53  54  55  56  57  58  59  60  61  62  63  64
  1    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3    1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  1
  4    1  4  4  2  1  1  4  2  1  4  4  4  4  4  1  1
  5    4  4  2  2  1  4  4  2  1  1  1  1  1  1  1  1
  6    4  2  2  1  4  4  4  4  4  4  4  1  1  1  1  1
  7    4  4  4  1  4  4  4  4  4  4  1  1  1  1  1  1
  8    1  4  4  1  4  4  4  2  2  1  1  1  1  1  2  2
  9    4  4  4  1  4  2  1  4  1  4  1  1  1  2  2  2
 10    4  2  2  1  1  1  1  1  4  4  4  2  2  1  1  1
 11    4  4  4  2  2  2  1  4  4  4  2  2  1  4  2  2
 12    4  4  2  2  2  2  1  4  2  1  1  2  2  1  4  2
 13    4  4  2  2  2  1  1  4  2  1  4  2  1  4  1  1
 14    1  4  4  2  2  1  1  4  1  4  1  1  1  1  1  1
 15    1  4  2  1  2  1  1  2  1  1  2  2  2  1  2  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE   7 USER INPUT ********
      *
      * CSFACT - Cloudshine shielding factor
  903  SECSFACT001      1.
********   RECORD NUMBER   903 REPLACES RECORD NUMBER    25   ********
  904  SECSFACT002      0.31
********   RECORD NUMBER   904 REPLACES RECORD NUMBER    26   ********
  905  SECSFACT003      0.31
********   RECORD NUMBER   905 REPLACES RECORD NUMBER    27   ********
      *
      * PROTIN - Inhalation protection factor
  906  SEPROTIN001     0.98
********   RECORD NUMBER   906 REPLACES RECORD NUMBER    28   ********
  907  SEPROTIN002     0.33
********   RECORD NUMBER   907 REPLACES RECORD NUMBER    29   ********
  908  SEPROTIN003     0.33
********   RECORD NUMBER   908 REPLACES RECORD NUMBER    30   ********
      *
      * BRRATE - Breathing rates
  909  SEBRRATE001      2.66E-04
********   RECORD NUMBER   909 REPLACES RECORD NUMBER    31   ********
  910  SEBRRATE002      2.66E-04
********   RECORD NUMBER   910 REPLACES RECORD NUMBER    32   ********
  911  SEBRRATE003      2.66E-04
********   RECORD NUMBER   911 REPLACES RECORD NUMBER    33   ********
      *
      * SKPFAC - skin protection factors
  912  SESKPFAC001      0.98
********   RECORD NUMBER   912 REPLACES RECORD NUMBER    34   ********
  913  SESKPFAC002      0.33
********   RECORD NUMBER   913 REPLACES RECORD NUMBER    35   ********
  914  SESKPFAC003      0.33
********   RECORD NUMBER   914 REPLACES RECORD NUMBER    36   ********
      *
      * GSHFAC - groundshine shielding factors
  915  SEGSHFAC001      0.5
********   RECORD NUMBER   915 REPLACES RECORD NUMBER    37   ********
  916  SEGSHFAC002      0.05
********   RECORD NUMBER   916 REPLACES RECORD NUMBER    38   ********
  917  SEGSHFAC003      0.05
********   RECORD NUMBER   917 REPLACES RECORD NUMBER    39   ********
      *
```

```
    * EANAM2 - Name of emergency response cohort
 918  EZEANAM2001      '10-30 Special Facilities'
********  RECORD NUMBER   918 REPLACES RECORD NUMBER    42  ********
    *
    * WTFRAC - weighting fraction applied to results of emergency response cohort
 919  EZWTFRAC001      0.
********  RECORD NUMBER   919 REPLACES RECORD NUMBER    44  ********
    *
    * TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
 920  TRAVELPOINT      CENTERPOINT
********  RECORD NUMBER   920 REPLACES RECORD NUMBER    46  ********
    *
    * ESPEED - evacuee travel speed during the three phases of evacuation
 921  EZESPEED001      0.447
********  RECORD NUMBER   921 REPLACES RECORD NUMBER    47  ********
 922  EZESPEED002      0.447
********  RECORD NUMBER   922 REPLACES RECORD NUMBER    48  ********
 923  EZESPEED003      8.941
********  RECORD NUMBER   923 REPLACES RECORD NUMBER    49  ********
    *
    * ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
 924  EZESPMUL001      0.7
********  RECORD NUMBER   924 REPLACES RECORD NUMBER    50  ********
 925  EZESPMUL002      0.7
********  RECORD NUMBER   925 REPLACES RECORD NUMBER    51  ********
 926  EZESPMUL003      0.7
********  RECORD NUMBER   926 REPLACES RECORD NUMBER    52  ********
    *
    * REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
 927  EZREFPNT001      ALARM
********  RECORD NUMBER   927 REPLACES RECORD NUMBER    53  ********
    *
    * DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
 928  EZDURBEG001      3600.
********  RECORD NUMBER   928 REPLACES RECORD NUMBER    54  ********
    *
    * DURMID - duration of middle phase of evacuation, in seconds.
 929  EZDURMID001      36000.
********  RECORD NUMBER   929 REPLACES RECORD NUMBER    55  ********
    *
    * NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
 930  EZNUMEVA001      18
********  RECORD NUMBER   930 REPLACES RECORD NUMBER    56  ********
    *
    * DLTSHL - delay from reference time point to when individual takes shelter.  DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
 931  EZDLTSHL001      54000.
********  RECORD NUMBER   931 REPLACES RECORD NUMBER    57  ********
 932  EZDLTSHL002      54000.
********  RECORD NUMBER   932 REPLACES RECORD NUMBER    58  ********
 933  EZDLTSHL003      54000.
********  RECORD NUMBER   933 REPLACES RECORD NUMBER    59  ********
 934  EZDLTSHL004      54000.
********  RECORD NUMBER   934 REPLACES RECORD NUMBER    60  ********
 935  EZDLTSHL005      54000.
********  RECORD NUMBER   935 REPLACES RECORD NUMBER    61  ********
 936  EZDLTSHL006      54000.
********  RECORD NUMBER   936 REPLACES RECORD NUMBER    62  ********
 937  EZDLTSHL007      54000.
********  RECORD NUMBER   937 REPLACES RECORD NUMBER    63  ********
 938  EZDLTSHL008      54000.
********  RECORD NUMBER   938 REPLACES RECORD NUMBER    64  ********
 939  EZDLTSHL009      54000.
********  RECORD NUMBER   939 REPLACES RECORD NUMBER    65  ********
 940  EZDLTSHL010      54000.
********  RECORD NUMBER   940 REPLACES RECORD NUMBER    66  ********
 941  EZDLTSHL011      54000.
********  RECORD NUMBER   941 REPLACES RECORD NUMBER    67  ********
 942  EZDLTSHL012      54000.
********  RECORD NUMBER   942 REPLACES RECORD NUMBER    68  ********
 943  EZDLTSHL013      54000.
********  RECORD NUMBER   943 REPLACES RECORD NUMBER    69  ********
 944  EZDLTSHL014      54000.
********  RECORD NUMBER   944 REPLACES RECORD NUMBER    70  ********
 945  EZDLTSHL015      54000.
********  RECORD NUMBER   945 REPLACES RECORD NUMBER    71  ********
 946  EZDLTSHL016      54000.
********  RECORD NUMBER   946 REPLACES RECORD NUMBER    72  ********
 947  EZDLTSHL017      54000.
********  RECORD NUMBER   947 REPLACES RECORD NUMBER    73  ********
 948  EZDLTSHL018      54000.
********  RECORD NUMBER   948 REPLACES RECORD NUMBER    74  ********
    *
    * DLTEVA -Delay time to begin evacuation
 949  EZDLTEVA001      54000.
********  RECORD NUMBER   949 REPLACES RECORD NUMBER    75  ********
 950  EZDLTEVA002      54000.
********  RECORD NUMBER   950 REPLACES RECORD NUMBER    76  ********
 951  EZDLTEVA003      54000.
********  RECORD NUMBER   951 REPLACES RECORD NUMBER    77  ********
 952  EZDLTEVA004      54000.
********  RECORD NUMBER   952 REPLACES RECORD NUMBER    78  ********
 953  EZDLTEVA005      54000.
********  RECORD NUMBER   953 REPLACES RECORD NUMBER    79  ********
 954  EZDLTEVA006      54000.
********  RECORD NUMBER   954 REPLACES RECORD NUMBER    80  ********
 955  EZDLTEVA007      54000.
********  RECORD NUMBER   955 REPLACES RECORD NUMBER    81  ********
 956  EZDLTEVA008      54000.
********  RECORD NUMBER   956 REPLACES RECORD NUMBER    82  ********
 957  EZDLTEVA009      54000.
********  RECORD NUMBER   957 REPLACES RECORD NUMBER    83  ********
 958  EZDLTEVA010      54000.
********  RECORD NUMBER   958 REPLACES RECORD NUMBER    84  ********
 959  EZDLTEVA011      54000.
********  RECORD NUMBER   959 REPLACES RECORD NUMBER    85  ********
 960  EZDLTEVA012      54000.
********  RECORD NUMBER   960 REPLACES RECORD NUMBER    86  ********
 961  EZDLTEVA013      54000.
********  RECORD NUMBER   961 REPLACES RECORD NUMBER    87  ********
 962  EZDLTEVA014      54000.
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******** RECORD NUMBER 962 REPLACES RECORD NUMBER 88 ********
963 EZDLTEVA015 54000.
******** RECORD NUMBER 963 REPLACES RECORD NUMBER 89 ********
964 EZDLTEVA016 54000.
******** RECORD NUMBER 964 REPLACES RECORD NUMBER 90 ********
965 EZDLTEVA017 54000.
******** RECORD NUMBER 965 REPLACES RECORD NUMBER 91 ********
966 EZDLTEVA018 54000.
******** RECORD NUMBER 966 REPLACES RECORD NUMBER 92 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
967 EZESPGRD001 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 967 REPLACES RECORD NUMBER 93 ********
968 EZESPGRD002 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 968 REPLACES RECORD NUMBER 94 ********
969 EZESPGRD003 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 969 REPLACES RECORD NUMBER 95 ********
970 EZESPGRD004 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 970 REPLACES RECORD NUMBER 96 ********
971 EZESPGRD005 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 971 REPLACES RECORD NUMBER 97 ********
972 EZESPGRD006 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 972 REPLACES RECORD NUMBER 98 ********
973 EZESPGRD007 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 973 REPLACES RECORD NUMBER 99 ********
974 EZESPGRD008 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 974 REPLACES RECORD NUMBER 100 ********
975 EZESPGRD009 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 975 REPLACES RECORD NUMBER 101 ********
976 EZESPGRD010 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 976 REPLACES RECORD NUMBER 102 ********
977 EZESPGRD011 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 977 REPLACES RECORD NUMBER 103 ********
978 EZESPGRD012 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 978 REPLACES RECORD NUMBER 104 ********
979 EZESPGRD013 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 979 REPLACES RECORD NUMBER 105 ********
980 EZESPGRD014 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 980 REPLACES RECORD NUMBER 106 ********
981 EZESPGRD015 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 981 REPLACES RECORD NUMBER 107 ********
982 EZESPGRD016 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 982 REPLACES RECORD NUMBER 108 ********
983 EZESPGRD017 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 983 REPLACES RECORD NUMBER 109 ********
984 EZESPGRD018 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 984 REPLACES RECORD NUMBER 110 ********
985 EZESPGRD019 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 985 REPLACES RECORD NUMBER 111 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
986 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 986 REPLACES RECORD NUMBER 112 ********
987 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 987 REPLACES RECORD NUMBER 113 ********
988 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 4 1 1
******** RECORD NUMBER 988 REPLACES RECORD NUMBER 114 ********
989 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2 1
4 4 4 4 4 1 1
******** RECORD NUMBER 989 REPLACES RECORD NUMBER 115 ********
990 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 1
1 1 1 1 1 1 1
******** RECORD NUMBER 990 REPLACES RECORD NUMBER 116 ********
991 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4 4
4 4 1 1 1 1 1
******** RECORD NUMBER 991 REPLACES RECORD NUMBER 117 ********
992 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 4 4 1 4 4 4 4 4
4 1 1 1 1 1 1
******** RECORD NUMBER 992 REPLACES RECORD NUMBER 118 ********
993 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 4 1 4 4 4 2 2
1 1 1 1 1 2 2
******** RECORD NUMBER 993 REPLACES RECORD NUMBER 119 ********
994 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 4 4 1 4 2 1 4 1
4 1 1 1 2 2 2
******** RECORD NUMBER 994 REPLACES RECORD NUMBER 120 ********
995 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 4 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1 4
4 4 2 2 1 1 1
******** RECORD NUMBER 995 REPLACES RECORD NUMBER 121 ********
996 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 4 4 2 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4 4
4 2 2 1 4 2 2
******** RECORD NUMBER 996 REPLACES RECORD NUMBER 122 ********
997 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 2 1 2 1 2 2 1 4 4 4 4 2 2 2 1 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4 2
1 1 2 2 1 4 2
******** RECORD NUMBER 997 REPLACES RECORD NUMBER 123 ********
998 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 4 4 4 1 2 1 4 4 4 4 2 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4 2
1 4 2 1 4 1 1
******** RECORD NUMBER 998 REPLACES RECORD NUMBER 124 ********
999 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4 1

4 1 1 1 1 1 1
******** RECORD NUMBER 999 REPLACES RECORD NUMBER 125 ********
1000 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2 1
1 2 2 2 1 2 1
******** RECORD NUMBER 1000 REPLACES RECORD NUMBER 126 ********
1001 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1001 REPLACES RECORD NUMBER 127 ********
1002 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1002 REPLACES RECORD NUMBER 128 ********
1003 EZIDIREC018 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1003 REPLACES RECORD NUMBER 129 ********
1004 EZIDIREC019 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1004 REPLACES RECORD NUMBER 130 ********
*
* LASMOV - The outermost spatial interval of the evacuation movement zone.
1005 EZLASMOV001 19
******** RECORD NUMBER 1005 REPLACES RECORD NUMBER 131 ********
*
* EFFACY, KI Ingestion
1006 EZEFFACY001 0.7
******** RECORD NUMBER 1006 REPLACES RECORD NUMBER 269 ********
*
* POPFRAC, KI Ingestion
1007 EZPOPFRC001 0.
******** RECORD NUMBER 1007 REPLACES RECORD NUMBER 270 ********
.
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE 7 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE 7
NUMBER OF RECORDS CHANGED = 105
NUMBER OF RECORDS ADDED = 0
****

With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:

IRAD	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1	1
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	4	4	2	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	2	1	2	1	2	2	1	4	4	4
13	1	1	1	1	1	2	1	4	4	4	1	2	1	4	4	4
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1
4	4	2	2	1	1	1	1	4	4	4	2	2	1	4	4	2
5	1	1	1	1	2	1	4	4	2	2	1	4	4	2	2	1
6	2	2	1	1	1	1	2	1	1	4	2	2	1	2	1	4
7	1	1	1	1	4	4	2	2	1	1	4	4	1	1	4	1
8	2	1	4	4	2	1	2	1	2	1	4	2	1	4	2	1
9	1	2	1	4	2	1	2	1	2	1	1	4	1	4	1	4
10	2	2	2	2	1	2	1	4	4	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	4	2	2	2	1	2	1	1	1	1	4	1	1	1	1	4
13	4	2	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

```
  3    1  1  1  1  1  1  1  1  1  1  1  1  1  4  1  1
  4    1  4  4  2  1  1  4  2  1  4  4  4  4  4  1  1
  5    4  4  2  2  1  4  4  2  1  1  1  1  1  1  1  1
  6    4  2  2  1  4  4  4  4  4  4  4  1  1  1  1  1
  7    4  4  4  1  4  4  4  4  4  4  1  1  1  1  1  1
  8    1  4  4  1  4  4  4  2  2  1  1  1  1  1  2  2
  9    4  4  4  1  4  2  1  4  1  4  1  1  1  2  2  2
 10    4  2  2  1  1  1  1  1  4  4  4  2  2  1  1  1
 11    4  4  4  2  2  2  1  4  4  4  2  2  1  4  2  2
 12    4  4  2  2  2  2  1  4  2  1  1  2  2  1  4  2
 13    4  4  2  2  2  1  1  4  2  1  4  2  1  4  1  1
 14    1  4  4  2  2  1  1  4  1  4  1  1  1  1  1  1
 15    1  4  2  1  2  1  1  2  1  1  2  2  2  1  2  1
 16    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19    1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE   8 USER INPUT ********
     *
     * CSFACT - Cloudshine shielding factor
 1008  SECSFACT001      1.
********   RECORD NUMBER  1008 REPLACES RECORD NUMBER    25  ********
 1009  SECSFACT002      0.6
********   RECORD NUMBER  1009 REPLACES RECORD NUMBER    26  ********
 1010  SECSFACT003      0.5
********   RECORD NUMBER  1010 REPLACES RECORD NUMBER    27  ********
     *
     * PROTIN - Inhalation protection factor
 1011  SEPROTIN001      0.98
********   RECORD NUMBER  1011 REPLACES RECORD NUMBER    28  ********
 1012  SEPROTIN002      0.46
********   RECORD NUMBER  1012 REPLACES RECORD NUMBER    29  ********
 1013  SEPROTIN003      0.33
********   RECORD NUMBER  1013 REPLACES RECORD NUMBER    30  ********
     *
     * BRRATE - Breathing rates
 1014  SEBRRATE001      2.66E-04
********   RECORD NUMBER  1014 REPLACES RECORD NUMBER    31  ********
 1015  SEBRRATE002      2.66E-04
********   RECORD NUMBER  1015 REPLACES RECORD NUMBER    32  ********
 1016  SEBRRATE003      2.66E-04
********   RECORD NUMBER  1016 REPLACES RECORD NUMBER    33  ********
     *
     * SKPFAC - skin protection factors
 1017  SESKPFAC001      0.98
********   RECORD NUMBER  1017 REPLACES RECORD NUMBER    34  ********
 1018  SESKPFAC002      0.46
********   RECORD NUMBER  1018 REPLACES RECORD NUMBER    35  ********
 1019  SESKPFAC003      0.33
********   RECORD NUMBER  1019 REPLACES RECORD NUMBER    36  ********
     *
     * GSHFAC - groundshine shielding factors
 1020  SEGSHFAC001      0.5
********   RECORD NUMBER  1020 REPLACES RECORD NUMBER    37  ********
 1021  SEGSHFAC002      0.18
********   RECORD NUMBER  1021 REPLACES RECORD NUMBER    38  ********
 1022  SEGSHFAC003      0.1
********   RECORD NUMBER  1022 REPLACES RECORD NUMBER    39  ********
     *
     * EANAM2 - Name of emergency response cohort
 1023  EZEANAM2001      '30-40 Shadow'
********   RECORD NUMBER  1023 REPLACES RECORD NUMBER    42  ********
     *
     * WTFRAC - weighting fraction applied to results of emergency response cohort
 1024  EZWTFRAC001      0.
********   RECORD NUMBER  1024 REPLACES RECORD NUMBER    44  ********
     *
     * TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
 1025  TRAVELPOINT      CENTERPOINT
********   RECORD NUMBER  1025 REPLACES RECORD NUMBER    46  ********
     *
     * ESPEED - evacuee travel speed during the three phases of evacuation
 1026  EZESPEED001      6.706
********   RECORD NUMBER  1026 REPLACES RECORD NUMBER    47  ********
 1027  EZESPEED002      2.235
********   RECORD NUMBER  1027 REPLACES RECORD NUMBER    48  ********
 1028  EZESPEED003      8.941
********   RECORD NUMBER  1028 REPLACES RECORD NUMBER    49  ********
     *
     * ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
 1029  EZESPMUL001      0.7
********   RECORD NUMBER  1029 REPLACES RECORD NUMBER    50  ********
 1030  EZESPMUL002      0.7
********   RECORD NUMBER  1030 REPLACES RECORD NUMBER    51  ********
 1031  EZESPMUL003      0.7
********   RECORD NUMBER  1031 REPLACES RECORD NUMBER    52  ********
     *
     * REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
 1032  EZREFPNT001      ALARM
********   RECORD NUMBER  1032 REPLACES RECORD NUMBER    53  ********
     *
     * DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
 1033  EZDURBEG001      3600.
********   RECORD NUMBER  1033 REPLACES RECORD NUMBER    54  ********
     *
     * DURMID - duration of middle phase of evacuation, in seconds.
 1034  EZDURMID001      21600.
********   RECORD NUMBER  1034 REPLACES RECORD NUMBER    55  ********
     *
     * NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
 1035  EZNUMEVA001      18
********   RECORD NUMBER  1035 REPLACES RECORD NUMBER    56  ********
     *
     * DLTSHL - delay from reference time point to when individual takes shelter.  DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
 1036  EZDLTSHL001      86400.
********   RECORD NUMBER  1036 REPLACES RECORD NUMBER    57  ********
 1037  EZDLTSHL002      86400.
```

******** RECORD NUMBER 1037 REPLACES RECORD NUMBER 58 ********
1038 EZDLTSHL003 86400.
******** RECORD NUMBER 1038 REPLACES RECORD NUMBER 59 ********
1039 EZDLTSHL004 86400.
******** RECORD NUMBER 1039 REPLACES RECORD NUMBER 60 ********
1040 EZDLTSHL005 86400.
******** RECORD NUMBER 1040 REPLACES RECORD NUMBER 61 ********
1041 EZDLTSHL006 86400.
******** RECORD NUMBER 1041 REPLACES RECORD NUMBER 62 ********
1042 EZDLTSHL007 86400.
******** RECORD NUMBER 1042 REPLACES RECORD NUMBER 63 ********
1043 EZDLTSHL008 86400.
******** RECORD NUMBER 1043 REPLACES RECORD NUMBER 64 ********
1044 EZDLTSHL009 86400.
******** RECORD NUMBER 1044 REPLACES RECORD NUMBER 65 ********
1045 EZDLTSHL010 86400.
******** RECORD NUMBER 1045 REPLACES RECORD NUMBER 66 ********
1046 EZDLTSHL011 86400.
******** RECORD NUMBER 1046 REPLACES RECORD NUMBER 67 ********
1047 EZDLTSHL012 86400.
******** RECORD NUMBER 1047 REPLACES RECORD NUMBER 68 ********
1048 EZDLTSHL013 86400.
******** RECORD NUMBER 1048 REPLACES RECORD NUMBER 69 ********
1049 EZDLTSHL014 86400.
******** RECORD NUMBER 1049 REPLACES RECORD NUMBER 70 ********
1050 EZDLTSHL015 86400.
******** RECORD NUMBER 1050 REPLACES RECORD NUMBER 71 ********
1051 EZDLTSHL016 86400.
******** RECORD NUMBER 1051 REPLACES RECORD NUMBER 72 ********
1052 EZDLTSHL017 86400.
******** RECORD NUMBER 1052 REPLACES RECORD NUMBER 73 ********
1053 EZDLTSHL018 86400.
******** RECORD NUMBER 1053 REPLACES RECORD NUMBER 74 ********
*
* DLTEVA -Delay time to begin evacuation
1054 EZDLTEVA001 28800.
******** RECORD NUMBER 1054 REPLACES RECORD NUMBER 75 ********
1055 EZDLTEVA002 28800.
******** RECORD NUMBER 1055 REPLACES RECORD NUMBER 76 ********
1056 EZDLTEVA003 28800.
******** RECORD NUMBER 1056 REPLACES RECORD NUMBER 77 ********
1057 EZDLTEVA004 28800.
******** RECORD NUMBER 1057 REPLACES RECORD NUMBER 78 ********
1058 EZDLTEVA005 28800.
******** RECORD NUMBER 1058 REPLACES RECORD NUMBER 79 ********
1059 EZDLTEVA006 28800.
******** RECORD NUMBER 1059 REPLACES RECORD NUMBER 80 ********
1060 EZDLTEVA007 28800.
******** RECORD NUMBER 1060 REPLACES RECORD NUMBER 81 ********
1061 EZDLTEVA008 28800.
******** RECORD NUMBER 1061 REPLACES RECORD NUMBER 82 ********
1062 EZDLTEVA009 28800.
******** RECORD NUMBER 1062 REPLACES RECORD NUMBER 83 ********
1063 EZDLTEVA010 28800.
******** RECORD NUMBER 1063 REPLACES RECORD NUMBER 84 ********
1064 EZDLTEVA011 28800.
******** RECORD NUMBER 1064 REPLACES RECORD NUMBER 85 ********
1065 EZDLTEVA012 28800.
******** RECORD NUMBER 1065 REPLACES RECORD NUMBER 86 ********
1066 EZDLTEVA013 28800.
******** RECORD NUMBER 1066 REPLACES RECORD NUMBER 87 ********
1067 EZDLTEVA014 28800.
******** RECORD NUMBER 1067 REPLACES RECORD NUMBER 88 ********
1068 EZDLTEVA015 28800.
******** RECORD NUMBER 1068 REPLACES RECORD NUMBER 89 ********
1069 EZDLTEVA016 28800.
******** RECORD NUMBER 1069 REPLACES RECORD NUMBER 90 ********
1070 EZDLTEVA017 28800.
******** RECORD NUMBER 1070 REPLACES RECORD NUMBER 91 ********
1071 EZDLTEVA018 28800.
******** RECORD NUMBER 1071 REPLACES RECORD NUMBER 92 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
1072 EZESPGRD001 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1072 REPLACES RECORD NUMBER 93 ********
1073 EZESPGRD002 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1073 REPLACES RECORD NUMBER 94 ********
1074 EZESPGRD003 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1074 REPLACES RECORD NUMBER 95 ********
1075 EZESPGRD004 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1075 REPLACES RECORD NUMBER 96 ********
1076 EZESPGRD005 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1076 REPLACES RECORD NUMBER 97 ********
1077 EZESPGRD006 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1077 REPLACES RECORD NUMBER 98 ********
1078 EZESPGRD007 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1078 REPLACES RECORD NUMBER 99 ********
1079 EZESPGRD008 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1079 REPLACES RECORD NUMBER 100 ********
1080 EZESPGRD009 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1080 REPLACES RECORD NUMBER 101 ********
1081 EZESPGRD010 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1081 REPLACES RECORD NUMBER 102 ********
1082 EZESPGRD011 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1082 REPLACES RECORD NUMBER 103 ********
1083 EZESPGRD012 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1083 REPLACES RECORD NUMBER 104 ********
1084 EZESPGRD013 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.

```
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1084 REPLACES RECORD NUMBER 105 ********
 1085 EZESPGRD014 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1085 REPLACES RECORD NUMBER 106 ********
 1086 EZESPGRD015 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1086 REPLACES RECORD NUMBER 107 ********
 1087 EZESPGRD016 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1087 REPLACES RECORD NUMBER 108 ********
 1088 EZESPGRD017 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1088 REPLACES RECORD NUMBER 109 ********
 1089 EZESPGRD018 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 1089 REPLACES RECORD NUMBER 110 ********
 1090 EZESPGRD019 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 1090 REPLACES RECORD NUMBER 111 ********
   *
   * IDIREC - destination direction of every spatial element in the evacuation and sheltering region
 1091 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1091 REPLACES RECORD NUMBER 112 ********
 1092 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1092 REPLACES RECORD NUMBER 113 ********
 1093 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 4 1 1
******** RECORD NUMBER 1093 REPLACES RECORD NUMBER 114 ********
 1094 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2 1
4 4 4 4 4 1 1
******** RECORD NUMBER 1094 REPLACES RECORD NUMBER 115 ********
 1095 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1095 REPLACES RECORD NUMBER 116 ********
 1096 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4 4
4 4 1 1 1 1 1
******** RECORD NUMBER 1096 REPLACES RECORD NUMBER 117 ********
 1097 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 4 4 1 4 4 4 4 4
4 1 1 1 1 1 1
******** RECORD NUMBER 1097 REPLACES RECORD NUMBER 118 ********
 1098 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 4 1 4 4 4 2 2
1 1 1 1 1 2 2
******** RECORD NUMBER 1098 REPLACES RECORD NUMBER 119 ********
 1099 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 4 4 1 4 2 1 4 1
4 1 1 1 2 2 2
******** RECORD NUMBER 1099 REPLACES RECORD NUMBER 120 ********
 1100 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 4 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1 4
4 4 2 2 1 1 1
******** RECORD NUMBER 1100 REPLACES RECORD NUMBER 121 ********
 1101 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 4 4 2 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4 4
4 2 2 1 4 2 2
******** RECORD NUMBER 1101 REPLACES RECORD NUMBER 122 ********
 1102 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 2 1 2 1 2 2 1 4 4 4 4 2 2 2 1 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4 2
1 1 2 2 1 4 2
******** RECORD NUMBER 1102 REPLACES RECORD NUMBER 123 ********
 1103 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 4 4 4 1 2 1 4 4 4 4 2 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4 2
1 4 2 1 4 1 1
******** RECORD NUMBER 1103 REPLACES RECORD NUMBER 124 ********
 1104 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4 1
4 1 1 1 1 1 1
******** RECORD NUMBER 1104 REPLACES RECORD NUMBER 125 ********
 1105 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2 1
1 2 2 2 1 2 1
******** RECORD NUMBER 1105 REPLACES RECORD NUMBER 126 ********
 1106 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1106 REPLACES RECORD NUMBER 127 ********
 1107 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1107 REPLACES RECORD NUMBER 128 ********
 1108 EZIDIREC018 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1108 REPLACES RECORD NUMBER 129 ********
 1109 EZIDIREC019 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1109 REPLACES RECORD NUMBER 130 ********
   *
   * LASMOV - The outermost spatial interval of the evacuation movement zone.
 1110 EZLASMOV001 19
******** RECORD NUMBER 1110 REPLACES RECORD NUMBER 131 ********
   *
   * EFFACY,  KI Ingestion
 1111 EZEFFACY001 0.7
******** RECORD NUMBER 1111 REPLACES RECORD NUMBER 269 ********
   *
   * POPFRAC, KI Ingestion
 1112 EZPOPFRC001 0.
******** RECORD NUMBER 1112 REPLACES RECORD NUMBER 270 ********
   .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE   8 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE   8
NUMBER OF RECORDS CHANGED          =  105
NUMBER OF RECORDS ADDED            =    0
****************************************************************************************************



With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:


IRAD    1  2  3  4  5  6  7  8  9 10 11 12 13 14 15 16
  1     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3     1  1  1  1  1  1  1  1  1  1  1  1  1  2  1  1
```

4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1
7	2	2	1	2	2	1	2	2	1	4	2	1	4	2	2	1
8	1	4	1	1	4	2	1	4	2	1	1	4	2	2	1	1
9	1	1	4	2	1	1	2	1	1	4	1	4	1	4	1	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
11	1	1	4	2	1	4	2	1	4	4	4	4	2	2	1	4
12	2	1	1	4	1	1	4	1	4	4	2	1	4	2	1	1
13	1	1	4	1	4	2	1	1	2	1	4	4	2	2	1	1
14	1	1	4	1	1	1	2	1	2	1	2	1	1	2	1	1
15	1	1	4	2	2	2	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	4
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	2	1	4	4	4	4	2	2
4	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2	1
6	1	1	1	1	1	1	1	1	1	1	2	2	2	2	2	2
7	1	1	1	1	1	1	1	1	1	1	2	2	1	4	4	2
8	2	1	1	1	4	4	4	1	1	1	1	1	1	4	1	4
9	1	2	1	1	1	4	4	4	1	1	4	1	2	2	2	1
10	1	1	1	1	1	1	1	1	1	1	1	1	1	4	4	4
11	4	2	2	2	2	2	2	1	4	4	1	1	4	2	2	1
12	4	4	2	2	2	1	2	1	2	1	2	2	1	4	4	4
13	1	1	1	1	1	2	1	4	4	4	1	2	1	4	4	4
14	1	1	1	1	1	4	1	1	1	1	2	1	4	1	1	1
15	1	1	1	1	1	1	1	4	4	4	2	2	2	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	2	2	2	1	1	1	4	4	4	4	2	2	2	1	1	1
4	4	2	2	1	1	1	1	4	4	4	2	2	1	4	4	2
5	1	1	1	1	2	1	4	4	2	2	1	4	4	2	2	1
6	2	2	1	1	1	1	2	1	1	4	2	2	1	2	1	4
7	1	1	1	1	4	4	2	2	1	1	4	4	1	1	4	1
8	2	1	4	4	2	1	2	1	2	1	4	2	1	4	2	1
9	1	2	1	4	2	1	2	1	2	1	1	4	1	4	1	4
10	2	2	2	2	1	2	1	4	4	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	4	2	2	2	1	2	1	1	1	1	4	1	1	1	1	4
13	4	2	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1	1
4	1	4	4	2	1	1	4	2	1	4	4	4	4	4	1	1
5	4	4	2	2	1	4	4	2	1	1	1	1	1	1	1	1
6	4	2	2	1	4	4	4	4	4	4	4	1	1	1	1	1
7	4	4	4	1	4	4	4	4	4	4	1	1	1	1	1	1
8	1	4	4	1	4	4	4	2	2	1	1	1	1	1	2	2
9	4	4	4	1	4	2	1	4	1	4	1	1	1	2	2	2
10	4	2	2	1	1	1	1	1	4	4	4	2	2	1	1	1
11	4	4	4	2	2	2	1	4	4	4	2	2	1	4	2	2
12	4	4	2	2	2	2	1	4	2	1	1	2	2	1	4	2
13	4	4	2	2	2	1	1	4	2	1	4	2	1	4	1	1
14	1	4	4	2	2	1	1	4	1	4	1	1	1	1	1	1
15	1	4	2	1	2	1	1	2	1	1	2	2	2	1	2	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

THE KI MODEL IS IN EFFECT

******** BEGINNING OF CHANGE CASE 9 USER INPUT ********
*
* CSFACT - Cloudshine shielding factor
1113 SECSFACT001 1.
******** RECORD NUMBER 1113 REPLACES RECORD NUMBER 25 ********
1114 SECSFACT002 0.6
******** RECORD NUMBER 1114 REPLACES RECORD NUMBER 26 ********
1115 SECSFACT003 0.5
******** RECORD NUMBER 1115 REPLACES RECORD NUMBER 27 ********
*
* PROTIN - Inhalation protection factor
1116 SEPROTIN001 0.98
******** RECORD NUMBER 1116 REPLACES RECORD NUMBER 28 ********
1117 SEPROTIN002 0.46
******** RECORD NUMBER 1117 REPLACES RECORD NUMBER 29 ********
1118 SEPROTIN003 0.33
******** RECORD NUMBER 1118 REPLACES RECORD NUMBER 30 ********
*
* BRRATE - Breathing rates
1119 SEBRRATE001 2.66E-04
******** RECORD NUMBER 1119 REPLACES RECORD NUMBER 31 ********
1120 SEBRRATE002 2.66E-04
******** RECORD NUMBER 1120 REPLACES RECORD NUMBER 32 ********
1121 SEBRRATE003 2.66E-04
******** RECORD NUMBER 1121 REPLACES RECORD NUMBER 33 ********
*
* SKPFAC - skin protection factors
1122 SESKPFAC001 0.98

******** RECORD NUMBER 1122 REPLACES RECORD NUMBER 34 ********
1123 SESKPFAC002 0.46
******** RECORD NUMBER 1123 REPLACES RECORD NUMBER 35 ********
1124 SESKPFAC003 0.33
******** RECORD NUMBER 1124 REPLACES RECORD NUMBER 36 ********
*
* GSHFAC - groundshine shielding factors
1125 SEGSHFAC001 0.5
******** RECORD NUMBER 1125 REPLACES RECORD NUMBER 37 ********
1126 SEGSHFAC002 0.18
******** RECORD NUMBER 1126 REPLACES RECORD NUMBER 38 ********
1127 SEGSHFAC003 0.1
******** RECORD NUMBER 1127 REPLACES RECORD NUMBER 39 ********
*
* EANAM2 - Name of emergency response cohort
1128 EZEANAM2001 '10-30 Tail'
******** RECORD NUMBER 1128 REPLACES RECORD NUMBER 42 ********
*
* WTFRAC - weighting fraction applied to results of emergency response cohort
1129 EZWTFRAC001 0.
******** RECORD NUMBER 1129 REPLACES RECORD NUMBER 44 ********
*
* TRAVELPOINT - determines whether boundary or centerpoint of destination is evacuee objective.
1130 TRAVELPOINT CENTERPOINT
******** RECORD NUMBER 1130 REPLACES RECORD NUMBER 46 ********
*
* ESPEED - evacuee travel speed during the three phases of evacuation
1131 EZESPEED001 0.447
******** RECORD NUMBER 1131 REPLACES RECORD NUMBER 47 ********
1132 EZESPEED002 4.47
******** RECORD NUMBER 1132 REPLACES RECORD NUMBER 48 ********
1133 EZESPEED003 8.941
******** RECORD NUMBER 1133 REPLACES RECORD NUMBER 49 ********
*
* ESPMUL - Multiplicative factor that affects ESPEED, applied during times of precipitation.
1134 EZESPMUL001 0.7
******** RECORD NUMBER 1134 REPLACES RECORD NUMBER 50 ********
1135 EZESPMUL002 0.7
******** RECORD NUMBER 1135 REPLACES RECORD NUMBER 51 ********
1136 EZESPMUL003 0.7
******** RECORD NUMBER 1136 REPLACES RECORD NUMBER 52 ********
*
* REFPNT - Defines reference time point for actions in evacuation ans sheltering zone.
1137 EZREFPNT001 ALARM
******** RECORD NUMBER 1137 REPLACES RECORD NUMBER 53 ********
*
* DURBEG - duration of initial phase (beginning) of evacuation, in seconds.
1138 EZDURBEG001 36000.
******** RECORD NUMBER 1138 REPLACES RECORD NUMBER 54 ********
*
* DURMID - duration of middle phase of evacuation, in seconds.
1139 EZDURMID001 7200.
******** RECORD NUMBER 1139 REPLACES RECORD NUMBER 55 ********
*
* NUMEVA - number of radial spatial elements (i.e. rings) of the sheltering and evacuation region.
1140 EZNUMEVA001 18
******** RECORD NUMBER 1140 REPLACES RECORD NUMBER 56 ********
*
* DLTSHL - delay from reference time point to when individual takes shelter. DLTEVA - delay elapsing between beginning of shelter period to when individuals begin evacuation.
1141 EZDLTSHL001 86400.
******** RECORD NUMBER 1141 REPLACES RECORD NUMBER 57 ********
1142 EZDLTSHL002 86400.
******** RECORD NUMBER 1142 REPLACES RECORD NUMBER 58 ********
1143 EZDLTSHL003 86400.
******** RECORD NUMBER 1143 REPLACES RECORD NUMBER 59 ********
1144 EZDLTSHL004 86400.
******** RECORD NUMBER 1144 REPLACES RECORD NUMBER 60 ********
1145 EZDLTSHL005 86400.
******** RECORD NUMBER 1145 REPLACES RECORD NUMBER 61 ********
1146 EZDLTSHL006 86400.
******** RECORD NUMBER 1146 REPLACES RECORD NUMBER 62 ********
1147 EZDLTSHL007 86400.
******** RECORD NUMBER 1147 REPLACES RECORD NUMBER 63 ********
1148 EZDLTSHL008 86400.
******** RECORD NUMBER 1148 REPLACES RECORD NUMBER 64 ********
1149 EZDLTSHL009 86400.
******** RECORD NUMBER 1149 REPLACES RECORD NUMBER 65 ********
1150 EZDLTSHL010 86400.
******** RECORD NUMBER 1150 REPLACES RECORD NUMBER 66 ********
1151 EZDLTSHL011 86400.
******** RECORD NUMBER 1151 REPLACES RECORD NUMBER 67 ********
1152 EZDLTSHL012 86400.
******** RECORD NUMBER 1152 REPLACES RECORD NUMBER 68 ********
1153 EZDLTSHL013 86400.
******** RECORD NUMBER 1153 REPLACES RECORD NUMBER 69 ********
1154 EZDLTSHL014 86400.
******** RECORD NUMBER 1154 REPLACES RECORD NUMBER 70 ********
1155 EZDLTSHL015 86400.
******** RECORD NUMBER 1155 REPLACES RECORD NUMBER 71 ********
1156 EZDLTSHL016 86400.
******** RECORD NUMBER 1156 REPLACES RECORD NUMBER 72 ********
1157 EZDLTSHL017 86400.
******** RECORD NUMBER 1157 REPLACES RECORD NUMBER 73 ********
1158 EZDLTSHL018 86400.
******** RECORD NUMBER 1158 REPLACES RECORD NUMBER 74 ********
*
* DLTEVA -Delay time to begin evacuation
1159 EZDLTEVA001 57600.
******** RECORD NUMBER 1159 REPLACES RECORD NUMBER 75 ********
1160 EZDLTEVA002 57600.
******** RECORD NUMBER 1160 REPLACES RECORD NUMBER 76 ********
1161 EZDLTEVA003 57600.
******** RECORD NUMBER 1161 REPLACES RECORD NUMBER 77 ********
1162 EZDLTEVA004 57600.
******** RECORD NUMBER 1162 REPLACES RECORD NUMBER 78 ********
1163 EZDLTEVA005 57600.
******** RECORD NUMBER 1163 REPLACES RECORD NUMBER 79 ********
1164 EZDLTEVA006 57600.
******** RECORD NUMBER 1164 REPLACES RECORD NUMBER 80 ********
1165 EZDLTEVA007 57600.

******** RECORD NUMBER 1165 REPLACES RECORD NUMBER 81 ********
1166 EZDLTEVA008 57600.
******** RECORD NUMBER 1166 REPLACES RECORD NUMBER 82 ********
1167 EZDLTEVA009 57600.
******** RECORD NUMBER 1167 REPLACES RECORD NUMBER 83 ********
1168 EZDLTEVA010 57600.
******** RECORD NUMBER 1168 REPLACES RECORD NUMBER 84 ********
1169 EZDLTEVA011 57600.
******** RECORD NUMBER 1169 REPLACES RECORD NUMBER 85 ********
1170 EZDLTEVA012 57600.
******** RECORD NUMBER 1170 REPLACES RECORD NUMBER 86 ********
1171 EZDLTEVA013 57600.
******** RECORD NUMBER 1171 REPLACES RECORD NUMBER 87 ********
1172 EZDLTEVA014 57600.
******** RECORD NUMBER 1172 REPLACES RECORD NUMBER 88 ********
1173 EZDLTEVA015 57600.
******** RECORD NUMBER 1173 REPLACES RECORD NUMBER 89 ********
1174 EZDLTEVA016 57600.
******** RECORD NUMBER 1174 REPLACES RECORD NUMBER 90 ********
1175 EZDLTEVA017 57600.
******** RECORD NUMBER 1175 REPLACES RECORD NUMBER 91 ********
1176 EZDLTEVA018 57600.
******** RECORD NUMBER 1176 REPLACES RECORD NUMBER 92 ********
*
* ESPGRD_NET - Evacuation speed multiplier for network evacuation
1177 EZESPGRD001 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1177 REPLACES RECORD NUMBER 93 ********
1178 EZESPGRD002 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1178 REPLACES RECORD NUMBER 94 ********
1179 EZESPGRD003 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1179 REPLACES RECORD NUMBER 95 ********
1180 EZESPGRD004 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1180 REPLACES RECORD NUMBER 96 ********
1181 EZESPGRD005 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1181 REPLACES RECORD NUMBER 97 ********
1182 EZESPGRD006 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1182 REPLACES RECORD NUMBER 98 ********
1183 EZESPGRD007 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1183 REPLACES RECORD NUMBER 99 ********
1184 EZESPGRD008 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1184 REPLACES RECORD NUMBER 100 ********
1185 EZESPGRD009 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1185 REPLACES RECORD NUMBER 101 ********
1186 EZESPGRD010 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1186 REPLACES RECORD NUMBER 102 ********
1187 EZESPGRD011 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1187 REPLACES RECORD NUMBER 103 ********
1188 EZESPGRD012 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1188 REPLACES RECORD NUMBER 104 ********
1189 EZESPGRD013 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1189 REPLACES RECORD NUMBER 105 ********
1190 EZESPGRD014 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1190 REPLACES RECORD NUMBER 106 ********
1191 EZESPGRD015 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.5 1.5 1.5 1. 1. 1. 1. 1. 1.5 1.5 1.5 1.5 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 0.75 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1191 REPLACES RECORD NUMBER 107 ********
1192 EZESPGRD016 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1192 REPLACES RECORD NUMBER 108 ********
1193 EZESPGRD017 0.75 0.75 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 0.75 0.75 0.75
******** RECORD NUMBER 1193 REPLACES RECORD NUMBER 109 ********
1194 EZESPGRD018 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 1194 REPLACES RECORD NUMBER 110 ********
1195 EZESPGRD019 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1.
1. 1. 1. 1. 1. 1. 1.
******** RECORD NUMBER 1195 REPLACES RECORD NUMBER 111 ********
*
* IDIREC - destination direction of every spatial element in the evacuation and sheltering region
1196 EZIDIREC001 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1196 REPLACES RECORD NUMBER 112 ********
1197 EZIDIREC002 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1197 REPLACES RECORD NUMBER 113 ********
1198 EZIDIREC003 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 1 1 1 1 1 1 1 1 2 1 4 4 4 4 2 2 2 2 2 1 1 1 4 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 4 1 1
******** RECORD NUMBER 1198 REPLACES RECORD NUMBER 114 ********
1199 EZIDIREC004 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 2 2 1 1 1 1 4 4 4 2 2 1 4 4 2 1 4 4 2 1 1 4 2 1
4 4 4 4 4 1 1
******** RECORD NUMBER 1199 REPLACES RECORD NUMBER 115 ********
1200 EZIDIREC005 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 2 1 4 4 2 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1200 REPLACES RECORD NUMBER 116 ********
1201 EZIDIREC006 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 1 1 1 1 2 1 1 4 2 2 1 2 1 4 4 2 2 1 4 4 4 4 4
4 4 1 1 1 1 1
******** RECORD NUMBER 1201 REPLACES RECORD NUMBER 117 ********
1202 EZIDIREC007 2 2 1 2 2 1 2 2 1 4 2 1 4 2 2 1 1 1 1 1 1 1 1 1 1 1 2 2 1 4 4 2 1 1 1 1 4 4 2 2 1 1 4 4 1 1 4 1 4 4 4 1 4 4 4 4 4
4 1 1 1 1 1 1
******** RECORD NUMBER 1202 REPLACES RECORD NUMBER 118 ********
1203 EZIDIREC008 1 4 1 1 4 2 1 4 2 1 1 4 2 2 1 1 2 1 1 1 4 4 4 1 1 1 1 1 1 4 1 4 2 1 4 4 2 1 2 1 2 1 4 2 1 4 2 1 1 4 4 1 4 4 4 2 2
1 1 1 1 1 2 2
******** RECORD NUMBER 1203 REPLACES RECORD NUMBER 119 ********
1204 EZIDIREC009 1 1 4 2 1 1 2 1 1 4 1 4 1 4 1 1 1 2 1 1 1 4 4 4 1 1 4 1 2 2 2 1 1 2 1 4 2 1 2 1 2 1 1 4 1 4 1 4 4 4 4 1 4 2 1 4 1
4 1 1 1 2 2 2

```
******** RECORD NUMBER 1204 REPLACES RECORD NUMBER 120 ********
 1205 EZIDIREC010 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 2 1 2 1 4 4 1 1 1 4 2 1 4 4 2 2 1 1 1 1 1 4
4 4 2 2 1 1 1
******** RECORD NUMBER 1205 REPLACES RECORD NUMBER 121 ********
 1206 EZIDIREC011 1 1 4 2 1 4 2 1 4 4 4 4 2 2 1 4 4 2 2 2 2 2 2 1 4 4 1 1 4 2 2 1 1 1 4 2 1 4 1 4 2 1 4 1 1 2 1 1 4 4 4 2 2 2 1 4 4
4 2 2 1 4 2 2
******** RECORD NUMBER 1206 REPLACES RECORD NUMBER 122 ********
 1207 EZIDIREC012 2 1 1 4 1 1 4 1 4 4 2 1 4 2 1 1 4 4 2 2 2 1 2 1 2 1 2 2 1 4 4 4 4 2 2 2 1 2 1 1 1 1 4 1 1 1 1 4 4 4 2 2 2 2 1 4 2
1 1 2 2 1 4 2
******** RECORD NUMBER 1207 REPLACES RECORD NUMBER 123 ********
 1208 EZIDIREC013 1 1 4 1 4 2 1 1 2 1 4 4 2 2 1 1 1 1 1 1 1 2 1 4 4 4 1 2 1 4 4 4 4 2 2 1 4 4 1 1 4 2 1 1 2 1 2 1 4 4 2 2 2 1 1 4 2
1 4 2 1 4 1 1
******** RECORD NUMBER 1208 REPLACES RECORD NUMBER 124 ********
 1209 EZIDIREC014 1 1 4 1 1 1 2 1 2 1 2 1 1 2 1 1 1 1 1 1 1 4 1 1 1 1 2 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 2 1 1 4 4 2 2 1 1 4 1
4 1 1 1 1 1 1
******** RECORD NUMBER 1209 REPLACES RECORD NUMBER 125 ********
 1210 EZIDIREC015 1 1 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 4 4 2 2 2 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 2 1 1 1 4 2 1 2 1 1 2 1
1 2 2 2 1 2 1
******** RECORD NUMBER 1210 REPLACES RECORD NUMBER 126 ********
 1211 EZIDIREC016 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1211 REPLACES RECORD NUMBER 127 ********
 1212 EZIDIREC017 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1212 REPLACES RECORD NUMBER 128 ********
 1213 EZIDIREC018 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1213 REPLACES RECORD NUMBER 129 ********
 1214 EZIDIREC019 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 1 1 1 1 1 1
******** RECORD NUMBER 1214 REPLACES RECORD NUMBER 130 ********
    *
    * LASMOV - The outermost spatial interval of the evacuation movement zone.
 1215 EZLASMOV001 19
******** RECORD NUMBER 1215 REPLACES RECORD NUMBER 131 ********
    *
    * EFFACY, KI Ingestion
 1216 EZEFFACY001 0.7
******** RECORD NUMBER 1216 REPLACES RECORD NUMBER 269 ********
    *
    * POPFRAC, KI Ingestion
 1217 EZPOPFRC001 0.
******** RECORD NUMBER 1217 REPLACES RECORD NUMBER 270 ********
    .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE 9 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE 9
NUMBER OF RECORDS CHANGED = 105
NUMBER OF RECORDS ADDED = 0
*****************************************************************************************************



With 1=forwards, 2=rightwards, 3=backwards, and 4=leftwards,
The Evacuation Network For This Scenario Was Defined As Follows:


IRAD    1  2  3  4  5  6  7  8  9 10 11 12 13 14 15 16
  1     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3     1  1  1  1  1  1  1  1  1  1  1  1  1  2  1  1
  4     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  5     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4
  6     1  1  1  1  1  1  1  1  1  1  1  1  1  1  4  1
  7     2  2  1  2  2  1  2  2  1  4  2  1  4  2  2  1
  8     1  4  1  1  4  2  1  4  2  1  1  4  2  2  1  1
  9     1  1  4  2  1  1  2  1  1  4  1  4  1  4  1  1
 10     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 11     1  1  4  2  1  4  2  1  4  4  4  4  2  2  1  4
 12     2  1  1  4  1  1  4  1  4  4  2  1  4  2  1  1
 13     1  1  4  1  4  2  1  1  2  1  4  4  2  2  1  1
 14     1  1  4  1  1  1  2  1  2  1  2  1  1  2  1  1
 15     1  1  4  2  2  2  1  1  1  1  1  1  1  1  1  1
 16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  4
 17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD   17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32
  1     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3     1  1  1  1  1  1  1  1  2  1  4  4  4  4  2  2
  4     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  5     1  1  1  1  1  1  1  1  1  2  2  2  2  2  2  1
  6     1  1  1  1  1  1  1  1  1  1  2  2  2  2  2  2
  7     1  1  1  1  1  1  1  1  1  1  2  2  1  4  4  2
  8     2  1  1  1  4  4  4  1  1  1  1  1  1  4  1  4
  9     1  2  1  1  1  4  4  4  1  1  4  1  2  2  2  1
 10     1  1  1  1  1  1  1  1  1  1  1  1  1  4  4  4
 11     4  2  2  2  2  2  2  1  4  4  1  1  4  2  2  1
 12     4  4  2  2  2  1  2  1  2  1  2  2  1  4  4  4
 13     1  1  1  1  1  2  1  4  4  4  1  2  1  4  4  4
 14     1  1  1  1  1  4  1  1  1  1  2  1  4  1  1  1
 15     1  1  1  1  1  1  1  4  4  4  2  2  2  1  1  1
 16     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 17     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 18     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
 19     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1

IRAD   33 34 35 36 37 38 39 40 41 42 43 44 45 46 47 48
  1     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  2     1  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  3     2  2  2  1  1  1  4  4  4  4  2  2  2  1  1  1
  4     4  2  2  1  1  1  1  4  4  4  2  2  1  4  4  2
  5     1  1  1  1  2  1  4  4  2  2  1  4  4  2  2  1
  6     2  2  1  1  1  1  2  1  1  4  2  2  1  2  1  4
  7     1  1  1  1  4  4  2  2  1  1  4  4  1  1  4  1
  8     2  1  4  4  2  1  2  1  2  1  4  2  1  4  2  1
  9     1  2  1  4  2  1  2  1  2  1  1  4  1  4  1  4
```

10	2	2	2	2	1	2	1	4	4	1	1	1	4	2	1	4
11	1	1	4	2	1	4	1	4	2	1	4	1	1	2	1	1
12	4	2	2	2	1	2	1	1	1	1	4	1	1	1	1	4
13	4	2	2	1	4	4	1	1	4	2	1	1	2	1	2	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	2	1
15	1	1	1	1	1	1	1	1	1	1	1	4	1	2	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

IRAD	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
3	1	1	1	1	1	1	1	1	1	1	1	1	1	4	1	1
4	1	4	4	2	1	1	4	2	1	4	4	4	4	4	1	1
5	4	4	2	2	1	4	4	2	1	1	1	1	1	1	1	1
6	4	2	2	1	4	4	4	4	4	4	4	1	1	1	1	1
7	4	4	4	1	4	4	4	4	4	4	1	1	1	1	1	1
8	1	4	4	1	4	4	4	2	2	1	1	1	1	1	2	2
9	4	4	4	1	4	2	1	4	1	4	1	1	1	2	2	2
10	4	2	2	1	1	1	1	1	4	4	4	2	2	1	1	1
11	4	4	4	2	2	2	1	4	4	4	2	2	1	4	2	2
12	4	4	2	2	2	2	1	4	2	1	1	2	2	1	4	2
13	4	4	2	2	2	1	1	4	2	1	4	2	1	4	1	1
14	1	4	4	2	2	1	1	4	1	4	1	1	1	1	1	1
15	1	4	2	1	2	1	1	2	1	1	2	2	2	1	2	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

```
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE  10 USER INPUT ********
     *
     * CSFACT - Cloudshine shielding factor
 1218  SECSFACT001     1.
********   RECORD NUMBER  1218 REPLACES RECORD NUMBER    25  ********
 1219  SECSFACT002     0.5
********   RECORD NUMBER  1219 REPLACES RECORD NUMBER    26  ********
 1220  SECSFACT003     0.5
********   RECORD NUMBER  1220 REPLACES RECORD NUMBER    27  ********
     *
     * PROTIN - Inhalation protection factor
 1221  SEPROTIN001     0.98
********   RECORD NUMBER  1221 REPLACES RECORD NUMBER    28  ********
 1222  SEPROTIN002     0.33
********   RECORD NUMBER  1222 REPLACES RECORD NUMBER    29  ********
 1223  SEPROTIN003     0.33
********   RECORD NUMBER  1223 REPLACES RECORD NUMBER    30  ********
     *
     * BRRATE - Breathing rates
 1224  SEBRRATE001     2.66E-04
********   RECORD NUMBER  1224 REPLACES RECORD NUMBER    31  ********
 1225  SEBRRATE002     2.66E-04
********   RECORD NUMBER  1225 REPLACES RECORD NUMBER    32  ********
 1226  SEBRRATE003     2.66E-04
********   RECORD NUMBER  1226 REPLACES RECORD NUMBER    33  ********
     *
     * SKPFAC - skin protection factors
 1227  SESKPFAC001     0.98
********   RECORD NUMBER  1227 REPLACES RECORD NUMBER    34  ********
 1228  SESKPFAC002     0.33
********   RECORD NUMBER  1228 REPLACES RECORD NUMBER    35  ********
 1229  SESKPFAC003     0.33
********   RECORD NUMBER  1229 REPLACES RECORD NUMBER    36  ********
     *
     * GSHFAC - groundshine shielding factors
 1230  SEGSHFAC001     0.5
********   RECORD NUMBER  1230 REPLACES RECORD NUMBER    37  ********
 1231  SEGSHFAC002     0.1
********   RECORD NUMBER  1231 REPLACES RECORD NUMBER    38  ********
 1232  SEGSHFAC003     0.1
********   RECORD NUMBER  1232 REPLACES RECORD NUMBER    39  ********
     *
     * EANAM2 - Name of emergency response cohort
 1233  EZEANAM2001     '30-40 Shelter in Place'
********   RECORD NUMBER  1233 REPLACES RECORD NUMBER    42  ********
     *
     * WTFRAC - weighting fraction applied to results of emergency response cohort
 1234  EZWTFRAC001     0.
********   RECORD NUMBER  1234 REPLACES RECORD NUMBER    44  ********
     *
     * LASMOV2  (used for no evacuation), always 0
 1235  EZLASMOV001     0
********   RECORD NUMBER  1235 REPLACES RECORD NUMBER   131  ********
     *
     * EFFACY,  KI Ingestion
 1236  EZEFFACY001     0.7
********   RECORD NUMBER  1236 REPLACES RECORD NUMBER   269  ********
     *
     * POPFRAC, KI Ingestion
 1237  EZPOPFRC001     0.
********   RECORD NUMBER  1237 REPLACES RECORD NUMBER   270  ********
     .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE  10 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE  10
NUMBER OF RECORDS CHANGED          =   20
NUMBER OF RECORDS ADDED            =    0
****************************************************************************************************


NO EVACUATION REQUESTED
THE KI MODEL IS IN EFFECT


******** BEGINNING OF CHANGE CASE  11 USER INPUT ********
     *
```

```
      * CSFACT - Cloudshine shielding factor
 1238  SECSFACT001      1.
********   RECORD NUMBER  1238 REPLACES RECORD NUMBER    25   ********
 1239  SECSFACT002      0.6
********   RECORD NUMBER  1239 REPLACES RECORD NUMBER    26   ********
 1240  SECSFACT003      0.5
********   RECORD NUMBER  1240 REPLACES RECORD NUMBER    27   ********
      *
      * PROTIN - Inhalation protection factor
 1241  SEPROTIN001      0.98
********   RECORD NUMBER  1241 REPLACES RECORD NUMBER    28   ********
 1242  SEPROTIN002      0.46
********   RECORD NUMBER  1242 REPLACES RECORD NUMBER    29   ********
 1243  SEPROTIN003      0.33
********   RECORD NUMBER  1243 REPLACES RECORD NUMBER    30   ********
      *
      * BRRATE - Breathing rates
 1244  SEBRRATE001      2.66E-04
********   RECORD NUMBER  1244 REPLACES RECORD NUMBER    31   ********
 1245  SEBRRATE002      2.66E-04
********   RECORD NUMBER  1245 REPLACES RECORD NUMBER    32   ********
 1246  SEBRRATE003      2.66E-04
********   RECORD NUMBER  1246 REPLACES RECORD NUMBER    33   ********
      *
      * SKPFAC - skin protection factors
 1247  SESKPFAC001      0.98
********   RECORD NUMBER  1247 REPLACES RECORD NUMBER    34   ********
 1248  SESKPFAC002      0.46
********   RECORD NUMBER  1248 REPLACES RECORD NUMBER    35   ********
 1249  SESKPFAC003      0.33
********   RECORD NUMBER  1249 REPLACES RECORD NUMBER    36   ********
      *
      * GSHFAC - groundshine shielding factors
 1250  SEGSHFAC001      0.5
********   RECORD NUMBER  1250 REPLACES RECORD NUMBER    37   ********
 1251  SEGSHFAC002      0.18
********   RECORD NUMBER  1251 REPLACES RECORD NUMBER    38   ********
 1252  SEGSHFAC003      0.1
********   RECORD NUMBER  1252 REPLACES RECORD NUMBER    39   ********
      *
      * EANAM2 - Name of emergency response cohort
 1253  EZEANAM2001      Nonevacuees
********   RECORD NUMBER  1253 REPLACES RECORD NUMBER    42   ********
      *
      * WTFRAC - weighting fraction applied to results of emergency response cohort
 1254  EZWTFRAC001      0.005
********   RECORD NUMBER  1254 REPLACES RECORD NUMBER    44   ********
      *
      * LASMOV2  (used for no evacuation), always 0
 1255  EZLASMOV001      0
********   RECORD NUMBER  1255 REPLACES RECORD NUMBER   131   ********
      *
      * EFFACY,  KI Ingestion
 1256  EZEFFACY001      0.7
********   RECORD NUMBER  1256 REPLACES RECORD NUMBER   269   ********
      *
      * POPFRAC, KI Ingestion
 1257  EZPOPFRC001      0.
********   RECORD NUMBER  1257 REPLACES RECORD NUMBER   270   ********
      .
******** TERMINATOR RECORD ENCOUNTERED -- END OF CHANGE CASE  11 USER INPUT ********

USER INPUT PROCESSING SUMMARY - CHANGE CASE  11
NUMBER OF RECORDS CHANGED          =   20
NUMBER OF RECORDS ADDED            =    0
****************************************************************************************************


NO EVACUATION REQUESTED
THE KI MODEL IS IN EFFECT



******** WARNING -- THE FOLLOWING RECORDS WERE NEVER ACCESSED ********

      EZWTFRAC001      0.005
      LCDTHNUM001      1
      LCDTHANN001      1.E-04
      LCDTHLIF001      10000.
      STFRACLD001      1.0


USER INPUT IS READ FROM UNIT 26
RECORD IDENTIFIER FIELDS  11 CHARACTERS LONG ARE EXPECTED.
THE FIRST 499 COLUMNS OF EACH INPUT RECORD ARE PROCESSED.

RECORD
NUMBER                              RECORD

      * File created using WinMACCS version 3.7.0 11/13/2012 10:59:18 AM
      *
      * CHNAME - description
   1  CHCHNAME001      'OCP3 high density no spray, EARLY input'
      *
      * EVACST - daily cost
   2  CHEVACST001      172.
      *
      * RELCST - daily cost due to intermediate
   3  CHRELCST001      172.
      *
      * DUR_INTPHAS, intermediate-phase period
   4  DUR_INTPHAS      0.E+00
      *
      * TMPACT - long term dose period
   5  CHTMPACT001      3.16E+07
      *
      * Form 'Long Term Dose Criterion' Comment:
      * Value of DSCRLT (0.005) from Pennsylvania Bureau of Radiation Protection.
      *
      * DSCRTI - dose criterion for phase
```

```
 6  CHDSCRTI001      1.00000E+05
   *
   * DSCRLT - dose criterion for habitation
 7  CHDSCRLT001      .005
   *
   * EXPTIM - long term exposure period
 8  CHEXPTIM001      1.58E+09
   *
   * CRTOCR - critical organ
 9  CHCRTOCR001      L-ICRP60ED
   *
   * Form 'Number of Plan Levels' Comment:
   * From NUREG-1150.
   *
   * LVLDEC - number of decontamination levels
10  CHLVLDEC001      2
   *
   * TIMDEC - time for each level
11  CHTIMDEC001      3.15E+07
12  CHTIMDEC002      3.15E+07
   *
   * DSRFCT - effectiveness of decontamination
13  CHDSRFCT001      3.
14  CHDSRFCT002      15.
   *
   * CDFRM - farmland decontamination cost
15  CHCDFRM0001      1330.
16  CHCDFRM0002      2960.
   *
   * CDNFRM - nonfarmland decontamination cost
17  CHCDNFRM001 7110.
18  CHCDNFRM002      19000.
   *
   * FRFDL - fraction farmland cost due labor
19  CHFRFDL0001      .3
20  CHFRFDL0002      .35
   *
   * FRNFDL - fraction nonfarmland cost due labor
21  CHFRNFDL001      .7
22  CHFRNFDL002      .5
   *
   * TFWKF - fraction time farmland worker
23  CHTFWKF0001      0.1
24  CHTFWKF0002      0.33
   *
   * TFWKNF - fraction time nonfarmland worker
25  CHTFWKNF001      0.33
26  CHTFWKNF002      0.33
   *
   * DLBCST - labor cost decontamination worker
27  CHDLBCST001      84000.
   *
   * DPRATE - depreciation rate applies to improvements
28  CHDPRATE001      .2
   *
   * DSRATE - rate of return
29  CHDSRATE001      .12
   *
   * POPCST - Per capita removal cost
30  CHPOPCST001      12000.
   *
   * NGWTRM - number weathering terms
31  CHNGWTRM001      2
   *
   * GWCOEF - groundshine coefficient
32  CHGWCOEF001      0.5
33  CHGWCOEF002      0.5
   *
   * TGWHLF - groundshine half lives
34  CHTGWHLF001      1.6E7
35  CHTGWHLF002      2.8E9
   *
   * NRWTRM - number resuspension terms
36  CHNRWTRM001      3
   *
   * RWCOEF - resuspension coefficient
37  CHRWCOEF001      1.0E-5
38  CHRWCOEF002      1.0E-7
39  CHRWCOEF003      1.0E-9
   *
   * TRWHLF - resuspension half lives
40  CHTRWHLF001      1.6E7
41  CHTRWHLF002      1.6E8
42  CHTRWHLF003      1.6E9
   *
   * VALWF - value of farm wealth
43  CHVALWF0001      9040.
   *
   * FRFIM - fraction of farm wealth due improvements
44  CHFRFIM0001      .25
   *
   * VALWNF - value of nonfarm wealth
45  CHVALWNF001      2.10000E+05
   *
   * FRNFIM - fraction nonfarm wealth due improvements
46  CHFRNFIM001      .8
   *
   * FDPATH, value = OLD, NEW or OFF to use models MACCS food, Comida2 or no food model respectively
47  CHFDPATH001      NEW
   *
   * DOSEMILK
48  DOSEMILK001      0.025
49  DOSEMILK002      0.075
   *
   * DOSEOTHR
50  DOSEOTHR001      0.025
51  DOSEOTHR002      0.075
   *
   * DOSELONG
52  DOSELONG001      0.005
```

```
  53  DOSELONG002      0.015
      *
      * Form 'Water Ingestion Radionuclides' Comment:
      *
      *
      * NUMWPI - size of array NAMWPI
  54  CHNUMWPI001      4
      *
      * popflg=FILE,NAMWPI, WSHFRI, WSHRTA, WINGF - water ingestion data
  55  CHWTRISO001      Sr-89   0.01    0.004   0.
  56  CHWTRISO002      Sr-90   0.01    0.004   0.
  57  CHWTRISO003      Cs-134  0.005   0.001   0.
  58  CHWTRISO004      Cs-137  0.005   0.001   0.
      *
      * KSWTCH - chronc output diagnostic switch
  59  CHKSWTCH001      0
      *
      * FRCFRM_FILE - popflg = FILE, dummy variable
  60  CHFRCFRM001      1.0
      *
      * FRMPRD_FILE - popflg=FILE, dummy variable
  61  CHFRMPRD001      0.0
      *
      * DPFRCT_FILE - popflg=FILE, dummy variable
  62  CHDPFRCT001      0.0
      *
      * Form 'Shielding and Exposure' Comment:
      * Data are taken directly from NUREG-1150 for normal activity.
      *
      * LPROTIN - Inhalation protection factor used in CHRONC
  63  CHLPROTIN01      .46
      *
      * LBRRATE - Breathing rate used in CHRONC
  64  CHLBRRATE01      2.66E-04
      *
      * LGSHFAC - groundshine shielding factor used in CHRONC
  65  CHLGSHFAC01      .18
      *
      * NXUM9=0
  66  TYPE9NUMBER      0
      *
      * NXUM9, number of type9 results
  67  TYPE9NUMBER      4
********   RECORD NUMBER    67 REPLACES RECORD NUMBER    66  ********
      *
      * ORGNAM7, IX1DS9, IX2DS9, CCDF9 - Population Dose
  68  TYPE9OUT001      L-ICRP60ED      1       12      NONE
  69  TYPE9OUT002      L-ICRP60ED      1       19      NONE
  70  TYPE9OUT003      L-ICRP60ED      1       21      NONE
  71  TYPE9OUT004      L-ICRP60ED      1       26      NONE
      *
      * NXUM10=0
  72  TYP10NUMBER      0
      *
      * NXUM10, number of type10 results
  73  TYP10NUMBER      10
********   RECORD NUMBER    73 REPLACES RECORD NUMBER    72  ********
      *
      *  I1DS10, I2DS10, CCDF10 - Economic Cost
  74  TYP10OUT001      1       26      NONE
  75  TYP10OUT002      1       12      NONE
  76  TYP10OUT003      13      15      NONE
  77  TYP10OUT004      16      17      NONE
  78  TYP10OUT005      18      18      NONE
  79  TYP10OUT006      19      19      NONE
  80  TYP10OUT007      20      21      NONE
  81  TYP10OUT008      22      23      NONE
  82  TYP10OUT009      24      25      NONE
  83  TYP10OUT010      26      26      NONE
      *
      * FLAG11 - Action Distance
  84  TYP11FLAG11      .TRUE.  NONE
      *
      * NUM12=0
  85  TYP12NUMBER      0
      *
      * NUM12, number of type 12 results
  86  TYP12NUMBER      10
********   RECORD NUMBER    86 REPLACES RECORD NUMBER    85  ********
      *
      * I1DS12, I2DS12, Impacted Area/Population
  87  TYP12OUT001      1       26      NONE
  88  TYP12OUT002      1       12      NONE
  89  TYP12OUT003      13      15      NONE
  90  TYP12OUT004      16      17      NONE
  91  TYP12OUT005      18      18      NONE
  92  TYP12OUT006      19      19      NONE
  93  TYP12OUT007      20      21      NONE
  94  TYP12OUT008      22      23      NONE
  95  TYP12OUT009      24      25      NONE
  96  TYP12OUT010      26      26      NONE
      *
      * NUM13=0
  97  TYP13NUMBER      0
      *
      * NUM13, number of type 13 results
  98  TYP13NUMBER      18
********   RECORD NUMBER    98 REPLACES RECORD NUMBER    97  ********
      *
      * IRAD13, ORGN13, Max Individual Food Ingestion Dose at a Distance
  99  TYP13OUT001      12      EFFECTIVE       NONE
 100  TYP13OUT002      15      EFFECTIVE       NONE
 101  TYP13OUT003      17      EFFECTIVE       NONE
 102  TYP13OUT004      18      EFFECTIVE       NONE
 103  TYP13OUT005      19      EFFECTIVE       NONE
 104  TYP13OUT006      21      EFFECTIVE       NONE
 105  TYP13OUT007      23      EFFECTIVE       NONE
 106  TYP13OUT008      25      EFFECTIVE       NONE
 107  TYP13OUT009      26      EFFECTIVE       NONE
 108  TYP13OUT010      12      THYROID NONE
```

109 TYP13OUT011 15 THYROID NONE
110 TYP13OUT012 17 THYROID NONE
111 TYP13OUT013 18 THYROID NONE
112 TYP13OUT014 19 THYROID NONE
113 TYP13OUT015 21 THYROID NONE
114 TYP13OUT016 23 THYROID NONE
115 TYP13OUT017 25 THYROID NONE
116 TYP13OUT018 26 THYROID NONE
*
* COMIDA2_TH - use for premade comida2, dose AT or PL models
117 BIN_FILE001 'C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Data\samp_a_FGR13GyEquivDCF.bin'
.
******** TERMINATOR RECORD ENCOUNTERED -- END OF BASE CASE USER INPUT ********

USER INPUT PROCESSING SUMMARY - BASE CASE

NUMBER OF RECORDS READ = 249
NUMBER OF BLANK OR COMMENT RECORDS READ = 131
NUMBER OF TERMINATOR RECORDS = 1
NUMBER OF RECORDS PROCESSED = 117
NUMBER OF PROCESSED RECORDS DUPLICATED = 4
NUMBER OF PROCESSED RECORDS SORTED = 113
****

READING COMIDA2 FILE: C:\Program Files\WinMACCS\SPF Scoping Study\R4 (version 3.7.0)\Late 30-mile evac\3.4 HighDensity\Data\samp_a_FGR13GyEquivDCF.bin
COMIDA2 binary file header =
COMIDA2 20120302 19:05:30 Version 1.13.0.1, 06/20/07

COMIDA2 descriptive title =
FGR13DF 5/13/2008 12:23:56 Version 1.03, Gy-Equivalent DCFs

Internal Dose Coefficients derived from FGR 13, EPA 402-R-99-001

COMIDA2 LASTSTOR = 9

A SITE DATA FILE IS BEING USED FOR BOTH "EARLY" AND "CHRONC"

8 CANCER EFFECTS ARE DEFINED IN THE MODEL.

INDEX	CANCER EFFECT	ORGAN	ALPHA	BETA	CFRISK	CIRISK
1	LEUKEMIA	L-RED MARR	1.000E+00	0.000E+00	1.110E-02	1.130E-02
2	BONE	L-BONE SUR	1.000E+00	0.000E+00	1.900E-04	2.710E-04
3	BREAST	L-BREAST	1.000E+00	0.000E+00	5.060E-03	1.010E-02
4	LUNG	L-LUNGS	1.000E+00	0.000E+00	1.980E-02	2.080E-02
5	THYROID	L-THYROID	1.000E+00	0.000E+00	6.480E-04	6.480E-03
6	LIVER	L-LIVER	1.000E+00	0.000E+00	3.000E-03	3.160E-03
7	COLON	L-LOWER LI	1.000E+00	0.000E+00	2.080E-02	3.780E-02
8	RESIDUAL	L-BLAD WAL	1.000E+00	0.000E+00	4.930E-02	1.690E-01

TIME OF HOTSPOT RELOCATION IS 1.4400E+04.
TIME OF NORMAL RETURN IS 5.760E+04 AND THE EMERGENCY PHASE ENDS AT 6.048E+05.

GROUNDSHINE SHIELDING FACTOR = 0.180

RESUSPENSION PROTECTION FACTOR = 0.460

BREATHING RATE (CUBIC M/S) = 2.660E-04

DISPERSION MODEL FLAG IS 3

WINDROSE PROBABILITIES BY WIND DIRECTION AND MET BIN NUMBER

BIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
1	0.0169	0.0099	0.0042	0.0113	0.0042	0.0000	0.0028	0.0085	0.0042	0.0099	0.0099	0.0071	0.0042	0.0071	0.0113	0.0155
2	0.0167	0.0143	0.0119	0.0167	0.0048	0.0072	0.0143	0.0167	0.0095	0.0119	0.0048	0.0167	0.0095	0.0143	0.0263	0.0286
3	0.0000	0.0000	0.0000	0.0000	0.0000	0.0111	0.0111	0.0111	0.0000	0.0222	0.0111	0.0222	0.0222	0.0000	0.0000	0.0333
4	0.0172	0.0210	0.0134	0.0095	0.0115	0.0095	0.0095	0.0076	0.0057	0.0076	0.0076	0.0115	0.0115	0.0210	0.0191	0.0134
5	0.0124	0.0212	0.0106	0.0124	0.0106	0.0071	0.0088	0.0053	0.0106	0.0053	0.0088	0.0053	0.0124	0.0071	0.0124	0.0265
6	0.0040	0.0054	0.0027	0.0040	0.0081	0.0027	0.0027	0.0054	0.0108	0.0040	0.0081	0.0108	0.0135	0.0108	0.0135	0.0148
7	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0048	0.0000	0.0096	0.0048	0.0385	0.0721
8	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
9	0.0226	0.0288	0.0309	0.0041	0.0082	0.0370	0.0123	0.0062	0.0165	0.0103	0.0185	0.0144	0.0041	0.0082	0.0103	0.0103
10	0.0282	0.0301	0.0214	0.0107	0.0136	0.0107	0.0117	0.0136	0.0097	0.0136	0.0175	0.0253	0.0224	0.0146	0.0224	0.0301
11	0.0103	0.0129	0.0078	0.0091	0.0052	0.0039	0.0039	0.0052	0.0091	0.0220	0.0272	0.0298	0.0310	0.0233	0.0336	0.0401
12	0.0085	0.0113	0.0028	0.0056	0.0056	0.0000	0.0085	0.0056	0.0113	0.0085	0.0113	0.0085	0.0282	0.0169	0.0339	0.0565
13	0.0176	0.0118	0.0412	0.0216	0.0137	0.0235	0.0314	0.0098	0.0275	0.0314	0.0255	0.0235	0.0196	0.0216	0.0275	0.0255
14	0.0053	0.0040	0.0160	0.0053	0.0093	0.0187	0.0120	0.0267	0.0293	0.0573	0.0600	0.0773	0.0960	0.0547	0.0667	0.0560
15	0.0000	0.0073	0.0000	0.0000	0.0073	0.0000	0.0073	0.0000	0.0219	0.0584	0.0803	0.0657	0.1168	0.1387	0.0949	0.1022
16	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.2500	0.2500	0.0000	0.2500	0.0000	0.2500	0.0000
17	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
18	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
19	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
20	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
21	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
22	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
23	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
24	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
25	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
26	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
27	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
28	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
29	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
30	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
31	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
32	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
33	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
34	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
35	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
36	0.0182	0.0231	0.0126	0.0105	0.0105	0.0112	0.0161	0.0105	0.0070	0.0168	0.0161	0.0105	0.0098	0.0098	0.0105	0.0154
37	0.0146	0.0162	0.0135	0.0094	0.0088	0.0102	0.0107	0.0100	0.0116	0.0177	0.0195	0.0210	0.0235	0.0186	0.0243	0.0288
38	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
39	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
40	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
41	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000

WINDROSE PROBABILITIES BY WIND DIRECTION AND MET BIN NUMBER

BIN	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
1	0.0028	0.0127	0.0071	0.0056	0.0099	0.0099	0.0071	0.0056	0.0071	0.0155	0.0099	0.0268	0.0395	0.0353	0.0226	0.0381
2	0.0048	0.0215	0.0239	0.0382	0.0549	0.0430	0.0406	0.0597	0.0358	0.0501	0.0740	0.0883	0.0644	0.0191	0.0143	0.0048

3	0.0333	0.0222	0.0000	0.0222	0.0000	0.0000	0.0222	0.0222	0.0111	0.0111	0.0000	0.0222	0.0111	0.0111	0.0000	0.0333
4	0.0153	0.0115	0.0191	0.0115	0.0248	0.0153	0.0172	0.0210	0.0248	0.0095	0.0229	0.0191	0.0248	0.0286	0.0172	0.0286
5	0.0106	0.0159	0.0177	0.0212	0.0389	0.0336	0.0319	0.0177	0.0195	0.0478	0.0602	0.0549	0.0460	0.0319	0.0283	0.0478
6	0.0175	0.0229	0.0565	0.0350	0.0579	0.0485	0.0848	0.0713	0.0646	0.0808	0.0619	0.0390	0.0565	0.0296	0.0094	0.0108
7	0.0192	0.0240	0.0337	0.0529	0.0721	0.0769	0.0577	0.1154	0.1250	0.0721	0.0962	0.0337	0.0337	0.0144	0.0000	0.0000
8	0.0313	0.0000	0.0313	0.0313	0.0313	0.0000	0.0000	0.0000	0.0000	0.4688	0.3750	0.0000	0.0000	0.0000	0.0000	0.0000
9	0.0062	0.0062	0.0144	0.0123	0.0082	0.0041	0.0123	0.0185	0.0144	0.0165	0.0082	0.0041	0.0041	0.0144	0.0041	0.0350
10	0.0233	0.0194	0.0262	0.0214	0.0204	0.0126	0.0224	0.0204	0.0117	0.0214	0.0262	0.0097	0.0175	0.0087	0.0126	0.0233
11	0.0453	0.0505	0.0582	0.0440	0.0388	0.0440	0.0310	0.0246	0.0207	0.0401	0.0285	0.0310	0.0207	0.0142	0.0246	0.0155
12	0.0311	0.0226	0.0452	0.0254	0.0678	0.0706	0.0650	0.0537	0.0537	0.0452	0.0282	0.0113	0.0113	0.0198	0.0141	0.0254
13	0.0157	0.0157	0.0137	0.0216	0.0098	0.0157	0.0216	0.0137	0.0196	0.0157	0.0333	0.0020	0.0000	0.0020	0.0020	0.0314
14	0.0307	0.0307	0.0280	0.0173	0.0293	0.0107	0.0200	0.0147	0.0227	0.0133	0.0133	0.0120	0.0133	0.0013	0.0040	0.0253
15	0.0438	0.0438	0.0146	0.0146	0.0219	0.0073	0.0073	0.0292	0.0000	0.0146	0.0000	0.0073	0.0073	0.0000	0.0073	0.0073
16	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
17	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
18	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
19	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
20	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
21	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
22	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
23	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
24	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
25	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
26	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
27	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
28	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
29	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
30	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
31	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
32	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
33	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
34	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
35	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
36	0.0042	0.0084	0.0049	0.0091	0.0098	0.0070	0.0126	0.0077	0.0133	0.0189	0.0210	0.0147	0.0175	0.0161	0.0119	0.0266
37	0.0177	0.0201	0.0248	0.0215	0.0282	0.0234	0.0282	0.0263	0.0250	0.0318	0.0322	0.0236	0.0251	0.0172	0.0131	0.0249
38	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
39	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
40	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
41	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000

WINDROSE PROBABILITIES BY WIND DIRECTION AND MET BIN NUMBER

BIN	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48
1	0.0240	0.0480	0.0452	0.0494	0.0212	0.0184	0.0169	0.0085	0.0071	0.0198	0.0240	0.0184	0.0240	0.0198	0.0226	0.0282
2	0.0000	0.0191	0.0095	0.0095	0.0024	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
3	0.0000	0.0222	0.0000	0.0444	0.0333	0.0222	0.0333	0.0111	0.0000	0.0222	0.0333	0.0222	0.0444	0.0222	0.0444	0.0556
4	0.0191	0.0439	0.0401	0.0305	0.0286	0.0134	0.0153	0.0057	0.0057	0.0153	0.0095	0.0076	0.0134	0.0038	0.0134	0.0134
5	0.0265	0.0159	0.0319	0.0071	0.0071	0.0053	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
6	0.0108	0.0121	0.0242	0.0013	0.0013	0.0000	0.0013	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
7	0.0000	0.0048	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
8	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
9	0.0123	0.0082	0.0103	0.0226	0.0062	0.0041	0.0062	0.0062	0.0082	0.0123	0.0082	0.0041	0.0267	0.0185	0.0288	0.0247
10	0.0107	0.0107	0.0165	0.0058	0.0087	0.0078	0.0029	0.0019	0.0000	0.0049	0.0019	0.0019	0.0019	0.0010	0.0019	0.0078
11	0.0091	0.0039	0.0078	0.0065	0.0039	0.0013	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
12	0.0169	0.0113	0.0056	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
13	0.0000	0.0059	0.0020	0.0176	0.0020	0.0000	0.0020	0.0000	0.0000	0.0020	0.0078	0.0020	0.0137	0.0039	0.0157	0.0216
14	0.0013	0.0000	0.0000	0.0027	0.0013	0.0013	0.0000	0.0013	0.0000	0.0000	0.0013	0.0040	0.0000	0.0000	0.0013	0.0040
15	0.0073	0.0000	0.0000	0.0073	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
16	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
17	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
18	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
19	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
20	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
21	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
22	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
23	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
24	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
25	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
26	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
27	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
28	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
29	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
30	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
31	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
32	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
33	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
34	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
35	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
36	0.0175	0.0392	0.0196	0.0126	0.0147	0.0084	0.0098	0.0098	0.0056	0.0105	0.0140	0.0098	0.0091	0.0077	0.0154	0.0224
37	0.0122	0.0191	0.0174	0.0132	0.0088	0.0056	0.0051	0.0034	0.0023	0.0058	0.0064	0.0047	0.0072	0.0047	0.0084	0.0112
38	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
39	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
40	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
41	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000

WINDROSE PROBABILITIES BY WIND DIRECTION AND MET BIN NUMBER

BIN	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64
1	0.0113	0.0311	0.0212	0.0099	0.0198	0.0071	0.0169	0.0099	0.0127	0.0099	0.0099	0.0141	0.0085	0.0155	0.0085	0.0155
2	0.0000	0.0000	0.0000	0.0000	0.0000	0.0024	0.0024	0.0000	0.0000	0.0024	0.0000	0.0048	0.0048	0.0191	0.0358	0.0263
3	0.0000	0.0444	0.0222	0.0111	0.0111	0.0111	0.0333	0.0111	0.0000	0.0111	0.0222	0.0111	0.0222	0.0000	0.0111	0.0000
4	0.0172	0.0057	0.0019	0.0095	0.0095	0.0115	0.0172	0.0153	0.0115	0.0115	0.0172	0.0191	0.0191	0.0153	0.0134	0.0191
5	0.0000	0.0000	0.0000	0.0000	0.0000	0.0053	0.0088	0.0000	0.0159	0.0265	0.0212	0.0230	0.0283	0.0177	0.0283	0.0301
6	0.0000	0.0000	0.0000	0.0000	0.0000	0.0013	0.0013	0.0027	0.0000	0.0081	0.0013	0.0013	0.0094	0.0202	0.0148	0.0202
7	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0096	0.0048	0.0144	0.0096
8	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0313	0.0000	0.0000
9	0.0309	0.0267	0.0206	0.0185	0.0144	0.0165	0.0309	0.0226	0.0103	0.0165	0.0412	0.0226	0.0267	0.0267	0.0309	0.0103
10	0.0049	0.0175	0.0107	0.0107	0.0097	0.0117	0.0155	0.0214	0.0204	0.0204	0.0292	0.0253	0.0311	0.0282	0.0311	0.0330
11	0.0000	0.0026	0.0000	0.0000	0.0026	0.0026	0.0013	0.0013	0.0103	0.0181	0.0194	0.0194	0.0194	0.0246	0.0246	0.0155
12	0.0000	0.0000	0.0000	0.0000	0.0000	0.0028	0.0028	0.0000	0.0056	0.0028	0.0028	0.0085	0.0141	0.0141	0.0339	0.0650
13	0.0451	0.0353	0.0431	0.0059	0.0608	0.0118	0.0255	0.0039	0.0000	0.0078	0.0118	0.0235	0.0059	0.0059	0.0118	0.0000
14	0.0040	0.0133	0.0093	0.0067	0.0107	0.0053	0.0093	0.0040	0.0013	0.0093	0.0000	0.0053	0.0067	0.0053	0.0067	0.0027
15	0.0000	0.0000	0.0000	0.0000	0.0073	0.0000	0.0000	0.0000	0.0000	0.0146	0.0000	0.0000	0.0146	0.0073	0.0073	0.0073
16	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
17	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
18	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
19	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
20	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
21	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
22	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
23	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308
24	0.0196	0.0084	0.0105	0.0161	0.0217	0.0112	0.0154	0.0273	0.0259	0.0301	0.0266	0.0287	0.0280	0.0259	0.0350	0.0308

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25 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
26 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
27 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
28 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
29 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
30 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
31 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
32 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
33 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
34 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
35 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
36 0.0196 0.0084 0.0105 0.0161 0.0217 0.0112 0.0154 0.0273 0.0259 0.0301 0.0266 0.0287 0.0280 0.0259 0.0350 0.0308
37 0.0104 0.0116 0.0095 0.0073 0.0126 0.0075 0.0121 0.0110 0.0112 0.0155 0.0161 0.0170 0.0183 0.0188 0.0227 0.0213
38 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000
39 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000
40 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000
41 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000 0.0000
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Processing a Site Data File with Header: SECPOP2000 Version: 3.13.1 MACCS2 Formatted Site: File for Peach Bottom Census
Lat: 39d45'32" Long: 76d16' 9" Population multiplier: 1.0533 Economic multip

THIS PROGRAM CURRENTLY ALLOWS THE GENERATION OF UP TO 3394 RESULTS

YOU HAVE REQUESTED 108 RESULTS FROM "EARLY" COMPOSED OF:

```
38 RESULTS OF TYPE 1
 1 RESULTS OF TYPE 2
 3 RESULTS OF TYPE 3
 0 RESULTS OF TYPE 4
 4 RESULTS OF TYPE 5
 0 RESULTS OF TYPE 6
 0 RESULTS OF TYPE 7
17 RESULTS OF TYPE 8
26 RESULTS OF TYPE A
 0 RESULTS OF TYPE B
 3 RESULTS OF TYPE C
16 RESULTS OF TYPE D
 0 RESULTS OF TYPE E
```

YOU HAVE REQUESTED 304 RESULTS FROM "CHRONC" COMPOSED OF:

```
 68 RESULTS OF TYPE 9
130 RESULTS OF TYPE 10
  8 RESULTS OF TYPE 11
 80 RESULTS OF TYPE 12
 18 RESULTS OF TYPE 13
```

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   TRIAL     DAY    PERIOD     BIN    PRBMET
     1       151      21        9   1.13E-03
```

WARNING!!

THE TOTAL RELEASE DURATION EXCEEDS 2 HOURS.

THIS MAY CAUSE ERRONEOUS RESULTS TO BE PRODUCED

WHEN USING THE Regulatory Guide 1.145 model.

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For Julian Day 151, selecting COMIDA2 results # 4 of 9
     2       152       4        1   1.14E-03
For Julian Day 152, selecting COMIDA2 results # 4 of 9
     3       152       9       36   1.43E-04
For Julian Day 152, selecting COMIDA2 results # 4 of 9
     4       152      10       35   1.14E-04
For Julian Day 152, selecting COMIDA2 results # 4 of 9
     5       152      11       34   1.14E-04
For Julian Day 152, selecting COMIDA2 results # 4 of 9
     6       152      12       32   3.23E-04
For Julian Day 152, selecting COMIDA2 results # 4 of 9
     7       152      19       10   1.14E-03
For Julian Day 152, selecting COMIDA2 results # 4 of 9
     8       152      24       36   1.43E-04
For Julian Day 152, selecting COMIDA2 results # 4 of 9
     9       153       1       36   1.43E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
    10       153       3       36   1.43E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
    11       153       4       35   1.14E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
    12       153       5       35   1.14E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
    13       153       6       34   1.14E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
    14       153       7       34   1.14E-04
For Julian Day 153, selecting COMIDA2 results # 4 of 9
    15       154       6        6   1.15E-03
For Julian Day 154, selecting COMIDA2 results # 4 of 9
    16       154      17       10   1.14E-03
For Julian Day 154, selecting COMIDA2 results # 4 of 9
    17       154      18       11   1.15E-03
For Julian Day 154, selecting COMIDA2 results # 4 of 9
    18       155       4        4   1.15E-03
For Julian Day 155, selecting COMIDA2 results # 4 of 9
    19       155      17       11   1.15E-03
For Julian Day 155, selecting COMIDA2 results # 4 of 9
    20       156       1        5   1.13E-03
For Julian Day 156, selecting COMIDA2 results # 4 of 9
    21       156      13       10   1.14E-03
For Julian Day 156, selecting COMIDA2 results # 4 of 9
    22       156      18        9   1.13E-03
For Julian Day 156, selecting COMIDA2 results # 4 of 9
    23       157       6        1   1.14E-03
For Julian Day 157, selecting COMIDA2 results # 4 of 9
    24       157      12        3   8.56E-04
For Julian Day 157, selecting COMIDA2 results # 4 of 9
    25       158       4       21   1.13E-03
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For Julian Day 158, selecting COMIDA2 results # 4 of 9
26 158 8 25 1.52E-04
For Julian Day 158, selecting COMIDA2 results # 4 of 9
27 158 9 24 1.14E-04
For Julian Day 158, selecting COMIDA2 results # 4 of 9
28 158 13 12 1.15E-03
For Julian Day 158, selecting COMIDA2 results # 4 of 9
29 158 19 17 1.14E-03
For Julian Day 158, selecting COMIDA2 results # 4 of 9
30 158 22 14 1.14E-03
For Julian Day 158, selecting COMIDA2 results # 4 of 9
31 159 4 4 1.15E-03
For Julian Day 159, selecting COMIDA2 results # 4 of 9
32 159 11 26 2.38E-04
For Julian Day 159, selecting COMIDA2 results # 4 of 9
33 159 12 25 1.52E-04
For Julian Day 159, selecting COMIDA2 results # 4 of 9
34 159 13 24 1.14E-04
For Julian Day 159, selecting COMIDA2 results # 4 of 9
35 159 14 22 1.09E-03
For Julian Day 159, selecting COMIDA2 results # 4 of 9
36 159 18 18 5.99E-04
For Julian Day 159, selecting COMIDA2 results # 4 of 9
37 159 24 14 1.14E-03
For Julian Day 159, selecting COMIDA2 results # 4 of 9
38 160 4 5 1.13E-03
For Julian Day 160, selecting COMIDA2 results # 4 of 9
39 160 6 20 1.12E-03
For Julian Day 160, selecting COMIDA2 results # 4 of 9
40 160 7 19 1.11E-03
For Julian Day 160, selecting COMIDA2 results # 4 of 9
41 160 19 11 1.15E-03
For Julian Day 160, selecting COMIDA2 results # 4 of 9
42 161 6 2 1.14E-03
For Julian Day 161, selecting COMIDA2 results # 4 of 9
43 161 11 6 1.15E-03
For Julian Day 161, selecting COMIDA2 results # 4 of 9
44 161 14 11 1.15E-03
For Julian Day 161, selecting COMIDA2 results # 4 of 9
45 161 17 15 1.12E-03
For Julian Day 161, selecting COMIDA2 results # 4 of 9
46 163 1 10 1.14E-03
For Julian Day 163, selecting COMIDA2 results # 4 of 9
47 163 6 1 1.14E-03
For Julian Day 163, selecting COMIDA2 results # 4 of 9
48 163 21 14 1.14E-03
For Julian Day 163, selecting COMIDA2 results # 4 of 9
49 165 5 1 1.14E-03
For Julian Day 165, selecting COMIDA2 results # 4 of 9
50 165 6 1 1.14E-03
For Julian Day 165, selecting COMIDA2 results # 4 of 9

TRIAL DAY PERIOD BIN PRBMET
51 165 15 11 1.15E-03
For Julian Day 165, selecting COMIDA2 results # 4 of 9
52 165 21 11 1.15E-03
For Julian Day 165, selecting COMIDA2 results # 4 of 9
53 166 6 2 1.14E-03
For Julian Day 166, selecting COMIDA2 results # 4 of 9
54 167 10 4 1.15E-03
For Julian Day 167, selecting COMIDA2 results # 5 of 9
55 167 15 14 1.14E-03
For Julian Day 167, selecting COMIDA2 results # 5 of 9
56 167 18 13 1.14E-03
For Julian Day 167, selecting COMIDA2 results # 5 of 9
57 167 24 13 1.14E-03
For Julian Day 167, selecting COMIDA2 results # 5 of 9
58 169 3 4 1.15E-03
For Julian Day 169, selecting COMIDA2 results # 5 of 9
59 169 8 1 1.14E-03
For Julian Day 169, selecting COMIDA2 results # 5 of 9
60 169 12 6 1.15E-03
For Julian Day 169, selecting COMIDA2 results # 5 of 9
61 169 17 10 1.14E-03
For Julian Day 169, selecting COMIDA2 results # 5 of 9
62 170 7 26 2.38E-04
For Julian Day 170, selecting COMIDA2 results # 5 of 9
63 170 9 24 1.14E-04
For Julian Day 170, selecting COMIDA2 results # 5 of 9
64 170 13 10 1.14E-03
For Julian Day 170, selecting COMIDA2 results # 5 of 9
65 171 1 17 1.14E-03
For Julian Day 171, selecting COMIDA2 results # 5 of 9
66 171 7 5 1.13E-03
For Julian Day 171, selecting COMIDA2 results # 5 of 9
67 171 15 14 1.14E-03
For Julian Day 171, selecting COMIDA2 results # 5 of 9
68 172 1 9 1.13E-03
For Julian Day 172, selecting COMIDA2 results # 5 of 9
69 172 20 21 1.13E-03
For Julian Day 172, selecting COMIDA2 results # 5 of 9
70 173 3 10 1.14E-03
For Julian Day 173, selecting COMIDA2 results # 5 of 9
71 173 21 14 1.14E-03
For Julian Day 173, selecting COMIDA2 results # 5 of 9
72 174 2 4 1.15E-03
For Julian Day 174, selecting COMIDA2 results # 5 of 9
73 174 5 1 1.14E-03
For Julian Day 174, selecting COMIDA2 results # 5 of 9
74 174 6 5 1.13E-03
For Julian Day 174, selecting COMIDA2 results # 5 of 9
75 174 16 19 1.11E-03
For Julian Day 174, selecting COMIDA2 results # 5 of 9
76 175 3 36 1.43E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
77 175 4 36 1.43E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
78 175 5 35 1.14E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9

79 175 6 35 1.14E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
80 175 7 34 1.14E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
81 175 10 32 3.23E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
82 175 11 27 3.71E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
83 175 14 20 1.12E-03
For Julian Day 175, selecting COMIDA2 results # 5 of 9
84 175 18 17 1.14E-03
For Julian Day 175, selecting COMIDA2 results # 5 of 9
85 175 20 10 1.14E-03
For Julian Day 175, selecting COMIDA2 results # 5 of 9
86 175 22 26 2.38E-04
For Julian Day 175, selecting COMIDA2 results # 5 of 9
87 176 1 25 1.52E-04
For Julian Day 176, selecting COMIDA2 results # 5 of 9
88 176 3 24 1.14E-04
For Julian Day 176, selecting COMIDA2 results # 5 of 9
89 176 16 27 3.71E-04
For Julian Day 176, selecting COMIDA2 results # 5 of 9
90 176 20 32 3.23E-04
For Julian Day 176, selecting COMIDA2 results # 5 of 9
91 176 23 32 3.23E-04
For Julian Day 176, selecting COMIDA2 results # 5 of 9
92 177 2 17 1.14E-03
For Julian Day 177, selecting COMIDA2 results # 5 of 9
93 177 12 32 3.23E-04
For Julian Day 177, selecting COMIDA2 results # 5 of 9
94 177 15 25 1.52E-04
For Julian Day 177, selecting COMIDA2 results # 5 of 9
95 177 18 25 1.52E-04
For Julian Day 177, selecting COMIDA2 results # 5 of 9
96 177 21 32 3.23E-04
For Julian Day 177, selecting COMIDA2 results # 5 of 9
97 177 24 22 1.09E-03
For Julian Day 177, selecting COMIDA2 results # 5 of 9
98 178 1 17 1.14E-03
For Julian Day 178, selecting COMIDA2 results # 5 of 9
99 178 21 27 3.71E-04
For Julian Day 178, selecting COMIDA2 results # 5 of 9
100 179 10 1 1.14E-03
For Julian Day 179, selecting COMIDA2 results # 5 of 9

TRIAL DAY PERIOD BIN PRBMET
101 179 13 11 1.15E-03
For Julian Day 179, selecting COMIDA2 results # 5 of 9
102 179 16 20 1.12E-03
For Julian Day 179, selecting COMIDA2 results # 5 of 9
103 179 19 19 1.11E-03
For Julian Day 179, selecting COMIDA2 results # 5 of 9
104 180 7 4 1.15E-03
For Julian Day 180, selecting COMIDA2 results # 5 of 9
105 180 10 31 1.14E-04
For Julian Day 180, selecting COMIDA2 results # 5 of 9
106 180 11 30 1.14E-04
For Julian Day 180, selecting COMIDA2 results # 5 of 9
107 180 12 30 1.14E-04
For Julian Day 180, selecting COMIDA2 results # 5 of 9
108 180 13 29 1.14E-04
For Julian Day 180, selecting COMIDA2 results # 5 of 9
109 181 14 14 1.14E-03
For Julian Day 181, selecting COMIDA2 results # 5 of 9
110 181 21 13 1.14E-03
For Julian Day 181, selecting COMIDA2 results # 5 of 9
111 182 1 9 1.13E-03
For Julian Day 182, selecting COMIDA2 results # 5 of 9
112 182 11 5 1.13E-03
For Julian Day 182, selecting COMIDA2 results # 5 of 9
113 182 17 10 1.14E-03
For Julian Day 182, selecting COMIDA2 results # 5 of 9
114 182 21 10 1.14E-03
For Julian Day 182, selecting COMIDA2 results # 5 of 9
115 183 12 11 1.15E-03
For Julian Day 183, selecting COMIDA2 results # 5 of 9
116 185 12 9 1.13E-03
For Julian Day 185, selecting COMIDA2 results # 5 of 9
117 185 18 21 1.13E-03
For Julian Day 185, selecting COMIDA2 results # 5 of 9
118 185 22 17 1.14E-03
For Julian Day 185, selecting COMIDA2 results # 5 of 9
119 186 1 10 1.14E-03
For Julian Day 186, selecting COMIDA2 results # 5 of 9
120 186 5 5 1.13E-03
For Julian Day 186, selecting COMIDA2 results # 5 of 9
121 186 20 32 3.23E-04
For Julian Day 186, selecting COMIDA2 results # 5 of 9
122 186 22 17 1.14E-03
For Julian Day 186, selecting COMIDA2 results # 5 of 9
123 186 23 10 1.14E-03
For Julian Day 186, selecting COMIDA2 results # 5 of 9
124 188 11 4 1.15E-03
For Julian Day 188, selecting COMIDA2 results # 5 of 9
125 188 17 14 1.14E-03
For Julian Day 188, selecting COMIDA2 results # 5 of 9
126 189 8 4 1.15E-03
For Julian Day 189, selecting COMIDA2 results # 5 of 9
127 189 11 10 1.14E-03
For Julian Day 189, selecting COMIDA2 results # 5 of 9
128 189 15 14 1.14E-03
For Julian Day 189, selecting COMIDA2 results # 5 of 9
129 190 11 6 1.15E-03
For Julian Day 190, selecting COMIDA2 results # 5 of 9
130 190 20 9 1.13E-03
For Julian Day 190, selecting COMIDA2 results # 5 of 9
131 191 22 10 1.14E-03
For Julian Day 191, selecting COMIDA2 results # 5 of 9
132 191 24 9 1.13E-03

For Julian Day 191, selecting COMIDA2 results # 5 of 9
133 192 4 1 1.14E-03
For Julian Day 192, selecting COMIDA2 results # 6 of 9
134 192 9 5 1.13E-03
For Julian Day 192, selecting COMIDA2 results # 6 of 9
135 192 12 5 1.13E-03
For Julian Day 192, selecting COMIDA2 results # 6 of 9
136 193 12 26 2.38E-04
For Julian Day 193, selecting COMIDA2 results # 6 of 9
137 193 14 25 1.52E-04
For Julian Day 193, selecting COMIDA2 results # 6 of 9
138 193 15 24 1.14E-04
For Julian Day 193, selecting COMIDA2 results # 6 of 9
139 194 19 19 1.11E-03
For Julian Day 194, selecting COMIDA2 results # 6 of 9
140 194 24 13 1.14E-03
For Julian Day 194, selecting COMIDA2 results # 6 of 9
141 195 4 1 1.14E-03
For Julian Day 195, selecting COMIDA2 results # 6 of 9
142 195 19 10 1.14E-03
For Julian Day 195, selecting COMIDA2 results # 6 of 9
143 195 24 10 1.14E-03
For Julian Day 195, selecting COMIDA2 results # 6 of 9
144 196 1 36 1.43E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
145 196 3 36 1.43E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
146 196 4 35 1.14E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
147 196 5 35 1.14E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
148 196 6 34 1.14E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
149 196 9 18 5.99E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9
150 196 12 3 8.56E-04
For Julian Day 196, selecting COMIDA2 results # 6 of 9

TRIAL DAY PERIOD BIN PRBMET
151 196 20 14 1.14E-03
For Julian Day 196, selecting COMIDA2 results # 6 of 9
152 197 1 11 1.15E-03
For Julian Day 197, selecting COMIDA2 results # 6 of 9
153 197 9 5 1.13E-03
For Julian Day 197, selecting COMIDA2 results # 6 of 9
154 197 11 11 1.15E-03
For Julian Day 197, selecting COMIDA2 results # 6 of 9
155 197 24 14 1.14E-03
For Julian Day 197, selecting COMIDA2 results # 6 of 9
156 198 7 14 1.14E-03
For Julian Day 198, selecting COMIDA2 results # 6 of 9
157 199 1 9 1.13E-03
For Julian Day 199, selecting COMIDA2 results # 6 of 9
158 199 8 36 1.43E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
159 199 9 36 1.43E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
160 199 10 36 1.43E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
161 199 11 35 1.14E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
162 199 12 34 1.14E-04
For Julian Day 199, selecting COMIDA2 results # 6 of 9
163 199 18 20 1.12E-03
For Julian Day 199, selecting COMIDA2 results # 6 of 9
164 200 9 4 1.15E-03
For Julian Day 200, selecting COMIDA2 results # 6 of 9
165 201 2 4 1.15E-03
For Julian Day 201, selecting COMIDA2 results # 6 of 9
166 201 18 10 1.14E-03
For Julian Day 201, selecting COMIDA2 results # 6 of 9
167 201 20 9 1.13E-03
For Julian Day 201, selecting COMIDA2 results # 6 of 9
168 202 2 3 8.56E-04
For Julian Day 202, selecting COMIDA2 results # 6 of 9
169 202 12 10 1.14E-03
For Julian Day 202, selecting COMIDA2 results # 6 of 9
170 202 15 14 1.14E-03
For Julian Day 202, selecting COMIDA2 results # 6 of 9
171 203 5 5 1.13E-03
For Julian Day 203, selecting COMIDA2 results # 6 of 9
172 203 11 26 2.38E-04
For Julian Day 203, selecting COMIDA2 results # 6 of 9
173 203 13 22 1.09E-03
For Julian Day 203, selecting COMIDA2 results # 6 of 9
174 203 14 18 5.99E-04
For Julian Day 203, selecting COMIDA2 results # 6 of 9
175 203 22 10 1.14E-03
For Julian Day 203, selecting COMIDA2 results # 6 of 9
176 204 8 1 1.14E-03
For Julian Day 204, selecting COMIDA2 results # 6 of 9
177 204 11 4 1.15E-03
For Julian Day 204, selecting COMIDA2 results # 6 of 9
178 205 9 4 1.15E-03
For Julian Day 205, selecting COMIDA2 results # 6 of 9
179 205 23 10 1.14E-03
For Julian Day 205, selecting COMIDA2 results # 6 of 9
180 206 9 2 1.14E-03
For Julian Day 206, selecting COMIDA2 results # 6 of 9
181 206 14 11 1.15E-03
For Julian Day 206, selecting COMIDA2 results # 6 of 9
182 206 24 9 1.13E-03
For Julian Day 206, selecting COMIDA2 results # 6 of 9
183 207 6 1 1.14E-03
For Julian Day 207, selecting COMIDA2 results # 6 of 9
184 207 11 5 1.13E-03
For Julian Day 207, selecting COMIDA2 results # 6 of 9
185 208 9 6 1.15E-03
For Julian Day 208, selecting COMIDA2 results # 6 of 9

186 208 20 9 1.13E-03
For Julian Day 208, selecting COMIDA2 results # 6 of 9
187 209 5 26 2.38E-04
For Julian Day 209, selecting COMIDA2 results # 6 of 9
188 209 6 25 1.52E-04
For Julian Day 209, selecting COMIDA2 results # 6 of 9
189 209 17 14 1.14E-03
For Julian Day 209, selecting COMIDA2 results # 6 of 9
190 209 18 15 1.12E-03
For Julian Day 209, selecting COMIDA2 results # 6 of 9
191 209 22 14 1.14E-03
For Julian Day 209, selecting COMIDA2 results # 6 of 9
192 210 10 4 1.15E-03
For Julian Day 210, selecting COMIDA2 results # 6 of 9
193 210 24 10 1.14E-03
For Julian Day 210, selecting COMIDA2 results # 6 of 9
194 211 15 13 1.14E-03
For Julian Day 211, selecting COMIDA2 results # 6 of 9
195 211 16 14 1.14E-03
For Julian Day 211, selecting COMIDA2 results # 6 of 9
196 212 5 26 2.38E-04
For Julian Day 212, selecting COMIDA2 results # 6 of 9
197 212 6 25 1.52E-04
For Julian Day 212, selecting COMIDA2 results # 6 of 9
198 212 7 24 1.14E-04
For Julian Day 212, selecting COMIDA2 results # 6 of 9
199 212 8 23 1.14E-04
For Julian Day 212, selecting COMIDA2 results # 6 of 9
200 212 13 10 1.14E-03
For Julian Day 212, selecting COMIDA2 results # 6 of 9

TRIAL DAY PERIOD BIN PRBMET
201 212 14 9 1.13E-03
For Julian Day 212, selecting COMIDA2 results # 6 of 9
202 213 3 3 8.56E-04
For Julian Day 213, selecting COMIDA2 results # 6 of 9
203 213 7 5 1.13E-03
For Julian Day 213, selecting COMIDA2 results # 6 of 9
204 213 20 14 1.14E-03
For Julian Day 213, selecting COMIDA2 results # 6 of 9
205 213 21 14 1.14E-03
For Julian Day 213, selecting COMIDA2 results # 6 of 9
206 214 8 5 1.13E-03
For Julian Day 214, selecting COMIDA2 results # 6 of 9
207 215 12 11 1.15E-03
For Julian Day 215, selecting COMIDA2 results # 6 of 9
208 215 14 14 1.14E-03
For Julian Day 215, selecting COMIDA2 results # 6 of 9
209 215 15 13 1.14E-03
For Julian Day 215, selecting COMIDA2 results # 6 of 9
210 216 6 6 1.15E-03
For Julian Day 216, selecting COMIDA2 results # 6 of 9
211 216 20 14 1.14E-03
For Julian Day 216, selecting COMIDA2 results # 6 of 9
212 217 2 4 1.15E-03
For Julian Day 217, selecting COMIDA2 results # 6 of 9
213 217 5 1 1.14E-03
For Julian Day 217, selecting COMIDA2 results # 6 of 9
214 217 20 14 1.14E-03
For Julian Day 217, selecting COMIDA2 results # 6 of 9
215 217 24 10 1.14E-03
For Julian Day 217, selecting COMIDA2 results # 6 of 9
216 218 4 1 1.14E-03
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217 218 17 31 1.14E-04
For Julian Day 218, selecting COMIDA2 results # 6 of 9
218 218 18 30 1.14E-04
For Julian Day 218, selecting COMIDA2 results # 6 of 9
219 218 19 29 1.14E-04
For Julian Day 218, selecting COMIDA2 results # 6 of 9
220 218 20 28 1.14E-04
For Julian Day 218, selecting COMIDA2 results # 6 of 9
221 218 22 17 1.14E-03
For Julian Day 218, selecting COMIDA2 results # 6 of 9
222 219 8 1 1.14E-03
For Julian Day 219, selecting COMIDA2 results # 6 of 9
223 219 19 10 1.14E-03
For Julian Day 219, selecting COMIDA2 results # 6 of 9
224 219 23 12 1.15E-03
For Julian Day 219, selecting COMIDA2 results # 6 of 9
225 220 8 5 1.13E-03
For Julian Day 220, selecting COMIDA2 results # 6 of 9
226 220 16 11 1.15E-03
For Julian Day 220, selecting COMIDA2 results # 6 of 9
227 221 12 4 1.15E-03
For Julian Day 221, selecting COMIDA2 results # 6 of 9
228 221 15 14 1.14E-03
For Julian Day 221, selecting COMIDA2 results # 6 of 9
229 222 14 10 1.14E-03
For Julian Day 222, selecting COMIDA2 results # 7 of 9
230 222 17 9 1.13E-03
For Julian Day 222, selecting COMIDA2 results # 7 of 9
231 222 18 14 1.14E-03
For Julian Day 222, selecting COMIDA2 results # 7 of 9
232 222 23 12 1.15E-03
For Julian Day 222, selecting COMIDA2 results # 7 of 9
233 223 6 1 1.14E-03
For Julian Day 223, selecting COMIDA2 results # 7 of 9
234 224 18 14 1.14E-03
For Julian Day 224, selecting COMIDA2 results # 7 of 9
235 225 9 1 1.14E-03
For Julian Day 225, selecting COMIDA2 results # 7 of 9
236 225 13 10 1.14E-03
For Julian Day 225, selecting COMIDA2 results # 7 of 9
237 226 17 11 1.15E-03
For Julian Day 226, selecting COMIDA2 results # 7 of 9
238 227 2 10 1.14E-03
For Julian Day 227, selecting COMIDA2 results # 7 of 9
239 227 5 1 1.14E-03

For Julian Day 227, selecting COMIDA2 results # 7 of 9
240 227 19 14 1.14E-03
For Julian Day 227, selecting COMIDA2 results # 7 of 9
241 227 21 15 1.12E-03
For Julian Day 227, selecting COMIDA2 results # 7 of 9
242 228 3 4 1.15E-03
For Julian Day 228, selecting COMIDA2 results # 7 of 9
243 228 9 1 1.14E-03
For Julian Day 228, selecting COMIDA2 results # 7 of 9
244 228 24 10 1.14E-03
For Julian Day 228, selecting COMIDA2 results # 7 of 9
245 229 9 1 1.14E-03
For Julian Day 229, selecting COMIDA2 results # 7 of 9
246 229 14 14 1.14E-03
For Julian Day 229, selecting COMIDA2 results # 7 of 9
247 230 1 9 1.13E-03
For Julian Day 230, selecting COMIDA2 results # 7 of 9
248 230 11 4 1.15E-03
For Julian Day 230, selecting COMIDA2 results # 7 of 9
249 231 2 4 1.15E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9
250 231 4 1 1.14E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9

TRIAL DAY PERIOD BIN PRBMET
251 231 9 6 1.15E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9
252 231 12 5 1.13E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9
253 231 15 11 1.15E-03
For Julian Day 231, selecting COMIDA2 results # 7 of 9
254 232 22 11 1.15E-03
For Julian Day 232, selecting COMIDA2 results # 7 of 9
255 233 12 10 1.14E-03
For Julian Day 233, selecting COMIDA2 results # 7 of 9
256 233 18 14 1.14E-03
For Julian Day 233, selecting COMIDA2 results # 7 of 9
257 233 24 14 1.14E-03
For Julian Day 233, selecting COMIDA2 results # 7 of 9
258 234 5 1 1.14E-03
For Julian Day 234, selecting COMIDA2 results # 7 of 9
259 234 8 1 1.14E-03
For Julian Day 234, selecting COMIDA2 results # 7 of 9
260 235 10 5 1.13E-03
For Julian Day 235, selecting COMIDA2 results # 7 of 9
261 235 20 9 1.13E-03
For Julian Day 235, selecting COMIDA2 results # 7 of 9
262 235 21 13 1.14E-03
For Julian Day 235, selecting COMIDA2 results # 7 of 9
263 236 5 1 1.14E-03
For Julian Day 236, selecting COMIDA2 results # 7 of 9
264 236 11 10 1.14E-03
For Julian Day 236, selecting COMIDA2 results # 7 of 9
265 237 9 1 1.14E-03
For Julian Day 237, selecting COMIDA2 results # 7 of 9
266 238 2 4 1.15E-03
For Julian Day 238, selecting COMIDA2 results # 7 of 9
267 238 22 32 3.23E-04
For Julian Day 238, selecting COMIDA2 results # 7 of 9
268 239 1 21 1.13E-03
For Julian Day 239, selecting COMIDA2 results # 7 of 9
269 239 4 17 1.14E-03
For Julian Day 239, selecting COMIDA2 results # 7 of 9
270 239 12 11 1.15E-03
For Julian Day 239, selecting COMIDA2 results # 7 of 9
271 239 24 9 1.13E-03
For Julian Day 239, selecting COMIDA2 results # 7 of 9
272 240 3 4 1.15E-03
For Julian Day 240, selecting COMIDA2 results # 7 of 9
273 240 9 5 1.13E-03
For Julian Day 240, selecting COMIDA2 results # 7 of 9
274 240 12 10 1.14E-03
For Julian Day 240, selecting COMIDA2 results # 7 of 9
275 241 9 31 1.14E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
276 241 10 31 1.14E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
277 241 11 30 1.14E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
278 241 12 29 1.14E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
279 241 13 27 3.71E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
280 241 14 27 3.71E-04
For Julian Day 241, selecting COMIDA2 results # 7 of 9
281 241 16 20 1.12E-03
For Julian Day 241, selecting COMIDA2 results # 7 of 9
282 242 10 10 1.14E-03
For Julian Day 242, selecting COMIDA2 results # 7 of 9
283 242 13 9 1.13E-03
For Julian Day 242, selecting COMIDA2 results # 7 of 9
284 242 23 10 1.14E-03
For Julian Day 242, selecting COMIDA2 results # 7 of 9
285 243 18 9 1.13E-03
For Julian Day 243, selecting COMIDA2 results # 7 of 9
286 243 22 21 1.13E-03
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287 244 14 22 1.09E-03
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288 244 21 17 1.14E-03
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289 245 5 17 1.14E-03
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290 245 14 19 1.11E-03
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291 245 20 14 1.14E-03
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292 246 2 3 8.56E-04
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293 246 12 10 1.14E-03
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294 246 17 13 1.14E-03
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295 247 4 1 1.14E-03
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296 247 17 20 1.12E-03
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297 248 1 17 1.14E-03
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298 248 2 27 3.71E-04
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299 248 3 32 3.23E-04
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300 248 20 10 1.14E-03
For Julian Day 248, selecting COMIDA2 results # 7 of 9

TRIAL DAY PERIOD BIN PRBMET
301 249 9 4 1.15E-03
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302 249 10 4 1.15E-03
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303 250 3 3 8.56E-04
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304 250 12 9 1.13E-03
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305 250 13 13 1.14E-03
For Julian Day 250, selecting COMIDA2 results # 7 of 9
306 251 8 26 2.38E-04
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307 251 10 24 1.14E-04
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308 251 14 14 1.14E-03
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309 251 20 13 1.14E-03
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310 252 16 13 1.14E-03
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311 253 3 4 1.15E-03
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312 253 15 19 1.11E-03
For Julian Day 253, selecting COMIDA2 results # 7 of 9
313 253 16 18 5.99E-04
For Julian Day 253, selecting COMIDA2 results # 7 of 9
314 254 1 9 1.13E-03
For Julian Day 254, selecting COMIDA2 results # 7 of 9
315 254 6 1 1.14E-03
For Julian Day 254, selecting COMIDA2 results # 7 of 9
316 254 18 13 1.14E-03
For Julian Day 254, selecting COMIDA2 results # 7 of 9
317 254 22 9 1.13E-03
For Julian Day 254, selecting COMIDA2 results # 7 of 9
318 255 1 10 1.14E-03
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319 255 12 10 1.14E-03
For Julian Day 255, selecting COMIDA2 results # 7 of 9
320 256 4 21 1.13E-03
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321 256 5 20 1.12E-03
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322 256 6 19 1.11E-03
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323 256 8 17 1.14E-03
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324 256 16 11 1.15E-03
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325 256 18 26 2.38E-04
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326 256 19 25 1.52E-04
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327 256 20 23 1.14E-04
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328 257 20 10 1.14E-03
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329 258 9 17 1.14E-03
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330 258 15 11 1.15E-03
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331 259 13 10 1.14E-03
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332 259 17 14 1.14E-03
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333 259 21 14 1.14E-03
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334 260 5 5 1.13E-03
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335 261 13 15 1.12E-03
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336 261 15 14 1.14E-03
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337 262 7 5 1.13E-03
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338 262 9 4 1.15E-03
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339 263 17 11 1.15E-03
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340 263 19 11 1.15E-03
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341 263 24 15 1.12E-03
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342 264 6 1 1.14E-03
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343 264 10 10 1.14E-03
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344 265 8 2 1.14E-03
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345 266 6 2 1.14E-03
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346 266 11 4 1.15E-03

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347 266 15 10 1.14E-03
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348 267 7 21 1.13E-03
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349 267 8 20 1.12E-03
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350 267 12 11 1.15E-03
For Julian Day 267, selecting COMIDA2 results # 8 of 9

TRIAL DAY PERIOD BIN PRBMET
351 268 16 14 1.14E-03
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352 268 23 14 1.14E-03
For Julian Day 268, selecting COMIDA2 results # 8 of 9
353 268 24 13 1.14E-03
For Julian Day 268, selecting COMIDA2 results # 8 of 9
354 269 7 5 1.13E-03
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355 269 9 10 1.14E-03
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356 269 12 9 1.13E-03
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357 269 16 13 1.14E-03
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358 270 23 9 1.13E-03
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359 271 10 7 1.13E-03
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360 271 11 30 1.14E-04
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361 271 12 27 3.71E-04
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362 271 15 22 1.09E-03
For Julian Day 271, selecting COMIDA2 results # 8 of 9
363 272 3 6 1.15E-03
For Julian Day 272, selecting COMIDA2 results # 8 of 9
364 272 16 14 1.14E-03
For Julian Day 272, selecting COMIDA2 results # 8 of 9
365 273 3 3 8.56E-04
For Julian Day 273, selecting COMIDA2 results # 8 of 9
366 273 10 4 1.15E-03
For Julian Day 273, selecting COMIDA2 results # 8 of 9
367 273 13 11 1.15E-03
For Julian Day 273, selecting COMIDA2 results # 8 of 9
368 273 18 17 1.14E-03
For Julian Day 273, selecting COMIDA2 results # 8 of 9
369 274 4 1 1.14E-03
For Julian Day 274, selecting COMIDA2 results # 8 of 9
370 274 8 5 1.13E-03
For Julian Day 274, selecting COMIDA2 results # 8 of 9
371 274 9 10 1.14E-03
For Julian Day 274, selecting COMIDA2 results # 8 of 9
372 274 16 15 1.12E-03
For Julian Day 274, selecting COMIDA2 results # 8 of 9
373 274 21 14 1.14E-03
For Julian Day 274, selecting COMIDA2 results # 8 of 9
374 275 22 9 1.13E-03
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375 276 6 10 1.14E-03
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376 276 8 1 1.14E-03
For Julian Day 276, selecting COMIDA2 results # 8 of 9
377 276 18 13 1.14E-03
For Julian Day 276, selecting COMIDA2 results # 8 of 9
378 276 23 14 1.14E-03
For Julian Day 276, selecting COMIDA2 results # 8 of 9
379 277 20 5 1.13E-03
For Julian Day 277, selecting COMIDA2 results # 8 of 9
380 277 22 6 1.15E-03
For Julian Day 277, selecting COMIDA2 results # 8 of 9
381 278 2 2 1.14E-03
For Julian Day 278, selecting COMIDA2 results # 8 of 9
382 278 18 18 5.99E-04
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383 279 1 17 1.14E-03
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384 279 5 17 1.14E-03
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385 279 23 2 1.14E-03
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386 280 5 1 1.14E-03
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387 280 19 1 1.14E-03
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388 281 13 1 1.14E-03
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389 281 20 1 1.14E-03
For Julian Day 281, selecting COMIDA2 results # 8 of 9
390 282 2 1 1.14E-03
For Julian Day 282, selecting COMIDA2 results # 8 of 9
391 282 16 20 1.12E-03
For Julian Day 282, selecting COMIDA2 results # 8 of 9
392 282 18 19 1.11E-03
For Julian Day 282, selecting COMIDA2 results # 8 of 9
393 283 2 1 1.14E-03
For Julian Day 283, selecting COMIDA2 results # 8 of 9
394 283 11 9 1.13E-03
For Julian Day 283, selecting COMIDA2 results # 8 of 9
395 283 12 13 1.14E-03
For Julian Day 283, selecting COMIDA2 results # 8 of 9
396 283 24 9 1.13E-03
For Julian Day 283, selecting COMIDA2 results # 8 of 9
397 284 13 18 5.99E-04
For Julian Day 284, selecting COMIDA2 results # 8 of 9
398 284 15 17 1.14E-03
For Julian Day 284, selecting COMIDA2 results # 8 of 9
399 286 23 14 1.14E-03
For Julian Day 286, selecting COMIDA2 results # 8 of 9

400 287 4 2 1.14E-03
For Julian Day 287, selecting COMIDA2 results # 9 of 9

TRIAL	DAY	PERIOD	BIN	PRBMET
401	287	24	14	1.14E-03

For Julian Day 287, selecting COMIDA2 results # 9 of 9
402 288 4 1 1.14E-03
For Julian Day 288, selecting COMIDA2 results # 9 of 9
403 288 20 14 1.14E-03
For Julian Day 288, selecting COMIDA2 results # 9 of 9
404 288 23 14 1.14E-03
For Julian Day 288, selecting COMIDA2 results # 9 of 9
405 289 5 4 1.15E-03
For Julian Day 289, selecting COMIDA2 results # 9 of 9
406 289 12 10 1.14E-03
For Julian Day 289, selecting COMIDA2 results # 9 of 9
407 289 18 11 1.15E-03
For Julian Day 289, selecting COMIDA2 results # 9 of 9
408 289 19 11 1.15E-03
For Julian Day 289, selecting COMIDA2 results # 9 of 9
409 290 7 27 3.71E-04
For Julian Day 290, selecting COMIDA2 results # 9 of 9
410 290 14 12 1.15E-03
For Julian Day 290, selecting COMIDA2 results # 9 of 9
411 290 20 10 1.14E-03
For Julian Day 290, selecting COMIDA2 results # 9 of 9
412 291 6 1 1.14E-03
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413 291 24 26 2.38E-04
For Julian Day 291, selecting COMIDA2 results # 9 of 9
414 292 1 26 2.38E-04
For Julian Day 292, selecting COMIDA2 results # 9 of 9
415 292 5 25 1.52E-04
For Julian Day 292, selecting COMIDA2 results # 9 of 9
416 292 6 25 1.52E-04
For Julian Day 292, selecting COMIDA2 results # 9 of 9
417 292 7 24 1.14E-04
For Julian Day 292, selecting COMIDA2 results # 9 of 9
418 292 9 6 1.15E-03
For Julian Day 292, selecting COMIDA2 results # 9 of 9
419 292 17 22 1.09E-03
For Julian Day 292, selecting COMIDA2 results # 9 of 9
420 292 21 17 1.14E-03
For Julian Day 292, selecting COMIDA2 results # 9 of 9
421 292 24 17 1.14E-03
For Julian Day 292, selecting COMIDA2 results # 9 of 9
422 293 23 14 1.14E-03
For Julian Day 293, selecting COMIDA2 results # 9 of 9
423 294 5 2 1.14E-03
For Julian Day 294, selecting COMIDA2 results # 9 of 9
424 294 21 13 1.14E-03
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425 295 4 10 1.14E-03
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426 295 5 5 1.13E-03
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427 296 5 6 1.15E-03
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428 296 10 6 1.15E-03
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429 296 20 11 1.15E-03
For Julian Day 296, selecting COMIDA2 results # 9 of 9
430 297 12 6 1.15E-03
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431 298 1 11 1.15E-03
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432 298 2 5 1.13E-03
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433 298 9 6 1.15E-03
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434 298 15 11 1.15E-03
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435 298 23 10 1.14E-03
For Julian Day 298, selecting COMIDA2 results # 9 of 9
436 299 7 2 1.14E-03
For Julian Day 299, selecting COMIDA2 results # 9 of 9
437 299 17 14 1.14E-03
For Julian Day 299, selecting COMIDA2 results # 9 of 9
438 299 18 13 1.14E-03
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439 299 19 9 1.13E-03
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440 300 1 21 1.13E-03
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441 300 2 21 1.13E-03
For Julian Day 300, selecting COMIDA2 results # 9 of 9
442 300 14 22 1.09E-03
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443 301 15 12 1.15E-03
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444 302 11 12 1.15E-03
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445 304 17 20 1.12E-03
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446 304 19 19 1.11E-03
For Julian Day 304, selecting COMIDA2 results # 9 of 9
447 305 1 13 1.14E-03
For Julian Day 305, selecting COMIDA2 results # 9 of 9
448 306 3 6 1.15E-03
For Julian Day 306, selecting COMIDA2 results # 9 of 9
449 306 10 11 1.15E-03
For Julian Day 306, selecting COMIDA2 results # 9 of 9
450 306 14 11 1.15E-03
For Julian Day 306, selecting COMIDA2 results # 9 of 9

TRIAL	DAY	PERIOD	BIN	PRBMET
451	307	12	11	1.15E-03

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452 307 18 10 1.14E-03
For Julian Day 307, selecting COMIDA2 results # 9 of 9
453 307 22 15 1.12E-03
For Julian Day 307, selecting COMIDA2 results # 9 of 9
454 308 4 2 1.14E-03
For Julian Day 308, selecting COMIDA2 results # 9 of 9
455 308 18 14 1.14E-03
For Julian Day 308, selecting COMIDA2 results # 9 of 9
456 309 3 3 8.56E-04
For Julian Day 309, selecting COMIDA2 results # 9 of 9
457 309 4 4 1.15E-03
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458 309 7 1 1.14E-03
For Julian Day 309, selecting COMIDA2 results # 9 of 9
459 309 12 14 1.14E-03
For Julian Day 309, selecting COMIDA2 results # 9 of 9
460 309 13 9 1.13E-03
For Julian Day 309, selecting COMIDA2 results # 9 of 9
461 309 18 13 1.14E-03
For Julian Day 309, selecting COMIDA2 results # 9 of 9
462 310 9 4 1.15E-03
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463 310 19 13 1.14E-03
For Julian Day 310, selecting COMIDA2 results # 9 of 9
464 311 10 20 1.12E-03
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465 311 15 17 1.14E-03
For Julian Day 311, selecting COMIDA2 results # 9 of 9
466 312 3 22 1.09E-03
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467 312 21 12 1.15E-03
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468 313 2 5 1.13E-03
For Julian Day 313, selecting COMIDA2 results # 9 of 9
469 313 22 14 1.14E-03
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470 314 9 3 8.56E-04
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471 314 15 13 1.14E-03
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472 315 13 10 1.14E-03
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473 315 17 10 1.14E-03
For Julian Day 315, selecting COMIDA2 results # 9 of 9
474 316 2 17 1.14E-03
For Julian Day 316, selecting COMIDA2 results # 9 of 9
475 316 6 7 1.13E-03
For Julian Day 316, selecting COMIDA2 results # 9 of 9
476 316 10 6 1.15E-03
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477 316 11 7 1.13E-03
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478 317 9 12 1.15E-03
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479 317 22 5 1.13E-03
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480 318 12 9 1.13E-03
For Julian Day 318, selecting COMIDA2 results # 9 of 9
481 318 24 9 1.13E-03
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482 319 6 1 1.14E-03
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483 319 15 10 1.14E-03
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484 319 19 21 1.13E-03
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485 319 22 19 1.11E-03
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486 319 23 17 1.14E-03
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487 320 6 27 3.71E-04
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488 320 7 32 3.23E-04
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489 320 10 32 3.23E-04
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490 320 18 11 1.15E-03
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491 321 1 11 1.15E-03
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492 321 3 6 1.15E-03
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493 321 14 6 1.15E-03
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494 321 17 12 1.15E-03
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495 322 18 11 1.15E-03
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496 323 8 6 1.15E-03
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497 323 9 5 1.13E-03
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498 323 12 4 1.15E-03
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499 323 24 5 1.13E-03
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500 324 18 6 1.15E-03
For Julian Day 324, selecting COMIDA2 results # 9 of 9

TRIAL DAY PERIOD BIN PRBMET
501 325 4 2 1.14E-03
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502 325 16 11 1.15E-03
For Julian Day 325, selecting COMIDA2 results # 9 of 9
503 325 17 11 1.15E-03
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504 326 1 6 1.15E-03
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505 326 14 17 1.14E-03

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506 326 19 26 2.38E-04
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507 326 20 25 1.52E-04
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508 327 1 6 1.15E-03
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509 327 5 20 1.12E-03
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510 327 18 12 1.15E-03
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511 328 5 6 1.15E-03
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512 328 20 14 1.14E-03
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513 329 13 13 1.14E-03
For Julian Day 329, selecting COMIDA2 results # 9 of 9
514 329 14 14 1.14E-03
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515 330 6 10 1.14E-03
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516 330 13 21 1.13E-03
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519 331 20 13 1.14E-03
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520 331 24 13 1.14E-03
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521 332 10 11 1.15E-03
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522 333 2 10 1.14E-03
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523 333 13 9 1.13E-03
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525 333 24 10 1.14E-03
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526 334 12 12 1.15E-03
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530 335 24 12 1.15E-03
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534 338 8 6 1.15E-03
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538 339 24 10 1.14E-03
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539 340 1 11 1.15E-03
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540 340 19 10 1.14E-03
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541 341 18 6 1.15E-03
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542 342 5 7 1.13E-03
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543 342 13 6 1.15E-03
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544 342 23 11 1.15E-03
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545 343 16 15 1.12E-03
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546 343 17 14 1.14E-03
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547 343 24 13 1.14E-03
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548 344 8 2 1.14E-03
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549 344 12 15 1.12E-03
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550 345 3 13 1.14E-03
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TRIAL DAY PERIOD BIN PRBMET
551 346 1 13 1.14E-03
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552 346 3 4 1.15E-03
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553 346 4 1 1.14E-03
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554 346 18 10 1.14E-03
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555 347 4 17 1.14E-03
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556 347 7 5 1.13E-03
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557 347 18 21 1.13E-03
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558 348 5 4 1.15E-03
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559 348 9 14 1.14E-03
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560 348 15 10 1.14E-03
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561 348 19 10 1.14E-03
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562 349 1 9 1.13E-03
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563 349 19 12 1.15E-03
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564 349 21 16 1.14E-04
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565 349 23 16 1.14E-04
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566 350 7 6 1.15E-03
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567 350 16 13 1.14E-03
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568 351 6 14 1.14E-03
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569 351 19 13 1.14E-03
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570 352 2 13 1.14E-03
For Julian Day 352, selecting COMIDA2 results # 1 of 9
571 352 18 11 1.15E-03
For Julian Day 352, selecting COMIDA2 results # 1 of 9
572 353 1 10 1.14E-03
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573 353 5 5 1.13E-03
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574 353 9 6 1.15E-03
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575 353 12 11 1.15E-03
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576 353 22 14 1.14E-03
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577 355 1 20 1.12E-03
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578 355 18 14 1.14E-03
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579 356 4 19 1.11E-03
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580 356 19 22 1.09E-03
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581 356 21 10 1.14E-03
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582 356 22 10 1.14E-03
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583 357 13 11 1.15E-03
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584 357 17 15 1.12E-03
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585 358 4 6 1.15E-03
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586 358 24 21 1.13E-03
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587 359 5 18 5.99E-04
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588 359 14 17 1.14E-03
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589 360 15 6 1.15E-03
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590 360 22 6 1.15E-03
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591 361 12 11 1.15E-03
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593 363 5 1 1.14E-03
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594 363 8 4 1.15E-03
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595 363 15 14 1.14E-03
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596 363 19 9 1.13E-03
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597 364 1 9 1.13E-03
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598 364 9 5 1.13E-03
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599 364 20 10 1.14E-03
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600 365 1 10 1.14E-03
For Julian Day 365, selecting COMIDA2 results # 1 of 9

TRIAL DAY PERIOD BIN PRBMET
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602 1 7 6 1.15E-03
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603 1 9 5 1.13E-03
For Julian Day 1, selecting COMIDA2 results # 1 of 9
604 1 15 14 1.14E-03
For Julian Day 1, selecting COMIDA2 results # 1 of 9
605 1 17 13 1.14E-03
For Julian Day 1, selecting COMIDA2 results # 1 of 9
606 1 22 21 1.13E-03
For Julian Day 1, selecting COMIDA2 results # 1 of 9
607 1 24 20 1.12E-03
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608 2 5 17 1.14E-03
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609 2 8 17 1.14E-03
For Julian Day 2, selecting COMIDA2 results # 1 of 9
610 2 9 27 3.71E-04
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611 3 8 6 1.15E-03
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612 3 16 5 1.13E-03

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613 3 24 4 1.15E-03
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614 5 2 11 1.15E-03
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615 6 2 6 1.15E-03
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616 6 8 6 1.15E-03
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617 7 9 6 1.15E-03
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618 7 14 5 1.13E-03
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619 7 16 4 1.15E-03
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620 7 19 10 1.14E-03
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621 7 22 10 1.14E-03
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622 8 15 12 1.15E-03
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623 9 3 11 1.15E-03
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624 9 4 6 1.15E-03
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625 9 16 10 1.14E-03
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626 11 8 23 1.14E-04
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627 11 9 22 1.09E-03
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628 11 23 11 1.15E-03
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629 12 7 2 1.14E-03
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630 13 3 10 1.14E-03
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631 13 16 21 1.13E-03
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632 14 9 8 3.04E-04
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633 14 11 8 3.04E-04
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634 14 13 8 3.04E-04
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635 14 17 8 3.04E-04
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636 14 20 8 3.04E-04
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637 14 23 8 3.04E-04
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638 15 1 8 3.04E-04
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639 15 3 8 3.04E-04
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640 15 7 8 3.04E-04
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641 15 9 8 3.04E-04
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642 15 15 12 1.15E-03
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643 16 3 5 1.13E-03
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644 16 6 1 1.14E-03
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645 16 8 4 1.15E-03
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646 16 10 3 8.56E-04
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647 17 12 20 1.12E-03
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648 17 21 17 1.14E-03
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649 17 24 22 1.09E-03
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650 18 9 8 3.04E-04
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TRIAL DAY PERIOD BIN PRBMET
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652 19 2 9 1.13E-03
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653 19 5 10 1.14E-03
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654 19 12 11 1.15E-03
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655 19 13 12 1.15E-03
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656 19 19 10 1.14E-03
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657 20 15 13 1.14E-03
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658 20 19 14 1.14E-03
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659 21 6 6 1.15E-03
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660 21 10 7 1.13E-03
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661 21 14 6 1.15E-03
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662 21 24 11 1.15E-03
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663 22 2 4 1.15E-03
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664 22 20 17 1.14E-03
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665 22 22 19 1.11E-03
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666 23 16 10 1.14E-03
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667 24 6 1 1.14E-03
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668 24 18 11 1.15E-03
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669 24 20 16 1.14E-04
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670 25 12 7 1.13E-03
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671 25 18 7 1.13E-03
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672 26 7 7 1.13E-03
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673 26 20 6 1.15E-03
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674 27 5 5 1.13E-03
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675 27 12 14 1.14E-03
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676 28 4 10 1.14E-03
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677 28 15 13 1.14E-03
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678 28 17 13 1.14E-03
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679 29 3 20 1.12E-03
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680 29 4 19 1.11E-03
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681 29 10 6 1.15E-03
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682 29 12 9 1.13E-03
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683 29 23 18 5.99E-04
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684 30 6 4 1.15E-03
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685 30 22 10 1.14E-03
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686 31 1 21 1.13E-03
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687 31 11 7 1.13E-03
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688 31 17 6 1.15E-03
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689 32 5 1 1.14E-03
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690 32 20 10 1.14E-03
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691 33 6 5 1.13E-03
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692 33 16 17 1.14E-03
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693 33 24 17 1.14E-03
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694 34 4 6 1.15E-03
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695 34 10 6 1.15E-03
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696 34 15 9 1.13E-03
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697 34 19 13 1.14E-03
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698 35 11 18 5.99E-04
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699 35 15 22 1.09E-03
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700 35 16 17 1.14E-03
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TRIAL DAY PERIOD BIN PRBMET
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702 36 13 12 1.15E-03
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703 36 21 12 1.15E-03
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704 37 4 7 1.13E-03
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705 37 13 11 1.15E-03
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706 38 11 6 1.15E-03
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707 38 23 11 1.15E-03
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708 39 24 11 1.15E-03
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709 40 5 2 1.14E-03
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710 40 9 6 1.15E-03
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711 40 22 11 1.15E-03
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712 41 9 5 1.13E-03
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713 41 21 10 1.14E-03
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714 42 3 21 1.13E-03
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715 42 24 17 1.14E-03
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716 43 13 12 1.15E-03
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717 43 24 16 1.14E-04
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718 44 5 2 1.14E-03
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719 44 13 14 1.14E-03

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720 45 1 14 1.14E-03
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721 45 3 4 1.15E-03
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722 45 12 10 1.14E-03
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723 46 2 13 1.14E-03
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724 46 11 9 1.13E-03
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725 47 2 13 1.14E-03
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726 47 13 15 1.12E-03
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727 48 10 6 1.15E-03
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728 48 13 6 1.15E-03
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729 48 21 10 1.14E-03
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730 49 14 7 1.13E-03
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731 49 24 12 1.15E-03
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732 50 1 5 1.13E-03
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733 50 11 5 1.13E-03
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734 50 15 11 1.15E-03
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735 50 23 10 1.14E-03
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736 51 5 1 1.14E-03
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737 52 1 14 1.14E-03
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738 52 10 6 1.15E-03
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739 52 21 13 1.14E-03
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740 53 11 4 1.15E-03
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741 53 23 21 1.13E-03
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742 54 11 12 1.15E-03
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743 54 12 11 1.15E-03
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744 54 14 15 1.12E-03
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745 55 1 7 1.13E-03
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746 55 4 8 3.04E-04
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747 55 6 2 1.14E-03
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748 55 23 11 1.15E-03
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749 55 24 10 1.14E-03
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750 56 4 2 1.14E-03
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TRIAL DAY PERIOD BIN PRBMET
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752 57 2 7 1.13E-03
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753 58 8 6 1.15E-03
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754 58 10 6 1.15E-03
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755 59 1 5 1.13E-03
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756 59 5 2 1.14E-03
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757 59 19 10 1.14E-03
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758 59 22 6 1.15E-03
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759 60 23 20 1.12E-03
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760 61 1 20 1.12E-03
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761 61 15 7 1.13E-03
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762 62 17 7 1.13E-03
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763 62 21 6 1.15E-03
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764 63 4 2 1.14E-03
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765 63 14 12 1.15E-03
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766 64 1 6 1.15E-03
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767 64 17 12 1.15E-03
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768 64 18 11 1.15E-03
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769 65 2 4 1.15E-03
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770 65 6 1 1.14E-03
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771 66 4 6 1.15E-03
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772 66 5 2 1.14E-03
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773 66 8 2 1.14E-03
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774 66 13 10 1.14E-03
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775 66 19 10 1.14E-03
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776 66 23 14 1.14E-03
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777 67 10 5 1.13E-03
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778 67 24 5 1.13E-03
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779 68 11 6 1.15E-03
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780 68 24 12 1.15E-03
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781 69 8 7 1.13E-03
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782 69 19 11 1.15E-03
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783 70 5 1 1.14E-03
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784 70 16 11 1.15E-03
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785 71 4 17 1.14E-03
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786 71 5 3 8.56E-04
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787 71 7 4 1.15E-03
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788 71 22 18 5.99E-04
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789 72 3 9 1.13E-03
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790 72 8 2 1.14E-03
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791 72 21 12 1.15E-03
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792 73 23 12 1.15E-03
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793 74 4 7 1.13E-03
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794 75 9 2 1.14E-03
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795 75 15 11 1.15E-03
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796 75 24 5 1.13E-03
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797 76 17 12 1.15E-03
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798 77 2 6 1.15E-03
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799 77 3 2 1.14E-03
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800 77 8 2 1.14E-03
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TRIAL DAY PERIOD BIN PRBMET
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802 78 13 6 1.15E-03
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803 78 15 7 1.13E-03
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804 79 7 2 1.14E-03
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805 80 2 1 1.14E-03
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806 80 13 4 1.15E-03
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807 80 23 6 1.15E-03
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808 81 9 6 1.15E-03
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809 81 21 6 1.15E-03
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810 82 21 5 1.13E-03
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811 83 4 2 1.14E-03
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812 83 14 11 1.15E-03
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813 83 19 5 1.13E-03
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814 84 6 1 1.14E-03
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815 84 11 5 1.13E-03
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816 84 13 10 1.14E-03
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817 84 20 11 1.15E-03
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818 85 21 12 1.15E-03
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819 86 1 6 1.15E-03
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820 86 6 2 1.14E-03
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821 86 10 6 1.15E-03
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822 86 12 11 1.15E-03
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823 86 14 14 1.14E-03
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824 86 19 13 1.14E-03
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825 87 9 1 1.14E-03
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826 87 15 4 1.15E-03

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827 87 18 21 1.13E-03
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828 87 24 19 1.11E-03
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829 88 23 13 1.14E-03
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830 89 5 10 1.14E-03
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831 90 13 12 1.15E-03
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832 91 2 5 1.13E-03
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833 91 10 7 1.13E-03
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834 92 7 1 1.14E-03
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835 92 16 13 1.14E-03
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836 92 21 10 1.14E-03
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837 93 1 5 1.13E-03
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838 93 5 21 1.13E-03
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839 93 10 36 1.43E-04
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840 93 11 35 1.14E-04
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842 93 13 33 1.14E-04
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843 93 15 27 3.71E-04
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844 93 19 12 1.15E-03
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845 94 4 2 1.14E-03
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846 94 16 11 1.15E-03
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847 95 3 7 1.13E-03
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848 95 7 2 1.14E-03
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849 95 10 6 1.15E-03
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850 95 24 10 1.14E-03
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TRIAL DAY PERIOD BIN PRBMET
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852 97 4 17 1.14E-03
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853 97 9 2 1.14E-03
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854 98 10 6 1.15E-03
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855 98 14 11 1.15E-03
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856 98 18 12 1.15E-03
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857 99 18 14 1.14E-03
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858 100 9 1 1.14E-03
For Julian Day 100, selecting COMIDA2 results # 3 of 9
859 100 16 9 1.13E-03
For Julian Day 100, selecting COMIDA2 results # 3 of 9
860 100 24 13 1.14E-03
For Julian Day 100, selecting COMIDA2 results # 3 of 9
861 101 15 11 1.15E-03
For Julian Day 101, selecting COMIDA2 results # 3 of 9
862 102 6 2 1.14E-03
For Julian Day 102, selecting COMIDA2 results # 3 of 9
863 102 7 6 1.15E-03
For Julian Day 102, selecting COMIDA2 results # 3 of 9
864 103 2 9 1.13E-03
For Julian Day 103, selecting COMIDA2 results # 3 of 9
865 103 21 20 1.12E-03
For Julian Day 103, selecting COMIDA2 results # 3 of 9
866 103 22 20 1.12E-03
For Julian Day 103, selecting COMIDA2 results # 3 of 9
867 104 6 10 1.14E-03
For Julian Day 104, selecting COMIDA2 results # 3 of 9
868 104 15 19 1.11E-03
For Julian Day 104, selecting COMIDA2 results # 3 of 9
869 104 22 14 1.14E-03
For Julian Day 104, selecting COMIDA2 results # 3 of 9
870 105 3 4 1.15E-03
For Julian Day 105, selecting COMIDA2 results # 3 of 9
871 105 8 2 1.14E-03
For Julian Day 105, selecting COMIDA2 results # 3 of 9
872 105 16 12 1.15E-03
For Julian Day 105, selecting COMIDA2 results # 3 of 9
873 105 20 11 1.15E-03
For Julian Day 105, selecting COMIDA2 results # 3 of 9
874 106 8 2 1.14E-03
For Julian Day 106, selecting COMIDA2 results # 3 of 9
875 106 10 1 1.14E-03
For Julian Day 106, selecting COMIDA2 results # 3 of 9
876 106 18 10 1.14E-03
For Julian Day 106, selecting COMIDA2 results # 3 of 9
877 106 19 5 1.13E-03
For Julian Day 106, selecting COMIDA2 results # 3 of 9
878 106 21 10 1.14E-03
For Julian Day 106, selecting COMIDA2 results # 3 of 9
879 107 7 1 1.14E-03
For Julian Day 107, selecting COMIDA2 results # 3 of 9

880 108 12 5 1.13E-03
For Julian Day 108, selecting COMIDA2 results # 3 of 9
881 109 7 2 1.14E-03
For Julian Day 109, selecting COMIDA2 results # 3 of 9
882 109 11 6 1.15E-03
For Julian Day 109, selecting COMIDA2 results # 3 of 9
883 109 20 11 1.15E-03
For Julian Day 109, selecting COMIDA2 results # 3 of 9
884 110 8 1 1.14E-03
For Julian Day 110, selecting COMIDA2 results # 3 of 9
885 110 16 13 1.14E-03
For Julian Day 110, selecting COMIDA2 results # 3 of 9
886 111 17 17 1.14E-03
For Julian Day 111, selecting COMIDA2 results # 3 of 9
887 111 24 17 1.14E-03
For Julian Day 111, selecting COMIDA2 results # 3 of 9
888 112 4 23 1.14E-04
For Julian Day 112, selecting COMIDA2 results # 3 of 9
889 112 13 32 3.23E-04
For Julian Day 112, selecting COMIDA2 results # 3 of 9
890 112 15 27 3.71E-04
For Julian Day 112, selecting COMIDA2 results # 3 of 9
891 112 19 17 1.14E-03
For Julian Day 112, selecting COMIDA2 results # 3 of 9
892 113 14 21 1.13E-03
For Julian Day 113, selecting COMIDA2 results # 3 of 9
893 114 8 2 1.14E-03
For Julian Day 114, selecting COMIDA2 results # 3 of 9
894 115 3 4 1.15E-03
For Julian Day 115, selecting COMIDA2 results # 3 of 9
895 115 5 1 1.14E-03
For Julian Day 115, selecting COMIDA2 results # 3 of 9
896 115 13 12 1.15E-03
For Julian Day 115, selecting COMIDA2 results # 3 of 9
897 116 14 10 1.14E-03
For Julian Day 116, selecting COMIDA2 results # 3 of 9
898 116 18 9 1.13E-03
For Julian Day 116, selecting COMIDA2 results # 3 of 9
899 116 19 14 1.14E-03
For Julian Day 116, selecting COMIDA2 results # 3 of 9
900 117 6 1 1.14E-03
For Julian Day 117, selecting COMIDA2 results # 3 of 9

TRIAL DAY PERIOD BIN PRBMET
901 117 7 1 1.14E-03
For Julian Day 117, selecting COMIDA2 results # 3 of 9
902 117 14 14 1.14E-03
For Julian Day 117, selecting COMIDA2 results # 3 of 9
903 117 16 13 1.14E-03
For Julian Day 117, selecting COMIDA2 results # 3 of 9
904 118 9 2 1.14E-03
For Julian Day 118, selecting COMIDA2 results # 3 of 9
905 119 10 1 1.14E-03
For Julian Day 119, selecting COMIDA2 results # 3 of 9
906 119 16 10 1.14E-03
For Julian Day 119, selecting COMIDA2 results # 3 of 9
907 119 17 11 1.15E-03
For Julian Day 119, selecting COMIDA2 results # 3 of 9
908 119 22 9 1.13E-03
For Julian Day 119, selecting COMIDA2 results # 3 of 9
909 121 4 1 1.14E-03
For Julian Day 121, selecting COMIDA2 results # 3 of 9
910 121 17 10 1.14E-03
For Julian Day 121, selecting COMIDA2 results # 3 of 9
911 122 3 4 1.15E-03
For Julian Day 122, selecting COMIDA2 results # 3 of 9
912 122 9 1 1.14E-03
For Julian Day 122, selecting COMIDA2 results # 3 of 9
913 122 18 10 1.14E-03
For Julian Day 122, selecting COMIDA2 results # 3 of 9
914 123 4 1 1.14E-03
For Julian Day 123, selecting COMIDA2 results # 3 of 9
915 123 21 14 1.14E-03
For Julian Day 123, selecting COMIDA2 results # 3 of 9
916 123 24 14 1.14E-03
For Julian Day 123, selecting COMIDA2 results # 3 of 9
917 124 8 2 1.14E-03
For Julian Day 124, selecting COMIDA2 results # 3 of 9
918 124 19 15 1.12E-03
For Julian Day 124, selecting COMIDA2 results # 3 of 9
919 125 2 5 1.13E-03
For Julian Day 125, selecting COMIDA2 results # 3 of 9
920 125 6 1 1.14E-03
For Julian Day 125, selecting COMIDA2 results # 3 of 9
921 125 16 13 1.14E-03
For Julian Day 125, selecting COMIDA2 results # 3 of 9
922 126 4 4 1.15E-03
For Julian Day 126, selecting COMIDA2 results # 3 of 9
923 126 10 6 1.15E-03
For Julian Day 126, selecting COMIDA2 results # 3 of 9
924 127 8 1 1.14E-03
For Julian Day 127, selecting COMIDA2 results # 3 of 9
925 127 20 18 5.99E-04
For Julian Day 127, selecting COMIDA2 results # 3 of 9
926 127 24 10 1.14E-03
For Julian Day 127, selecting COMIDA2 results # 3 of 9
927 128 6 4 1.15E-03
For Julian Day 128, selecting COMIDA2 results # 3 of 9
928 128 19 9 1.13E-03
For Julian Day 128, selecting COMIDA2 results # 3 of 9
929 129 10 1 1.14E-03
For Julian Day 129, selecting COMIDA2 results # 3 of 9
930 129 19 10 1.14E-03
For Julian Day 129, selecting COMIDA2 results # 3 of 9
931 130 3 10 1.14E-03
For Julian Day 130, selecting COMIDA2 results # 3 of 9
932 130 13 11 1.15E-03
For Julian Day 130, selecting COMIDA2 results # 3 of 9
933 130 20 9 1.13E-03

For Julian Day 130, selecting COMIDA2 results # 3 of 9
934 131 13 19 1.11E-03
For Julian Day 131, selecting COMIDA2 results # 3 of 9
935 131 20 17 1.14E-03
For Julian Day 131, selecting COMIDA2 results # 3 of 9
936 132 4 1 1.14E-03
For Julian Day 132, selecting COMIDA2 results # 3 of 9
937 132 14 10 1.14E-03
For Julian Day 132, selecting COMIDA2 results # 3 of 9
938 133 18 21 1.13E-03
For Julian Day 133, selecting COMIDA2 results # 3 of 9
939 133 19 21 1.13E-03
For Julian Day 133, selecting COMIDA2 results # 3 of 9
940 133 22 20 1.12E-03
For Julian Day 133, selecting COMIDA2 results # 3 of 9
941 133 23 20 1.12E-03
For Julian Day 133, selecting COMIDA2 results # 3 of 9
942 134 1 19 1.11E-03
For Julian Day 134, selecting COMIDA2 results # 3 of 9
943 134 19 18 5.99E-04
For Julian Day 134, selecting COMIDA2 results # 3 of 9
944 134 23 17 1.14E-03
For Julian Day 134, selecting COMIDA2 results # 3 of 9
945 135 18 20 1.12E-03
For Julian Day 135, selecting COMIDA2 results # 3 of 9
946 136 12 6 1.15E-03
For Julian Day 136, selecting COMIDA2 results # 3 of 9
947 136 16 14 1.14E-03
For Julian Day 136, selecting COMIDA2 results # 3 of 9
948 136 21 14 1.14E-03
For Julian Day 136, selecting COMIDA2 results # 3 of 9
949 137 1 4 1.15E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9
950 137 2 5 1.13E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9

TRIAL DAY PERIOD BIN PRBMET
951 137 8 17 1.14E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9
952 137 13 10 1.14E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9
953 137 15 11 1.15E-03
For Julian Day 137, selecting COMIDA2 results # 4 of 9
954 138 1 3 8.56E-04
For Julian Day 138, selecting COMIDA2 results # 4 of 9
955 138 9 5 1.13E-03
For Julian Day 138, selecting COMIDA2 results # 4 of 9
956 139 13 12 1.15E-03
For Julian Day 139, selecting COMIDA2 results # 4 of 9
957 140 5 2 1.14E-03
For Julian Day 140, selecting COMIDA2 results # 4 of 9
958 140 20 14 1.14E-03
For Julian Day 140, selecting COMIDA2 results # 4 of 9
959 140 22 14 1.14E-03
For Julian Day 140, selecting COMIDA2 results # 4 of 9
960 140 24 13 1.14E-03
For Julian Day 140, selecting COMIDA2 results # 4 of 9
961 141 16 12 1.15E-03
For Julian Day 141, selecting COMIDA2 results # 4 of 9
962 142 1 6 1.15E-03
For Julian Day 142, selecting COMIDA2 results # 4 of 9
963 142 6 2 1.14E-03
For Julian Day 142, selecting COMIDA2 results # 4 of 9
964 142 14 11 1.15E-03
For Julian Day 142, selecting COMIDA2 results # 4 of 9
965 142 21 15 1.12E-03
For Julian Day 142, selecting COMIDA2 results # 4 of 9
966 144 6 2 1.14E-03
For Julian Day 144, selecting COMIDA2 results # 4 of 9
967 144 7 1 1.14E-03
For Julian Day 144, selecting COMIDA2 results # 4 of 9
968 145 17 21 1.13E-03
For Julian Day 145, selecting COMIDA2 results # 4 of 9
969 146 12 5 1.13E-03
For Julian Day 146, selecting COMIDA2 results # 4 of 9
970 146 14 10 1.14E-03
For Julian Day 146, selecting COMIDA2 results # 4 of 9
971 146 24 14 1.14E-03
For Julian Day 146, selecting COMIDA2 results # 4 of 9
972 147 8 6 1.15E-03
For Julian Day 147, selecting COMIDA2 results # 4 of 9
973 147 12 11 1.15E-03
For Julian Day 147, selecting COMIDA2 results # 4 of 9
974 147 20 10 1.14E-03
For Julian Day 147, selecting COMIDA2 results # 4 of 9
975 147 22 14 1.14E-03
For Julian Day 147, selecting COMIDA2 results # 4 of 9
976 148 4 4 1.15E-03
For Julian Day 148, selecting COMIDA2 results # 4 of 9
977 148 9 1 1.14E-03
For Julian Day 148, selecting COMIDA2 results # 4 of 9
978 148 19 13 1.14E-03
For Julian Day 148, selecting COMIDA2 results # 4 of 9
979 149 1 9 1.13E-03
For Julian Day 149, selecting COMIDA2 results # 4 of 9
980 149 10 4 1.15E-03
For Julian Day 149, selecting COMIDA2 results # 4 of 9
981 149 23 13 1.14E-03
For Julian Day 149, selecting COMIDA2 results # 4 of 9
982 150 3 5 1.13E-03
For Julian Day 150, selecting COMIDA2 results # 4 of 9
983 150 17 10 1.14E-03
For Julian Day 150, selecting COMIDA2 results # 4 of 9
984 151 3 1 1.14E-03
For Julian Day 151, selecting COMIDA2 results # 4 of 9

"ATMOS" DESCRIPTION = OCP3 high density no spray
PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONS PROB TRIAL

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Source Term 1: Plume 1, at     0-0.2 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  1.21E+10  9.52E+09  2.20E+10  3.20E+10    ****     ****   3.98E+10  2.18E-02   73
 Cs-137   Ground Air Conc. (Bq-s/m3)    1.0000  4.65E+09  3.69E+09  7.95E+09  9.67E+09  1.29E+10  1.45E+10  1.80E+10  1.40E-03  159
 Cs-137   Center Ground Conc. (Bq/m2)   1.0000  4.18E+07  3.39E+07  7.37E+07  8.24E+07  1.08E+08  1.23E+08  1.55E+08  1.40E-03  159
 Total    Center Ground Conc. (Bq/m2)   1.0000  1.19E+08  9.75E+07  2.11E+08  2.36E+08  3.08E+08  3.51E+08  4.45E+08  1.40E-03  159
 Ground-Level Dilution, X/Q (s/m3)      1.0000  4.10E-05  3.33E-05  7.25E-05  7.95E-05  9.83E-05  1.15E-04  1.60E-04  1.40E-03  159
 Cs-137   Adjusted Source, Q (Bq)       1.0000  1.14E+14  1.01E+14  1.04E+14  1.05E+14  1.08E+14  1.09E+14  1.14E+14  3.04E-04  641
 Plume Sigma-y (m)                      1.0000  3.50E+01  3.34E+01    ****      ****      ****      ****   4.46E+01  1.57E-01   23
 Plume Sigma-z (m)                      1.0000  3.05E+01  2.80E+01  5.14E+01    ****      ****      ****   5.14E+01  9.99E-02    4
 Plume Height  (m)                      1.0000  5.00E+01    ****      ****      ****      ****      ****   5.00E+01  1.00E+00    1
 Plume Arrival Time (s)                 1.0000  1.46E+05  1.10E+05  1.37E+05    ****      ****      ****   1.46E+05  6.28E-02    8

Source Term 1: Plume 1, at   0.2-0.5 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  3.14E+09  2.51E+09  5.69E+09  7.86E+09    ****     ****   1.08E+10  2.18E-02   73
 Cs-137   Ground Air Conc. (Bq-s/m3)    1.0000  2.08E+09  1.62E+09  3.50E+09  4.39E+09  5.68E+09  6.12E+09  8.32E+09  1.40E-03  159
 Cs-137   Center Ground Conc. (Bq/m2)   1.0000  1.81E+07  1.42E+07  3.09E+07  3.42E+07  4.32E+07  4.77E+07  6.59E+07  1.40E-03  159
 Total    Center Ground Conc. (Bq/m2)   1.0000  5.19E+07  4.29E+07  9.51E+07  1.10E+08  1.41E+08  1.57E+08  1.92E+08  1.40E-03  159
 Ground-Level Dilution, X/Q (s/m3)      1.0000  1.86E-05  1.41E-05  3.26E-05  4.11E-05  5.61E-05  6.07E-05  7.66E-05  1.40E-03  159
 Cs-137   Adjusted Source, Q (Bq)       1.0000  1.12E+14  1.01E+14  1.04E+14  1.05E+14  1.07E+14  1.09E+14  1.14E+14  3.04E-04  641
 Plume Sigma-y (m)                      1.0000  1.03E+02  1.07E+02    ****      ****      ****      ****   1.39E+02  1.57E-01   23
 Plume Sigma-z (m)                      1.0000  5.52E+01  4.59E+01    ****      ****      ****      ****   1.60E+02  1.09E-01    4
 Plume Height  (m)                      1.0000  5.00E+01    ****      ****      ****      ****      ****   5.00E+01  1.00E+00    1
 Plume Arrival Time (s)                 1.0000  1.46E+05  1.10E+05  1.38E+05    ****      ****      ****   1.47E+05  6.28E-02    8

Source Term 1: Plume 1, at   0.5-1.2 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  1.22E+09  1.09E+09  2.32E+09  3.07E+09  4.15E+09    ****   4.30E+09  8.25E-03  284
 Cs-137   Ground Air Conc. (Bq-s/m3)    1.0000  1.14E+09  1.02E+09  2.02E+09  2.39E+09  3.30E+09  3.63E+09  3.95E+09  2.68E-03  158
 Cs-137   Center Ground Conc. (Bq/m2)   1.0000  9.21E+06  8.22E+06  1.64E+07  2.04E+07  2.40E+07  2.58E+07  3.30E+07  1.14E-03  588
 Total    Center Ground Conc. (Bq/m2)   1.0000  2.69E+07  2.40E+07  4.94E+07  5.72E+07  7.53E+07  8.18E+07  9.75E+07  1.14E-03  588
 Ground-Level Dilution, X/Q (s/m3)      1.0000  1.05E-05  8.75E-06  1.99E-05  2.35E-05  3.23E-05  3.53E-05  3.83E-05  2.68E-03  158
 Cs-137   Adjusted Source, Q (Bq)       1.0000  1.10E+14  1.01E+14  1.04E+14  1.06E+14  1.09E+14  1.10E+14  1.13E+14  1.15E-03  705
 Plume Sigma-y (m)                      1.0000  2.00E+02  1.50E+02    ****      ****      ****      ****   2.66E+02  1.35E-01    4
 Plume Sigma-z (m)                      1.0000  1.10E+02  4.60E+01  3.22E+02  3.78E+02    ****      ****   4.35E+02  2.75E-02   36
 Plume Height  (m)                      1.0000  5.00E+01    ****      ****      ****      ****      ****   5.00E+01  1.00E+00    1
 Plume Arrival Time (s)                 1.0000  1.47E+05  1.09E+05  1.34E+05  1.47E+05    ****      ****   1.48E+05  4.76E-02   49

Source Term 1: Plume 1, at   1.2-1.6 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  7.62E+08  7.18E+08  1.56E+09  2.04E+09  2.43E+09  2.62E+09  2.86E+09  2.29E-03  165
 Cs-137   Ground Air Conc. (Bq-s/m3)    1.0000  8.11E+08  7.50E+08  1.62E+09  2.06E+09  2.51E+09  2.73E+09  3.04E+09  2.29E-03  165
 Cs-137   Center Ground Conc. (Bq/m2)   1.0000  6.07E+06  5.67E+06  1.04E+07  1.15E+07  1.46E+07  1.61E+07  2.15E+07  1.14E-03  588
 Total    Center Ground Conc. (Bq/m2)   1.0000  1.80E+07  1.46E+07  3.20E+07  3.81E+07  5.25E+07  5.61E+07  6.47E+07  1.14E-03  588
 Ground-Level Dilution, X/Q (s/m3)      1.0000  7.74E-06  6.97E-06  1.59E-05  2.14E-05    ****      ****   3.01E-05  1.44E-02   50
 Cs-137   Adjusted Source, Q (Bq)       1.0000  1.08E+14  1.01E+14  1.04E+14  1.06E+14  1.09E+14  1.10E+14  1.13E+14  1.15E-03  705
 Plume Sigma-y (m)                      1.0000  2.73E+02  2.46E+02    ****      ****      ****      ****   4.06E+02  1.34E-01    4
 Plume Sigma-z (m)                      1.0000  1.69E+02  4.65E+01  7.04E+02  7.13E+02  7.35E+02  7.44E+02  7.55E+02  2.25E-03  312
 Plume Height  (m)                      1.0000  5.00E+01    ****      ****      ****      ****      ****   5.00E+01  1.00E+00    1
 Plume Arrival Time (s)                 1.0000  1.47E+05  1.09E+05  1.35E+05  1.48E+05    ****      ****   1.49E+05  4.76E-02   49

Source Term 1: Plume 1, at   1.6-2.1 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  6.17E+08  5.46E+08  1.15E+09  1.44E+09  2.11E+09  2.24E+09  2.40E+09  2.28E-03  578
 Cs-137   Ground Air Conc. (Bq-s/m3)    1.0000  6.82E+08  5.79E+08  1.25E+09  1.58E+09  2.24E+09  2.43E+09  2.68E+09  2.28E-03  578
 Cs-137   Center Ground Conc. (Bq/m2)   1.0000  4.78E+06  4.99E+06  9.74E+06  1.10E+07  1.38E+07  1.53E+07  1.71E+07  2.28E-03  578
 Total    Center Ground Conc. (Bq/m2)   1.0000  1.44E+07  1.16E+07  2.95E+07  3.32E+07  4.24E+07  4.72E+07  5.25E+07  2.28E-03  578
 Ground-Level Dilution, X/Q (s/m3)      1.0000  6.70E-06  5.59E-06  1.45E-05  2.04E-05  2.47E-05    ****   2.68E-05  5.02E-03   50
 Cs-137   Adjusted Source, Q (Bq)       1.0000  1.06E+14  1.01E+14  1.04E+14  1.05E+14  1.09E+14  1.10E+14  1.13E+14  1.15E-03  846
 Plume Sigma-y (m)                      1.0000  3.18E+02  2.75E+02    ****      ****      ****      ****   5.17E+02  1.27E-01    4
 Plume Sigma-z (m)                      1.0000  2.23E+02  6.67E+01  1.00E+03  1.01E+03  1.04E+03  1.05E+03  1.06E+03  2.25E-03  312
 Plume Height  (m)                      1.0000  5.00E+01    ****      ****      ****      ****      ****   5.00E+01  1.00E+00    1
 Plume Arrival Time (s)                 1.0000  1.47E+05  1.10E+05  1.36E+05  1.49E+05    ****      ****   1.50E+05  4.76E-02   49

Source Term 1: Plume 1, at   2.1-3.2 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  4.55E+08  3.53E+08  1.00E+09  1.09E+09  1.33E+09  1.45E+09  1.75E+09  1.14E-03  578
 Cs-137   Ground Air Conc. (Bq-s/m3)    1.0000  5.11E+08  4.34E+08  1.04E+09  1.15E+09  1.45E+09  1.61E+09  1.99E+09  1.14E-03  578
 Cs-137   Center Ground Conc. (Bq/m2)   1.0000  3.25E+06  3.20E+06  6.07E+06  7.11E+06  8.84E+06  9.71E+06  1.27E+07  5.99E-04  943
 Total    Center Ground Conc. (Bq/m2)   1.0000  1.00E+07  1.02E+07  2.04E+07  2.24E+07  2.81E+07  3.09E+07  4.00E+07  5.99E-04  943
 Ground-Level Dilution, X/Q (s/m3)      1.0000  5.22E-06  3.92E-06  1.09E-05  1.34E-05  2.02E-05  2.06E-05  2.16E-05  1.13E-03  441
 Cs-137   Adjusted Source, Q (Bq)       1.0000  1.03E+14  1.01E+14  1.04E+14  1.05E+14  1.08E+14  1.10E+14  1.13E+14  1.15E-03  846
 Plume Sigma-y (m)                      1.0000  3.95E+02  3.52E+02    ****      ****      ****      ****   7.01E+02  1.18E-01    4
 Plume Sigma-z (m)                      1.0000  3.23E+02  6.80E+01  1.04E+03  1.14E+03  1.42E+03  1.57E+03  1.65E+03  3.41E-03  201
 Plume Height  (m)                      1.0000  5.00E+01    ****      ****      ****      ****      ****   5.00E+01  1.00E+00    1
 Plume Arrival Time (s)                 1.0000  1.48E+05  1.09E+05  1.34E+05  1.46E+05    ****      ****   1.52E+05  3.66E-02   49

Source Term 1: Plume 1, at   3.2-4.0 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  3.37E+08  3.03E+08  7.33E+08  9.34E+08  1.10E+09  1.15E+09  1.26E+09  1.13E-03  441
 Cs-137   Ground Air Conc. (Bq-s/m3)    1.0000  3.75E+08  3.24E+08  8.13E+08  1.00E+09  1.16E+09  1.24E+09  1.43E+09  1.13E-03  441
 Cs-137   Center Ground Conc. (Bq/m2)   1.0000  2.18E+06  2.16E+06  3.73E+06  4.26E+06  5.99E+06  7.03E+06  8.95E+06  5.99E-04  943
 Total    Center Ground Conc. (Bq/m2)   1.0000  6.87E+06  7.12E+06  1.20E+07  1.35E+07  1.78E+07  2.01E+07  2.90E+07  5.99E-04  943
 Ground-Level Dilution, X/Q (s/m3)      1.0000  3.99E-06  3.21E-06  9.18E-06  1.07E-05  1.36E-05  1.51E-05  1.70E-05  2.27E-03  441
 Cs-137   Adjusted Source, Q (Bq)       1.0000  1.01E+14  1.00E+14  1.03E+14  1.05E+14  1.08E+14  1.10E+14  1.13E+14  1.15E-03  846
 Plume Sigma-y (m)                      1.0000  4.83E+02  3.66E+02  7.35E+02  8.22E+02    ****      ****   9.11E+02  2.65E-02    5
 Plume Sigma-z (m)                      1.0000  4.44E+02  9.70E+01  2.03E+03  2.15E+03    ****      ****   2.39E+03  1.40E-02   55
 Plume Height  (m)                      1.0000  5.00E+01    ****      ****      ****      ****      ****   5.00E+01  1.00E+00    1
 Plume Arrival Time (s)                 1.0000  1.49E+05  1.09E+05  1.34E+05  1.47E+05    ****      ****   1.53E+05  3.55E-02   49

Source Term 1: Plume 1, at   4.0-4.8 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  2.73E+08  2.28E+08  5.76E+08  7.10E+08  8.96E+08  9.91E+08  1.32E+09  1.14E-03  263
 Cs-137   Ground Air Conc. (Bq-s/m3)    1.0000  3.01E+08  2.50E+08  6.47E+08  7.80E+08  1.08E+09  1.19E+09  1.48E+09  1.14E-03  263
 Cs-137   Center Ground Conc. (Bq/m2)   1.0000  1.64E+06  1.46E+06  2.91E+06  3.38E+06  4.70E+06  5.37E+06  8.17E+06  1.14E-03  263
 Total    Center Ground Conc. (Bq/m2)   1.0000  5.24E+06  5.29E+06  1.01E+07  1.15E+07  1.53E+07  1.74E+07  2.60E+07  1.14E-03  263
 Ground-Level Dilution, X/Q (s/m3)      1.0000  3.29E-06  2.50E-06  7.39E-06  1.01E-05  1.21E-05  1.31E-05  1.55E-05  1.14E-03  263
 Cs-137   Adjusted Source, Q (Bq)       1.0000  9.92E+13  1.00E+14  1.03E+14  1.05E+14  1.08E+14  1.09E+14  1.13E+14  1.15E-03  846
 Plume Sigma-y (m)                      1.0000  5.55E+02  4.90E+02  1.05E+03    ****      ****      ****   1.08E+03  9.26E-02    4
 Plume Sigma-z (m)                      1.0000  5.52E+02  9.84E+01  3.00E+03  3.01E+03  3.04E+03  3.05E+03  3.06E+03  2.28E-03  181
 Plume Height  (m)                      1.0000  5.00E+01    ****      ****      ****      ****      ****   5.00E+01  1.00E+00    1
 Plume Arrival Time (s)                 1.0000  1.49E+05  1.09E+05  1.34E+05  1.46E+05    ****      ****   1.55E+05  3.16E-02   49

Source Term 1: Plume 1, at   4.8-5.6 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  2.25E+08  1.91E+08  4.72E+08  7.02E+08  7.79E+08  8.15E+08  8.97E+08  1.15E-03  787
 Cs-137   Ground Air Conc. (Bq-s/m3)    1.0000  2.46E+08  2.04E+08  5.31E+08  7.07E+08  8.16E+08  8.69E+08  9.90E+08  1.15E-03  787
 Cs-137   Center Ground Conc. (Bq/m2)   1.0000  1.27E+06  1.09E+06  2.33E+06  2.67E+06  3.69E+06  4.25E+06  6.00E+06  1.09E-03  466
 Total    Center Ground Conc. (Bq/m2)   1.0000  4.11E+06  3.83E+06  7.88E+06  9.15E+06  1.24E+07  1.40E+07  1.84E+07  1.09E-03  466
 Ground-Level Dilution, X/Q (s/m3)      1.0000  2.75E-06  2.09E-06  6.17E-06  9.79E-06  1.08E-05  1.12E-05  1.20E-05  1.14E-03  297
 Cs-137   Adjusted Source, Q (Bq)       1.0000  9.78E+13  9.52E+13  1.03E+14  1.04E+14  1.08E+14  1.09E+14  1.13E+14  1.15E-03  846
 Plume Sigma-y (m)                      1.0000  6.26E+02  5.31E+02    ****      ****      ****      ****   1.25E+03  1.03E-01    4
 Plume Sigma-z (m)                      1.0000  6.62E+02  9.97E+01  3.02E+03  3.15E+03  3.46E+03  3.61E+03  3.78E+03  2.29E-03  734
 Plume Height  (m)                      1.0000  5.00E+01    ****      ****      ****      ****      ****   5.00E+01  1.00E+00    1
 Plume Arrival Time (s)                 1.0000  1.50E+05  1.09E+05  1.35E+05  1.48E+05    ****      ****   1.57E+05  3.16E-02   49

Source Term 1: Plume 1, at   5.6-8.1 km
 Cs-137   Center Air Conc. (Bq-s/m3)    1.0000  1.61E+08  1.19E+08  3.59E+08  4.76E+08  5.55E+08  5.83E+08  6.48E+08  1.12E-03  296
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Cs-137 Ground Air Conc. (Bq-s/m3) 1.0000 1.73E+08 1.27E+08 3.80E+08 4.94E+08 5.76E+08 6.13E+08 7.03E+08 1.12E-03 296
Cs-137 Center Ground Conc. (Bq/m2) 1.0000 8.27E+05 8.16E+05 1.45E+06 1.73E+06 2.51E+06 2.93E+06 4.03E+06 1.14E-03 92
Total Center Ground Conc. (Bq/m2) 1.0000 2.73E+06 2.67E+06 5.25E+06 6.03E+06 8.43E+06 9.77E+06 1.20E+07 1.09E-03 466
Ground-Level Dilution, X/Q (s/m3) 1.0000 2.01E-06 1.37E-06 4.68E-06 6.36E-06 7.73E-06 8.13E-06 9.04E-06 1.13E-03 441
Cs-137 Adjusted Source, Q (Bq) 1.0000 9.56E+13 9.07E+13 1.03E+14 1.04E+14 1.08E+14 1.09E+14 1.12E+14 1.15E-03 846
Plume Sigma-y (m) 1.0000 7.64E+02 6.21E+02 1.10E+03 1.26E+03 **** **** 1.57E+03 1.68E-02 6
Plume Sigma-z (m) 1.0000 8.93E+02 1.93E+02 4.57E+03 5.08E+03 5.29E+03 **** 5.31E+03 8.36E-03 228
Plume Height (m) 1.0000 5.00E+01 **** **** **** **** **** 5.00E+01 1.00E+00 1
Plume Arrival Time (s) 1.0000 1.51E+05 1.09E+05 1.34E+05 1.47E+05 **** **** 1.60E+05 2.56E-02 49

Source Term 1: Plume 1, at 8.1-11.3 km
Cs-137 Center Air Conc. (Bq-s/m3) 1.0000 1.00E+08 8.00E+07 2.22E+08 2.88E+08 3.50E+08 3.76E+08 4.37E+08 1.14E-03 653
Cs-137 Ground Air Conc. (Bq-s/m3) 1.0000 1.06E+08 8.30E+07 2.32E+08 2.96E+08 3.60E+08 3.90E+08 4.64E+08 1.14E-03 653
Cs-137 Center Ground Conc. (Bq/m2) 1.0000 4.61E+05 4.44E+05 8.72E+05 1.05E+06 1.57E+06 1.87E+06 2.62E+06 1.14E-03 92
Total Center Ground Conc. (Bq/m2) 1.0000 1.55E+06 1.24E+06 3.04E+06 3.57E+06 5.16E+06 5.96E+06 7.84E+06 1.14E-03 92
Ground-Level Dilution, X/Q (s/m3) 1.0000 1.29E-06 9.37E-07 3.01E-06 4.08E-06 5.34E-06 5.55E-06 6.05E-06 1.12E-03 296
Cs-137 Adjusted Source, Q (Bq) 1.0000 9.25E+13 8.70E+13 1.02E+14 1.04E+14 1.07E+14 1.09E+14 1.12E+14 1.15E-03 846
Plume Sigma-y (m) 1.0000 9.99E+02 8.17E+02 1.86E+03 **** **** **** 2.12E+03 8.63E-02 4
Plume Sigma-z (m) 1.0000 1.33E+03 2.27E+02 6.11E+03 7.18E+03 7.70E+03 7.93E+03 8.21E+03 2.28E-03 228
Plume Height (m) 1.0000 5.00E+01 **** **** **** **** **** 5.00E+01 1.00E+00 1
Plume Arrival Time (s) 1.0000 1.52E+05 1.09E+05 1.34E+05 1.47E+05 **** **** 1.65E+05 1.96E-02 111

Source Term 1: Plume 1, at 11.3-16.1 km
Cs-137 Center Air Conc. (Bq-s/m3) 1.0000 5.92E+07 4.20E+07 1.27E+08 1.56E+08 2.19E+08 2.38E+08 2.82E+08 1.14E-03 653
Cs-137 Ground Air Conc. (Bq-s/m3) 1.0000 6.12E+07 4.45E+07 1.33E+08 1.66E+08 2.26E+08 2.46E+08 2.94E+08 1.14E-03 653
Cs-137 Center Ground Conc. (Bq/m2) 1.0000 2.49E+05 2.14E+05 4.64E+05 5.82E+05 9.70E+05 1.20E+06 1.90E+06 1.14E-03 92
Total Center Ground Conc. (Bq/m2) 1.0000 8.46E+05 7.48E+05 1.59E+06 1.99E+06 3.15E+06 3.92E+06 5.66E+06 1.14E-03 92
Ground-Level Dilution, X/Q (s/m3) 1.0000 7.80E-07 5.29E-07 1.78E-06 2.41E-06 3.23E-06 3.38E-06 3.91E-06 5.99E-04 788
Cs-137 Adjusted Source, Q (Bq) 1.0000 8.95E+13 8.43E+13 1.02E+14 1.03E+14 1.07E+14 1.08E+14 1.11E+14 1.15E-03 846
Plume Sigma-y (m) 1.0000 1.33E+03 1.02E+03 2.42E+03 **** **** **** 2.87E+03 7.13E-02 4
Plume Sigma-z (m) 1.0000 2.03E+03 3.06E+02 9.87E+03 1.06E+04 1.23E+04 **** 1.28E+04 6.85E-03 308
Plume Height (m) 1.0000 5.00E+01 **** **** **** **** **** 5.00E+01 1.00E+00 1
Plume Arrival Time (s) 1.0000 1.55E+05 1.08E+05 1.30E+05 1.41E+05 1.69E+05 **** 1.74E+05 7.94E-03 111

Source Term 1: Plume 1, at 64.4-80.5 km
Cs-137 Center Air Conc. (Bq-s/m3) 1.0000 3.00E+06 2.26E+06 5.94E+06 7.58E+06 1.14E+07 1.30E+07 1.75E+07 1.12E-03 545
Cs-137 Ground Air Conc. (Bq-s/m3) 1.0000 3.00E+06 2.26E+06 5.94E+06 7.61E+06 1.15E+07 1.31E+07 1.75E+07 1.12E-03 545
Cs-137 Center Ground Conc. (Bq/m2) 1.0000 1.30E+04 8.64E+03 2.40E+04 3.44E+04 8.26E+04 1.14E+05 2.12E+05 1.15E-03 324
Total Center Ground Conc. (Bq/m2) 1.0000 4.35E+04 3.07E+04 8.14E+04 1.11E+05 2.47E+05 3.31E+05 6.24E+05 1.15E-03 324
Ground-Level Dilution, X/Q (s/m3) 1.0000 4.48E-08 3.27E-08 9.41E-08 1.21E-07 2.04E-07 2.26E-07 2.82E-07 1.13E-03 517
Cs-137 Adjusted Source, Q (Bq) 1.0000 7.27E+13 7.12E+13 8.67E+13 9.44E+13 1.02E+14 1.03E+14 1.05E+14 1.15E-03 765
Plume Sigma-y (m) 1.0000 5.79E+03 5.17E+03 8.68E+03 1.02E+04 **** **** 1.21E+04 1.43E-02 382
Plume Sigma-z (m) 1.0000 8.41E+03 1.95E+03 2.72E+04 3.29E+04 4.37E+04 4.94E+04 5.40E+04 1.14E-03 235
Plume Height (m) 1.0000 5.00E+01 **** **** **** **** **** 5.00E+01 1.00E+00 1
Plume Arrival Time (s) 1.0000 1.88E+05 1.43E+05 2.09E+05 2.17E+05 2.34E+05 2.42E+05 2.60E+05 1.11E-03 312

Source Term 1: Plume 1, at 113-161 km
Cs-137 Center Air Conc. (Bq-s/m3) 1.0000 1.17E+06 9.85E+05 2.12E+06 2.75E+06 3.66E+06 4.05E+06 5.36E+06 1.15E-03 566
Cs-137 Ground Air Conc. (Bq-s/m3) 1.0000 1.17E+06 9.85E+05 2.12E+06 2.75E+06 3.66E+06 4.05E+06 5.36E+06 1.15E-03 566
Cs-137 Center Ground Conc. (Bq/m2) 1.0000 4.93E+03 3.18E+03 9.80E+03 1.47E+04 3.24E+04 3.71E+04 4.97E+04 1.14E-03 323
Total Center Ground Conc. (Bq/m2) 1.0000 1.64E+04 1.13E+04 3.19E+04 4.81E+04 1.02E+05 1.14E+05 1.46E+05 1.14E-03 323
Ground-Level Dilution, X/Q (s/m3) 1.0000 1.95E-08 1.47E-08 3.59E-08 4.72E-08 6.69E-08 7.32E-08 8.45E-08 1.15E-03 558
Cs-137 Adjusted Source, Q (Bq) 1.0000 6.44E+13 6.35E+13 7.59E+13 7.90E+13 8.67E+13 9.03E+13 9.83E+13 1.15E-03 765
Plume Sigma-y (m) 1.0000 1.00E+04 9.34E+03 1.30E+04 1.47E+04 1.95E+04 **** 2.10E+04 7.44E-03 382
Plume Sigma-z (m) 1.0000 1.01E+04 2.86E+03 3.13E+04 3.66E+04 5.04E+04 5.15E+04 5.39E+04 1.14E-03 140
Plume Height (m) 1.0000 5.00E+01 **** **** **** **** **** 5.00E+01 1.00E+00 1
Plume Arrival Time (s) 1.0000 2.22E+05 2.08E+05 2.44E+05 2.62E+05 3.03E+05 3.12E+05 3.31E+05 1.14E-03 515

"ATMOS" DESCRIPTION = OCP3 high density no spray
"EARLY" DESCRIPTION = OCP3 high density no spray, EARLY input
"CHRONC" DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 high density no spray

OVERALL RESULTS OBTAINED BY COMBINING12 EMERGENCY RESPONSE COHORTS FROM "EARLY" WITH THE WEIGHTING FRACTIONS BELOW APPLIED TO THEM:

	FRACTION OF THE SUMPOP

COHORT 1 = 0-10 Schools	0.000
COHORT 2 = 0-10 Early Evacuees	0.000
COHORT 3 = 0-10 Public	0.000
COHORT 4 = 10-20 Shadow	0.000
COHORT 5 = 0-10 Special Facilities	0.000
COHORT 6 = 0-10 Evacuation Tail	0.000
COHORT 7 = 10-30 Public	0.000
COHORT 8 = 10-30 Special Facilities	0.000
COHORT 9 = 30-40 Shadow	0.000
COHORT10 = 10-30 Tail	0.000
COHORT11 = 30-40 Shelter in Place	0.000
COHORT12 = Nonevacuees	0.000

AND THEN MERGING THE12 RESULTS ABOVE WITH THE SINGLE SET OF RESULTS FROM "CHRONC" DESCRIBED BELOW:

COHORT13 = OCP3 high density no spray, EARLY input

RESULTS WHICH ARE PRODUCED ONLY BY "EARLY" OR ONLY BY "CHRONC" ARE PRESENTED IN LATER SECTIONS.

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	PEAK 99.5TH	PEAK CONSEQ	PEAK PROB	TRIAL
HEALTH EFFECTS CASES											
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	1.0000	2.41E+02	2.19E+02	3.32E+02	3.62E+02	4.44E+02	4.85E+02	6.44E+02	1.12E-03	464
CAN INJ/TOTAL	0-32.2 km	1.0000	1.21E+03	1.07E+03	1.66E+03	2.00E+03	2.32E+03	2.47E+03	2.84E+03	1.12E-03	391
CAN INJ/TOTAL	0-48.3 km	1.0000	2.96E+03	2.83E+03	4.18E+03	4.86E+03	5.54E+03	5.81E+03	6.44E+03	1.14E-03	7
CAN INJ/TOTAL	0-64.4 km	1.0000	6.22E+03	5.56E+03	1.01E+04	1.08E+04	1.26E+04	1.34E+04	1.54E+04	1.14E-03	390
CAN INJ/TOTAL	0-80.5 km	1.0000	9.18E+03	8.36E+03	1.22E+04	1.37E+04	1.77E+04	1.97E+04	2.10E+04	1.14E-03	390

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 CAN INJ/TOTAL          0-161 km  1.0000  2.66E+04  2.51E+04  4.01E+04  4.64E+04  5.53E+04  5.84E+04  6.56E+04  1.14E-03  388
 CAN INJ/TOTAL          0-322 km  1.0000  6.01E+04  5.49E+04  1.05E+05  1.13E+05  1.33E+05  1.43E+05  1.67E+05  1.14E-03  388
 CAN INJ/TOTAL          0-805 km  1.0000  9.31E+04  8.80E+04  1.38E+05  1.60E+05  2.11E+05  2.25E+05  2.58E+05  1.14E-03  234
 CAN INJ/TOTAL         0-1609 km  1.0000  1.01E+05  1.00E+05  1.50E+05  1.79E+05  2.35E+05  2.59E+05  3.17E+05  1.13E-03  396
 CAN FAT/TOTAL          0-16.1 km  1.0000  1.06E+02  1.01E+02  1.37E+02  1.56E+02  2.08E+02  2.29E+02  2.80E+02  1.12E-03  464
 CAN FAT/TOTAL          0-32.2 km  1.0000  5.29E+02  5.09E+02  7.74E+02  8.58E+02  1.05E+03  1.11E+03  1.24E+03  1.12E-03  391
 CAN FAT/TOTAL          0-48.3 km  1.0000  1.30E+03  1.12E+03  1.88E+03  2.08E+03  2.37E+03  2.50E+03  2.82E+03  1.14E-03    7
 CAN FAT/TOTAL          0-64.4 km  1.0000  2.73E+03  2.41E+03  4.17E+03  5.02E+03  5.70E+03  6.02E+03  6.77E+03  1.14E-03  390
 CAN FAT/TOTAL          0-80.5 km  1.0000  4.03E+03  3.58E+03  6.26E+03  7.12E+03  7.95E+03  8.34E+03  9.23E+03  1.14E-03  390
 CAN FAT/TOTAL           0-161 km  1.0000  1.17E+04  1.07E+04  1.70E+04  2.03E+04  2.35E+04  2.51E+04  2.88E+04  1.14E-03  388
 CAN FAT/TOTAL           0-322 km  1.0000  2.63E+04  2.39E+04  4.32E+04  5.06E+04  5.81E+04  6.16E+04  7.32E+04  1.14E-03  388
 CAN FAT/TOTAL           0-805 km  1.0000  4.07E+04  3.71E+04  6.86E+04  7.67E+04  9.74E+04  1.03E+05  1.13E+05  1.14E-03  234
 CAN FAT/TOTAL          0-1609 km  1.0000  4.43E+04  4.13E+04  7.39E+04  8.39E+04  1.07E+05  1.16E+05  1.36E+05  1.13E-03  396
 CAN FAT/THYROID         0-16.1 km  1.0000  5.79E-01  5.37E-01  8.10E-01  9.01E-01  1.12E+00  1.21E+00  1.45E+00  1.12E-03  464
 CAN FAT/THYROID         0-80.5 km  1.0000  2.49E+01  2.26E+01  3.55E+01  3.98E+01  5.07E+01  5.27E+01  5.74E+01  1.14E-03  390
 CAN FAT/THYROID          0-161 km  1.0000  7.26E+01  7.19E+01  1.07E+02  1.16E+02  1.40E+02  1.51E+02  1.80E+02  1.14E-03  388
 CAN FAT/THYROID         0-1609 km  1.0000  2.72E+02  2.60E+02  4.27E+02  5.03E+02  6.36E+02  7.01E+02  8.03E+02  5.99E-04  397
 CAN FAT/BREAST          0-16.1 km  1.0000  7.97E+00  7.41E+00  1.10E+01  1.19E+01  1.45E+01  1.57E+01  1.88E+01  1.12E-03  464
 CAN FAT/BREAST          0-80.5 km  1.0000  3.82E+02  3.43E+02  5.93E+02  6.89E+02  7.68E+02  8.01E+02  8.76E+02  1.14E-03  390
 CAN FAT/BREAST           0-161 km  1.0000  1.11E+03  1.05E+03  1.63E+03  1.97E+03  2.28E+03  2.42E+03  2.74E+03  1.14E-03  387
 CAN FAT/BREAST          0-1609 km  1.0000  4.12E+03  3.80E+03  6.78E+03  7.59E+03  9.58E+03  1.02E+04  1.13E+04  5.99E-04  397
 CAN FAT/LUNG           0-16.1 km  1.0000  1.70E+01  1.44E+01  2.36E+01  2.63E+01  3.29E+01  3.57E+01  4.26E+01  1.12E-03  464
 CAN FAT/LUNG           0-80.5 km  1.0000  7.34E+02  6.74E+02  1.07E+03  1.15E+03  1.36E+03  1.46E+03  1.70E+03  1.14E-03  390
 CAN FAT/LUNG            0-161 km  1.0000  2.14E+03  2.09E+03  3.23E+03  3.59E+03  4.59E+03  5.03E+03  5.35E+03  1.14E-03  388
 CAN FAT/LUNG           0-1609 km  1.0000  7.95E+03  7.59E+03  1.23E+04  1.39E+04  1.84E+04  2.03E+04  2.29E+04  5.99E-04  397
 CAN FAT/LEUKEMIA          0-1609 km  1.0000  4.54E+03  4.29E+03  7.51E+03  8.47E+03  1.06E+04  1.12E+04  1.34E+04  5.99E-04  397
 CAN FAT/BONE           0-1609 km  1.0000  1.09E+02  1.03E+02  1.64E+02  2.00E+02  2.32E+02  2.47E+02  3.01E+02  5.99E-04  397
 CAN FAT/LIVER          0-1609 km  1.0000  1.15E+03  1.05E+03  1.85E+03  2.14E+03  2.72E+03  3.01E+03  3.51E+03  5.99E-04  397
 CAN FAT/COLON           0-1609 km  1.0000  8.34E+03  7.93E+03  1.32E+04  1.53E+04  2.09E+04  2.27E+04  2.71E+04  1.13E-03  396
 CAN FAT/RESIDUAL          0-1609 km  1.0000  1.78E+04  1.59E+04  2.98E+04  3.37E+04  4.46E+04  5.01E+04  5.74E+04  1.13E-03  396

                          PROB               QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
EARLY FATALITY DISTANCE (km)
 ERL FAT/TOTAL RISK > 0.000        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB               QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
POPULATION EXCEEDING DOSE
 EARLY dose A-RED MARR > 2.32 Sv       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-LUNGS > 13.6 Sv        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-STOMACH > 6.50 Sv       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB               QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
POPULATION DOSE (Sv)
 L-ICRP60ED  TOT LIF       0-16.1 km  1.0000  5.41E+03  5.16E+03  7.00E+03  7.59E+03  9.14E+03  9.91E+03  1.11E+04  1.14E-03  465
 L-ICRP60ED  TOT LIF       0-80.5 km  1.0000  7.96E+04  7.35E+04  1.10E+05  1.18E+05  1.40E+05  1.51E+05  1.76E+05  1.14E-03  390
 L-ICRP60ED  TOT LIF        0-161 km  1.0000  2.25E+05  2.15E+05  3.35E+05  3.74E+05  4.82E+05  5.11E+05  5.46E+05  1.14E-03  387
 L-ICRP60ED  TOT LIF       0-1609 km  1.0000  8.56E+05  8.17E+05  1.28E+06  1.45E+06  1.94E+06  2.09E+06  2.50E+06  5.99E-04  397

                          PROB               QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
POPULATION WEIGHTED RISK
 CAN FAT/TOTAL          0-16.1 km  1.0000  5.56E-04  4.93E-04  1.00E-03  1.08E-03  1.27E-03  1.37E-03  1.59E-03  1.14E-03  776
 CAN FAT/TOTAL          0-32.2 km  1.0000  8.02E-04  7.44E-04  1.17E-03  1.31E-03  1.69E-03  1.88E-03  2.18E-03  1.12E-03  391
 CAN FAT/TOTAL          0-48.3 km  1.0000  7.63E-04  7.36E-04  1.08E-03  1.15E-03  1.35E-03  1.44E-03  1.67E-03  1.14E-03  245
 CAN FAT/TOTAL          0-64.4 km  1.0000  6.94E-04  6.21E-04  1.07E-03  1.16E-03  1.39E-03  1.50E-03  1.78E-03  1.14E-03  390
 CAN FAT/TOTAL          0-80.5 km  1.0000  6.67E-04  6.07E-04  1.04E-03  1.11E-03  1.28E-03  1.36E-03  1.55E-03  1.14E-03  388
 CAN FAT/TOTAL           0-161 km  1.0000  5.84E-04  5.59E-04  9.09E-04  1.03E-03  1.19E-03  1.27E-03  1.46E-03  1.14E-03  387
 CAN FAT/TOTAL           0-322 km  1.0000  5.19E-04  4.64E-04  8.86E-04  1.01E-03  1.18E-03  1.26E-03  1.45E-03  1.14E-03  388
 CAN FAT/TOTAL           0-805 km  1.0000  3.33E-04  3.11E-04  5.56E-04  6.16E-04  7.46E-04  7.92E-04  8.99E-04  1.14E-03  234
 CAN FAT/TOTAL          0-1609 km  1.0000  1.98E-04  2.01E-04  3.19E-04  3.42E-04  4.02E-04  4.31E-04  5.08E-04  1.14E-03  234
 CAN FAT/TOTAL         16.1-32.2 km  1.0000  8.28E-04  7.66E-04  1.20E-03  1.35E-03  1.77E-03  1.99E-03  2.28E-03  1.12E-03  391
 CAN FAT/TOTAL         32.2-48.3 km  1.0000  7.43E-04  7.19E-04  1.08E-03  1.15E-03  1.36E-03  1.46E-03  1.70E-03  1.14E-03  245
 CAN FAT/TOTAL         48.3-64.4 km  1.0000  6.48E-04  5.46E-04  1.09E-03  1.20E-03  1.49E-03  1.63E-03  2.02E-03  1.13E-03  714
 CAN FAT/TOTAL         64.4-80.5 km  1.0000  6.21E-04  5.84E-04  1.01E-03  1.07E-03  1.21E-03  1.29E-03  1.45E-03  1.15E-03   72
 CAN FAT/TOTAL          80.5-161 km  1.0000  5.51E-04  5.39E-04  8.91E-04  1.02E-03  1.19E-03  1.27E-03  1.48E-03  1.14E-03  387
 CAN FAT/TOTAL           161-322 km  1.0000  4.77E-04  3.54E-04  1.02E-03  1.09E-03  1.26E-03  1.34E-03  1.53E-03  1.14E-03  718
 CAN FAT/TOTAL           322-805 km  0.9897  1.84E-04  1.48E-04  3.66E-04  4.46E-04  5.64E-04  6.04E-04  8.13E-04  1.13E-03  968
 CAN FAT/TOTAL          805-1609 km  0.9165  2.42E-05  3.85E-07  8.47E-05  1.41E-04  3.07E-04  3.44E-04  4.42E-04  1.13E-03  877

                          PROB               QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)
 L-ICRP60ED             0-0.2 km  1.0000  2.02E-01  1.65E-01  3.17E-01  3.59E-01  4.79E-01  5.80E-01  8.75E-01  1.11E-03  392
 L-ICRP60ED           0.2-0.5 km  1.0000  1.32E-01  1.15E-01  1.91E-01  2.17E-01  2.79E-01  3.02E-01  3.36E-01  1.52E-04  416
 L-ICRP60ED           0.5-1.2 km  1.0000  1.07E-01  1.05E-01  1.49E-01  1.74E-01  2.06E-01  2.11E-01  2.37E-01  1.52E-04  415
 L-ICRP60ED           1.2-1.6 km  1.0000  9.61E-02  9.25E-02  1.18E-01  1.28E-01  1.54E-01  1.67E-01  1.98E-01  1.13E-03  314
 L-ICRP60ED           1.6-2.1 km  1.0000  9.08E-02  8.54E-02  1.09E-01  1.15E-01  1.29E-01  1.36E-01  1.77E-01  1.52E-04  415
 L-ICRP60ED           2.1-3.2 km  1.0000  8.69E-02  8.00E-02  1.06E-01  1.11E-01  1.26E-01  1.33E-01  1.49E-01  1.13E-03  374
 L-ICRP60ED           3.2-4.0 km  1.0000  8.45E-02  7.71E-02  1.01E-01  1.05E-01  1.15E-01  1.20E-01  1.30E-01  1.15E-03  552
 L-ICRP60ED           4.0-4.8 km  1.0000  8.32E-02  7.62E-02  9.97E-02  1.03E-01  1.10E-01  1.13E-01  1.21E-01  1.15E-03  552
 L-ICRP60ED           4.8-5.6 km  1.0000  8.51E-02  7.69E-02  1.00E-01  1.03E-01  1.08E-01  1.11E-01  1.16E-01  1.15E-03  552
 L-ICRP60ED           5.6-8.1 km  1.0000  8.68E-02  7.77E-02  1.01E-01  1.06E-01  1.18E-01  1.23E-01  1.36E-01  1.14E-03  830
 L-ICRP60ED          8.1-11.3 km  1.0000  8.82E-02  7.89E-02  1.06E-01  1.15E-01  1.42E-01  1.56E-01  1.89E-01  1.12E-03  607
 L-ICRP60ED         11.3-16.1 km  1.0000  9.20E-02  8.37E-02  1.09E-01  1.16E-01  1.33E-01  1.42E-01  1.62E-01  1.12E-03  607
 L-ICRP60ED         16.1-20.9 km  1.0000  9.46E-02  8.81E-02  1.07E-01  1.12E-01  1.22E-01  1.27E-01  1.38E-01  1.14E-03  518
 L-ICRP60ED         20.9-25.8 km  1.0000  9.51E-02  8.75E-02  1.08E-01  1.13E-01  1.25E-01  1.31E-01  1.44E-01  1.14E-03  829
 L-ICRP60ED         25.8-32.2 km  1.0000  9.37E-02  8.43E-02  1.05E-01  1.09E-01  1.18E-01  1.22E-01  1.31E-01  1.14E-03  829
 L-ICRP60ED         32.2-40.2 km  1.0000  8.91E-02  7.96E-02  1.01E-01  1.03E-01  1.07E-01  1.09E-01  1.13E-01  1.12E-03  577
 L-ICRP60ED         40.2-48.3 km  1.0000  8.24E-02  7.51E-02  9.16E-02  9.98E-02  1.04E-01  1.06E-01  1.09E-01  1.14E-03  567
 L-ICRP60ED         48.3-64.4 km  1.0000  7.27E-02  7.03E-02  8.06E-02  8.55E-02  9.81E-02  1.01E-01  1.05E-01  1.13E-03  517
 L-ICRP60ED         64.4-80.5 km  1.0000  6.61E-02  5.95E-02  7.40E-02  7.70E-02  8.45E-02  8.79E-02  9.56E-02  1.14E-03  395
 L-ICRP60ED          80.5-113 km  1.0000  6.07E-02  5.53E-02  7.07E-02  7.38E-02  8.15E-02  8.51E-02  9.33E-02  1.14E-03  461
 L-ICRP60ED           113-161 km  1.0000  5.53E-02  5.27E-02  6.30E-02  6.80E-02  7.64E-02  7.99E-02  8.80E-02  1.14E-03  319
 L-ICRP60ED           161-241 km  1.0000  5.00E-02  5.02E-02  5.65E-02  5.95E-02  6.69E-02  7.04E-02  7.84E-02  1.14E-03  458
 L-ICRP60ED           241-322 km  1.0000  4.51E-02  3.95E-02  5.22E-02  5.36E-02  5.71E-02  5.87E-02  6.23E-02  1.14E-03  305
 L-ICRP60ED           322-563 km  1.0000  3.92E-02  3.42E-02  5.04E-02  5.19E-02  5.54E-02  5.70E-02  6.22E-02  5.99E-04  313
 L-ICRP60ED           563-805 km  1.0000  3.18E-02  3.11E-02  4.39E-02  5.02E-02  5.30E-02  5.42E-02  5.70E-02  1.13E-03  815
 L-ICRP60ED          805-1609 km  1.0000  1.01E-02  2.76E-03  3.04E-02  3.33E-02  4.12E-02  4.51E-02  5.08E-02  1.13E-03  827

                          PROB               QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)

                          PROB               QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ     PROB  TRIAL
GROUND CONC. (Bq/m2)
 AREA (ha) THAT EXCEEDS THRESHOLD
```

Cs-137 Area exceeds 3.70E+04 Bq/m2 1.0000 3.91E+04 3.52E+04 5.35E+04 5.65E+04 6.43E+04 6.79E+04 7.06E+04 1.14E-03 944
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.85E+05 Bq/m2 1.0000 3.82E+04 3.45E+04 5.32E+04 5.63E+04 6.41E+04 6.79E+04 7.06E+04 1.14E-03 944
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 5.55E+05 Bq/m2 1.0000 3.71E+04 3.37E+04 5.25E+04 5.53E+04 6.25E+04 6.59E+04 7.06E+04 1.14E-03 944
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.48E+06 Bq/m2 1.0000 3.56E+04 3.26E+04 5.15E+04 5.40E+04 6.03E+04 6.32E+04 7.06E+04 1.14E-03 944
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 3.70E+04 Bq/m2 1.0000 8.83E+05 8.30E+05 1.14E+06 1.22E+06 1.44E+06 1.55E+06 1.81E+06 1.14E-03 930
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.85E+05 Bq/m2 1.0000 8.16E+05 7.65E+05 1.11E+06 1.19E+06 1.40E+06 1.51E+06 1.76E+06 1.14E-03 930
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 5.55E+05 Bq/m2 1.0000 7.53E+05 7.22E+05 1.06E+06 1.14E+06 1.33E+06 1.43E+06 1.65E+06 1.14E-03 387
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.48E+06 Bq/m2 1.0000 6.56E+05 6.19E+05 9.42E+05 1.04E+06 1.21E+06 1.30E+06 1.49E+06 1.14E-03 387
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 3.70E+04 Bq/m2 1.0000 3.22E+06 3.07E+06 4.65E+06 5.17E+06 5.88E+06 6.22E+06 7.23E+06 1.14E-03 930
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.85E+05 Bq/m2 1.0000 2.88E+06 2.73E+06 3.95E+06 4.49E+06 5.50E+06 5.86E+06 6.71E+06 1.14E-03 930
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 5.55E+05 Bq/m2 1.0000 2.44E+06 2.26E+06 3.31E+06 3.57E+06 4.26E+06 4.59E+06 5.83E+06 1.14E-03 387
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.48E+06 Bq/m2 1.0000 1.79E+06 1.52E+06 2.52E+06 2.88E+06 3.44E+06 3.68E+06 4.26E+06 1.14E-03 387
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 3.70E+04 Bq/m2 1.0000 3.73E+07 3.36E+07 6.60E+07 7.41E+07 8.98E+07 9.75E+07 1.07E+08 5.99E-04 943
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.85E+05 Bq/m2 1.0000 2.32E+07 2.06E+07 4.10E+07 4.99E+07 6.12E+07 6.68E+07 8.13E+07 1.15E-03 911
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 5.55E+05 Bq/m2 1.0000 1.21E+07 9.66E+06 2.24E+07 2.53E+07 3.09E+07 3.19E+07 3.43E+07 1.14E-03 924
AREA (ha) THAT EXCEEDS THRESHOLD
Cs-137 Area exceeds 1.48E+06 Bq/m2 1.0000 4.88E+06 3.91E+06 8.79E+06 1.03E+07 1.31E+07 1.46E+07 1.82E+07 1.14E-03 179

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 high density no spray
"EARLY" DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 1 = 0-10 Schools

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
HEALTH EFFECTS CASES
ERL FAT/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
ERL FAT/TOTAL 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
ERL FAT/TOTAL 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-32.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-48.3 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-64.4 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-161 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-322 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-805 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-32.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-48.3 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-64.4 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-161 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-322 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-805 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/THYROID 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/THYROID 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/THYROID 0-161 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/THYROID 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/BREAST 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/BREAST 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/BREAST 0-161 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/BREAST 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/LUNG 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/LUNG 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/LUNG 0-161 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/LUNG 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/LEUKEMIA 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/BONE 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/LIVER 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/COLON 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/RESIDUAL 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
EARLY FATALITY DISTANCE (km)
ERL FAT/TOTAL RISK > 0.000 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
POPULATION EXCEEDING DOSE
EARLY dose A-RED MARR > 2.32 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
EARLY dose A-LUNGS > 13.6 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
EARLY dose A-STOMACH > 6.50 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
POPULATION DOSE (Sv)
L-ICRP60ED TOT LIF 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED TOT LIF 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED TOT LIF 0-161 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED TOT LIF 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
POPULATION WEIGHTED RISK

CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	16.1-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	32.2-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	48.3-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	64.4-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	80.5-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	161-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	322-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	805-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	PEAK 99.5TH	PEAK CONSEQ	PEAK PROB	TRIAL
L-ICRP60ED	0-0.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.2-0.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.5-1.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.2-1.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.6-2.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	2.1-3.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	3.2-4.0 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.0-4.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.8-5.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	5.6-8.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	8.1-11.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	11.3-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	16.1-20.9 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	20.9-25.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	25.8-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	32.2-40.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	40.2-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	48.3-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	64.4-80.5 km	1.0000	8.53E-03	7.51E-03	9.70E-03	1.14E-02	1.76E-02	2.15E-02	3.50E-02	1.11E-03	312
L-ICRP60ED	80.5-113 km	1.0000	6.64E-03	6.43E-03	8.39E-03	9.14E-03	1.29E-02	1.59E-02	2.15E-02	1.14E-03	518
L-ICRP60ED	113-161 km	1.0000	4.59E-03	3.98E-03	7.32E-03	7.85E-03	9.23E-03	9.90E-03	1.31E-02	1.14E-03	518
L-ICRP60ED	161-241 km	1.0000	2.70E-03	2.26E-03	5.00E-03	6.03E-03	7.62E-03	8.06E-03	9.29E-03	8.56E-04	456
L-ICRP60ED	241-322 km	1.0000	1.71E-03	1.29E-03	3.15E-03	3.95E-03	6.66E-03	7.34E-03	8.35E-03	1.14E-03	513
L-ICRP60ED	322-563 km	1.0000	8.66E-04	7.21E-04	1.55E-03	2.02E-03	3.21E-03	3.69E-03	5.01E-03	1.11E-03	312
L-ICRP60ED	563-805 km	1.0000	4.54E-04	3.60E-04	8.66E-04	1.08E-03	1.84E-03	2.79E-03	3.65E-03	1.13E-03	334
L-ICRP60ED	805-1609 km	1.0000	7.93E-05	2.07E-05	2.25E-04	2.83E-04	3.99E-04	4.58E-04	7.83E-04	3.04E-04	632

DOSE FOUND AT ALL LOCATIONS (Sv)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	PEAK 99.5TH	PEAK CONSEQ	PEAK PROB	TRIAL
AREA (ha) THAT EXCEEDS THRESHOLD										
L-ICRP60ED Area exceeds 1.00E-02 Sv	0.0643	1.58E+03	0.00E+00	0.00E+00	1.15E+04	4.04E+04	5.22E+04	1.06E+05	1.14E-03	518
AREA (ha) THAT EXCEEDS THRESHOLD										
L-ICRP60ED Area exceeds 5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD										
A-THYROID Area exceeds 5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 high density no spray
"EARLY" DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 2 = 0-10 Early Evacuees

HEALTH EFFECTS CASES

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	PEAK 99.5TH	PEAK CONSEQ	PEAK PROB	TRIAL
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LEUKEMIA	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BONE	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LIVER	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

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 CAN FAT/COLON             0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/RESIDUAL          0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                              PROB                    QUANTILES                     PEAK      PEAK  PEAK
                            NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
EARLY FATALITY DISTANCE (km)
 ERL FAT/TOTAL RISK > 0.000            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                              PROB                    QUANTILES                     PEAK      PEAK  PEAK
                            NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
POPULATION EXCEEDING DOSE
 EARLY dose A-RED MARR > 2.32 Sv       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-LUNGS > 13.6 Sv          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-STOMACH > 6.50 Sv        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                              PROB                    QUANTILES                     PEAK      PEAK  PEAK
                            NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
POPULATION DOSE (Sv)
 L-ICRP60ED  TOT LIF       0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF       0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF        0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF       0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                              PROB                    QUANTILES                     PEAK      PEAK  PEAK
                            NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
POPULATION WEIGHTED RISK
 CAN FAT/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          16.1-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          32.2-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          48.3-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          64.4-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           80.5-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            161-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            322-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           805-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                              PROB                    QUANTILES                     PEAK      PEAK  PEAK
                            NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)
 L-ICRP60ED                  0-0.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED                0.2-0.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED                0.5-1.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED                1.2-1.6 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED                1.6-2.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED                2.1-3.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED                3.2-4.0 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED                4.0-4.8 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED                4.8-5.6 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED                5.6-8.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               8.1-11.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED              11.3-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED              16.1-20.9 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED              20.9-25.8 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED              25.8-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED              32.2-40.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED              40.2-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED              48.3-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED              64.4-80.5 km  1.0000  8.56E-03  7.52E-03  9.72E-03  1.15E-02  1.80E-02  2.19E-02  3.54E-02  1.11E-03  312
 L-ICRP60ED               80.5-113 km  1.0000  6.67E-03  6.47E-03  8.40E-03  9.14E-03  1.36E-02  1.75E-02  2.17E-02  1.14E-03  518
 L-ICRP60ED                113-161 km  1.0000  4.62E-03  4.01E-03  7.34E-03  7.86E-03  9.24E-03  9.90E-03  1.33E-02  1.14E-03  518
 L-ICRP60ED                161-241 km  1.0000  2.72E-03  2.28E-03  5.02E-03  6.13E-03  7.67E-03  8.11E-03  9.33E-03  8.56E-04  456
 L-ICRP60ED                241-322 km  1.0000  1.72E-03  1.30E-03  3.17E-03  3.99E-03  6.68E-03  7.35E-03  8.41E-03  1.14E-03  513
 L-ICRP60ED                322-563 km  1.0000  8.73E-04  7.29E-04  1.56E-03  2.03E-03  3.21E-03  3.69E-03  5.03E-03  1.11E-03  312
 L-ICRP60ED                563-805 km  1.0000  4.57E-04  3.62E-04  8.76E-04  1.10E-03  1.86E-03  2.81E-03  3.66E-03  1.13E-03  334
 L-ICRP60ED               805-1609 km  1.0000  7.97E-05  2.07E-05  2.27E-04  2.90E-04  4.01E-04  4.59E-04  7.88E-04  3.04E-04  632

                              PROB                    QUANTILES                     PEAK      PEAK  PEAK
                            NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   1.00E-02 Sv  0.0654  1.67E+03  0.00E+00  0.00E+00  1.17E+04  4.04E+04  5.38E+04  1.16E+05  1.13E-03  517
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 AREA (ha) THAT EXCEEDS THRESHOLD
 A-THYROID  Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

 **** Indicates that the value is outside resolution of the analysis.
 Optionally increase number of trials for better resolution.

  "ATMOS"  DESCRIPTION = OCP3 high density no spray
  "EARLY"  DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM  1 OF  1:
   OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

   COHORT 3 = 0-10 Public

                              PROB                    QUANTILES                     PEAK      PEAK  PEAK
                            NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
HEALTH EFFECTS CASES
 ERL FAT/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL             0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL             0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
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 CAN INJ/TOTAL          0-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL         0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          0-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          0-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID       0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID       0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID        0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID       0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST        0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST        0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST         0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST        0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG          0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG          0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG           0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG          0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LEUKEMIA      0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BONE          0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LIVER         0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/COLON         0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/RESIDUAL      0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                           PROB          QUANTILES                PEAK     PEAK  PEAK
                         NON-ZERO  MEAN    50TH      90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
EARLY FATALITY DISTANCE (km)
 ERL FAT/TOTAL RISK > 0.000        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                           PROB          QUANTILES                PEAK     PEAK  PEAK
                         NON-ZERO  MEAN    50TH      90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
POPULATION EXCEEDING DOSE
 EARLY dose A-RED MARR > 2.32 Sv     0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-LUNGS > 13.6 Sv        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-STOMACH > 6.50 Sv      0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                           PROB          QUANTILES                PEAK     PEAK  PEAK
                         NON-ZERO  MEAN    50TH      90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
POPULATION DOSE (Sv)
 L-ICRP60ED  TOT LIF    0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF    0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF     0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF    0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                           PROB          QUANTILES                PEAK     PEAK  PEAK
                         NON-ZERO  MEAN    50TH      90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
POPULATION WEIGHTED RISK
 CAN FAT/TOTAL         0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          0-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          0-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL      16.1-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL      32.2-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL      48.3-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL      64.4-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL       80.5-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL        161-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL        322-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL       805-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                           PROB          QUANTILES                PEAK     PEAK  PEAK
                         NON-ZERO  MEAN    50TH      90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)
 L-ICRP60ED              0-0.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            0.2-0.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            0.5-1.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            1.2-1.6 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            1.6-2.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            2.1-3.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            3.2-4.0 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            4.0-4.8 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            4.8-5.6 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            5.6-8.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED           8.1-11.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED          11.3-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED          16.1-20.9 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED          20.9-25.8 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED          25.8-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED          32.2-40.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED          40.2-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED          48.3-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED          64.4-80.5 km  1.0000  8.53E-03  7.51E-03  9.70E-03  1.14E-02  1.76E-02  2.15E-02  3.50E-02  1.11E-03  312
 L-ICRP60ED           80.5-113 km  1.0000  6.64E-03  6.43E-03  8.39E-03  9.14E-03  1.29E-02  1.59E-02  2.15E-02  1.14E-03  518
 L-ICRP60ED            113-161 km  1.0000  4.59E-03  3.98E-03  7.32E-03  7.85E-03  9.23E-03  9.90E-03  1.31E-02  1.14E-03  518
 L-ICRP60ED            161-241 km  1.0000  2.70E-03  2.26E-03  5.00E-03  6.03E-03  7.62E-03  8.06E-03  9.29E-03  8.56E-04  456
 L-ICRP60ED            241-322 km  1.0000  1.71E-03  1.29E-03  3.15E-03  3.95E-03  6.66E-03  7.34E-03  8.35E-03  1.14E-03  513
 L-ICRP60ED            322-563 km  1.0000  8.66E-04  7.21E-04  1.55E-03  2.02E-03  3.21E-03  3.69E-03  5.01E-03  1.11E-03  312
 L-ICRP60ED            563-805 km  1.0000  4.54E-04  3.60E-04  8.66E-04  1.08E-03  1.84E-03  2.79E-03  3.65E-03  1.13E-03  334
 L-ICRP60ED           805-1609 km  1.0000  7.93E-05  2.07E-05  2.25E-04  2.83E-04  3.99E-04  4.58E-04  7.83E-04  3.04E-04  632

                           PROB          QUANTILES                PEAK     PEAK  PEAK
                         NON-ZERO  MEAN    50TH      90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   1.00E-02 Sv  0.0643  1.58E+03  0.00E+00  0.00E+00  1.15E+04  4.04E+04  5.22E+04  1.06E+05  1.14E-03  518
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 AREA (ha) THAT EXCEEDS THRESHOLD
 A-THYROID  Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
```

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 high density no spray
"EARLY" DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 4 = 10-20 Shadow

HEALTH EFFECTS CASES

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LEUKEMIA	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BONE	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LIVER	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/COLON	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/RESIDUAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

EARLY FATALITY DISTANCE (km)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL RISK > 0.000	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION EXCEEDING DOSE

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
EARLY dose A-RED MARR > 2.32 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-LUNGS > 13.6 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-STOMACH > 6.50 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION DOSE (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED TOT LIF	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED TOT LIF	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED TOT LIF	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED TOT LIF	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION WEIGHTED RISK

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	16.1-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	32.2-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	48.3-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	64.4-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	80.5-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	161-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	322-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	805-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.2-0.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.5-1.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.2-1.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.6-2.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	2.1-3.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	3.2-4.0 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.0-4.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.8-5.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	5.6-8.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	8.1-11.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

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 L-ICRP60ED            11.3-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            16.1-20.9 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            20.9-25.8 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            25.8-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            32.2-40.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            40.2-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            48.3-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED            64.4-80.5 km  1.0000  8.56E-03  7.52E-03  9.72E-03  1.15E-02  1.80E-02  2.19E-02  3.54E-02  1.11E-03  312
 L-ICRP60ED             80.5-113 km  1.0000  6.67E-03  6.47E-03  8.40E-03  9.14E-03  1.36E-02  1.75E-02  2.17E-02  1.14E-03  518
 L-ICRP60ED              113-161 km  1.0000  4.62E-03  4.01E-03  7.34E-03  7.86E-03  9.24E-03  9.90E-03  1.33E-02  1.14E-03  518
 L-ICRP60ED              161-241 km  1.0000  2.72E-03  2.28E-03  5.02E-03  6.13E-03  7.67E-03  8.11E-03  9.33E-03  8.56E-04  456
 L-ICRP60ED              241-322 km  1.0000  1.72E-03  1.30E-03  3.17E-03  3.99E-03  6.68E-03  7.35E-03  8.41E-03  1.14E-03  513
 L-ICRP60ED              322-563 km  1.0000  8.73E-04  7.29E-04  1.56E-03  2.03E-03  3.21E-03  3.69E-03  5.03E-03  1.11E-03  312
 L-ICRP60ED              563-805 km  1.0000  4.57E-04  3.62E-04  8.76E-04  1.10E-03  1.86E-03  2.81E-03  3.66E-03  1.13E-03  334
 L-ICRP60ED             805-1609 km  1.0000  7.97E-05  2.07E-05  2.27E-04  2.90E-04  4.01E-04  4.59E-04  7.88E-04  3.04E-04  632

                          PROB                QUANTILES                 PEAK    PEAK  PEAK
                         NON-ZERO  MEAN    50TH    90TH    95TH    99TH   99.5TH   CONSEQ    PROB  TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   1.00E-02 Sv  0.0654  1.67E+03  0.00E+00  0.00E+00  1.17E+04  4.04E+04  5.38E+04  1.16E+05  1.13E-03  517
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 AREA (ha) THAT EXCEEDS THRESHOLD
 A-THYROID  Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

 **** Indicates that the value is outside resolution of the analysis.
 Optionally increase number of trials for better resolution.

  "ATMOS"  DESCRIPTION = OCP3 high density no spray
  "EARLY"  DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM  1 OF  1:
   OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

   COHORT 5 = 0-10 Special Facilities

                          PROB                QUANTILES                 PEAK    PEAK  PEAK
                         NON-ZERO  MEAN    50TH    90TH    95TH    99TH   99.5TH   CONSEQ    PROB  TRIAL
HEALTH EFFECTS CASES
 ERL FAT/TOTAL            0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL            0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL            0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL            0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL            0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL            0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL            0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL            0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL            0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID          0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID          0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID           0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID          0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST           0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST           0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST            0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST           0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG             0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG              0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG             0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LEUKEMIA         0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BONE             0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LIVER            0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/COLON            0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/RESIDUAL         0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB                QUANTILES                 PEAK    PEAK  PEAK
                         NON-ZERO  MEAN    50TH    90TH    95TH    99TH   99.5TH   CONSEQ    PROB  TRIAL
EARLY FATALITY DISTANCE (km)
 ERL FAT/TOTAL RISK > 0.000          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB                QUANTILES                 PEAK    PEAK  PEAK
                         NON-ZERO  MEAN    50TH    90TH    95TH    99TH   99.5TH   CONSEQ    PROB  TRIAL
POPULATION EXCEEDING DOSE
 EARLY dose A-RED MARR > 2.32 Sv     0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-LUNGS > 13.6 Sv        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-STOMACH > 6.50 Sv      0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB                QUANTILES                 PEAK    PEAK  PEAK
                         NON-ZERO  MEAN    50TH    90TH    95TH    99TH   99.5TH   CONSEQ    PROB  TRIAL
POPULATION DOSE (Sv)
 L-ICRP60ED  TOT LIF      0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF      0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF       0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF      0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB                QUANTILES                 PEAK    PEAK  PEAK
                         NON-ZERO  MEAN    50TH    90TH    95TH    99TH   99.5TH   CONSEQ    PROB  TRIAL
POPULATION WEIGHTED RISK
 CAN FAT/TOTAL            0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
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		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	16.1-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	32.2-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	48.3-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	64.4-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	80.5-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	161-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	322-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	805-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.2-0.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	0.5-1.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.2-1.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	1.6-2.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	2.1-3.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	3.2-4.0 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.0-4.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	4.8-5.6 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	5.6-8.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	8.1-11.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	11.3-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	16.1-20.9 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	20.9-25.8 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	25.8-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	32.2-40.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	40.2-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	48.3-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED	64.4-80.5 km	1.0000	6.34E-03	5.96E-03	8.50E-03	9.40E-03	1.40E-02	1.70E-02	2.50E-02	1.11E-03	312
L-ICRP60ED	80.5-113 km	1.0000	4.39E-03	3.72E-03	7.25E-03	8.04E-03	1.02E-02	1.17E-02	1.53E-02	1.14E-03	518
L-ICRP60ED	113-161 km	1.0000	2.78E-03	2.30E-03	4.76E-03	6.16E-03	7.93E-03	8.49E-03	9.83E-03	1.14E-03	513
L-ICRP60ED	161-241 km	1.0000	1.61E-03	1.24E-03	2.88E-03	3.55E-03	5.27E-03	5.77E-03	7.15E-03	1.15E-03	330
L-ICRP60ED	241-322 km	1.0000	1.01E-03	8.29E-04	1.85E-03	2.41E-03	3.52E-03	3.94E-03	5.36E-03	1.14E-03	513
L-ICRP60ED	322-563 km	1.0000	5.10E-04	4.16E-04	9.15E-04	1.16E-03	1.97E-03	2.11E-03	2.68E-03	3.04E-04	635
L-ICRP60ED	563-805 km	1.0000	2.40E-04	2.03E-04	4.58E-04	5.85E-04	1.02E-03	1.21E-03	1.77E-03	1.13E-03	334
L-ICRP60ED	805-1609 km	1.0000	4.34E-05	1.08E-05	1.19E-04	1.48E-04	2.33E-04	2.72E-04	4.59E-04	3.04E-04	632

DOSE FOUND AT ALL LOCATIONS (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	1.00E-02 Sv	0.0262	4.23E+02	0.00E+00	0.00E+00	0.00E+00	1.52E+04	3.15E+04	4.20E+04	2.27E-03	517
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
AREA (ha) THAT EXCEEDS THRESHOLD											
A-THYROID Area exceeds	5.00E-02 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 high density no spray
"EARLY" DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 6 = 0-10 Evacuation Tail

HEALTH EFFECTS CASES

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LEUKEMIA	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BONE	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LIVER	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/COLON	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/RESIDUAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL

EARLY FATALITY DISTANCE (km)

ERL FAT/TOTAL RISK > 0.000 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
POPULATION EXCEEDING DOSE
EARLY dose A-RED MARR > 2.32 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
EARLY dose A-LUNGS > 13.6 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
EARLY dose A-STOMACH > 6.50 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
POPULATION DOSE (Sv)
L-ICRP60ED TOT LIF 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED TOT LIF 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED TOT LIF 0-161 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED TOT LIF 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
POPULATION WEIGHTED RISK
CAN FAT/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-32.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-48.3 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-64.4 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-161 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-322 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-805 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 16.1-32.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 32.2-48.3 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 48.3-64.4 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 64.4-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 80.5-161 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 161-322 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 322-805 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 805-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)
L-ICRP60ED 0-0.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 0.2-0.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 0.5-1.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 1.2-1.6 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 1.6-2.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 2.1-3.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 3.2-4.0 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 4.0-4.8 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 4.8-5.6 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 5.6-8.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 8.1-11.3 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 11.3-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 16.1-20.9 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 20.9-25.8 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 25.8-32.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 32.2-40.2 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 40.2-48.3 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 48.3-64.4 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
L-ICRP60ED 64.4-80.5 km 1.0000 8.56E-03 7.52E-03 9.72E-03 1.15E-02 1.80E-02 2.19E-02 3.54E-02 1.11E-03 312
L-ICRP60ED 80.5-113 km 1.0000 6.67E-03 6.47E-03 8.40E-03 9.14E-03 1.36E-02 1.75E-02 2.17E-02 1.14E-03 518
L-ICRP60ED 113-161 km 1.0000 4.62E-03 4.01E-03 7.34E-03 7.86E-03 9.24E-03 9.90E-03 1.33E-02 1.14E-03 518
L-ICRP60ED 161-241 km 1.0000 2.72E-03 2.28E-03 5.02E-03 6.13E-03 7.67E-03 8.11E-03 9.33E-03 8.56E-04 456
L-ICRP60ED 241-322 km 1.0000 1.72E-03 1.30E-03 3.17E-03 3.99E-03 6.68E-03 7.35E-03 8.41E-03 1.14E-03 513
L-ICRP60ED 322-563 km 1.0000 8.73E-04 7.29E-04 1.56E-03 2.03E-03 3.21E-03 3.69E-03 5.03E-03 1.11E-03 312
L-ICRP60ED 563-805 km 1.0000 4.57E-04 3.62E-04 8.76E-04 1.10E-03 1.86E-03 2.81E-03 3.66E-03 1.13E-03 334
L-ICRP60ED 805-1609 km 1.0000 7.97E-05 2.07E-05 2.27E-04 2.90E-04 4.01E-04 4.59E-04 7.88E-04 3.04E-04 632

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)
AREA (ha) THAT EXCEEDS THRESHOLD
L-ICRP60ED Area exceeds 1.00E-02 Sv 0.0654 1.67E+03 0.00E+00 0.00E+00 1.17E+04 4.04E+04 5.38E+04 1.16E+05 1.13E-03 517
AREA (ha) THAT EXCEEDS THRESHOLD
L-ICRP60ED Area exceeds 5.00E-02 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
AREA (ha) THAT EXCEEDS THRESHOLD
A-THYROID Area exceeds 5.00E-02 Sv 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 high density no spray
"EARLY" DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 7 = 10-30 Public

PROB QUANTILES PEAK PEAK PEAK
NON-ZERO MEAN 50TH 90TH 95TH 99TH 99.5TH CONSEQ PROB TRIAL
HEALTH EFFECTS CASES
ERL FAT/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
ERL FAT/TOTAL 0-80.5 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
ERL FAT/TOTAL 0-1609 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN INJ/TOTAL 0-32.2 km 0.6734 5.62E-01 3.19E-02 5.65E-01 1.48E+00 1.20E+01 1.89E+01 5.08E+01 1.15E-03 410
CAN INJ/TOTAL 0-48.3 km 0.6957 8.05E-01 5.08E-02 1.01E+00 1.95E+00 1.65E+01 2.50E+01 5.39E+01 1.14E-03 693
CAN INJ/TOTAL 0-64.4 km 0.6957 8.05E-01 5.08E-02 1.01E+00 1.95E+00 1.65E+01 2.50E+01 5.39E+01 1.14E-03 693
CAN INJ/TOTAL 0-80.5 km 0.6957 8.05E-01 5.08E-02 1.01E+00 1.95E+00 1.65E+01 2.50E+01 5.39E+01 1.14E-03 693
CAN INJ/TOTAL 0-161 km 0.6957 8.05E-01 5.08E-02 1.01E+00 1.95E+00 1.65E+01 2.50E+01 5.39E+01 1.14E-03 693
CAN INJ/TOTAL 0-322 km 0.6957 8.05E-01 5.08E-02 1.01E+00 1.95E+00 1.65E+01 2.50E+01 5.39E+01 1.14E-03 693
CAN INJ/TOTAL 0-805 km 0.6957 8.05E-01 5.08E-02 1.01E+00 1.95E+00 1.65E+01 2.50E+01 5.39E+01 1.14E-03 693
CAN INJ/TOTAL 0-1609 km 0.6957 8.05E-01 5.08E-02 1.01E+00 1.95E+00 1.65E+01 2.50E+01 5.39E+01 1.14E-03 693
CAN FAT/TOTAL 0-16.1 km 0.0000 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0.00E+00 0
CAN FAT/TOTAL 0-32.2 km 0.6734 2.66E-01 1.54E-02 2.82E-01 6.78E-01 5.82E+00 9.31E+00 2.38E+01 1.15E-03 410
CAN FAT/TOTAL 0-48.3 km 0.6957 3.83E-01 2.61E-02 4.95E-01 9.77E-01 7.54E+00 1.31E+01 2.52E+01 1.14E-03 693
CAN FAT/TOTAL 0-64.4 km 0.6957 3.83E-01 2.61E-02 4.95E-01 9.77E-01 7.54E+00 1.31E+01 2.52E+01 1.14E-03 693

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 CAN FAT/TOTAL            0-80.5 km  0.6957  3.83E-01  2.61E-02  4.95E-01  9.77E-01  7.54E+00  1.31E+01  2.52E+01  1.14E-03  693
 CAN FAT/TOTAL             0-161 km  0.6957  3.83E-01  2.61E-02  4.95E-01  9.77E-01  7.54E+00  1.31E+01  2.52E+01  1.14E-03  693
 CAN FAT/TOTAL             0-322 km  0.6957  3.83E-01  2.61E-02  4.95E-01  9.77E-01  7.54E+00  1.31E+01  2.52E+01  1.14E-03  693
 CAN FAT/TOTAL             0-805 km  0.6957  3.83E-01  2.61E-02  4.95E-01  9.77E-01  7.54E+00  1.31E+01  2.52E+01  1.14E-03  693
 CAN FAT/TOTAL            0-1609 km  0.6957  3.83E-01  2.61E-02  4.95E-01  9.77E-01  7.54E+00  1.31E+01  2.52E+01  1.14E-03  693
 CAN FAT/THYROID          0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID          0-80.5 km  0.6957  2.39E-03  2.01E-04  3.33E-03  6.57E-03  4.57E-02  6.79E-02  1.49E-01  1.14E-03  693
 CAN FAT/THYROID           0-161 km  0.6957  2.39E-03  2.01E-04  3.33E-03  6.57E-03  4.57E-02  6.79E-02  1.49E-01  1.14E-03  693
 CAN FAT/THYROID          0-1609 km  0.6957  2.39E-03  2.01E-04  3.33E-03  6.57E-03  4.57E-02  6.79E-02  1.49E-01  1.14E-03  693
 CAN FAT/BREAST           0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST           0-80.5 km  0.6957  2.05E-02  1.17E-03  2.41E-02  4.85E-02  4.30E-01  6.73E-01  1.42E+00  1.14E-03  693
 CAN FAT/BREAST            0-161 km  0.6957  2.05E-02  1.17E-03  2.41E-02  4.85E-02  4.30E-01  6.73E-01  1.42E+00  1.14E-03  693
 CAN FAT/BREAST           0-1609 km  0.6957  2.05E-02  1.17E-03  2.41E-02  4.85E-02  4.30E-01  6.73E-01  1.42E+00  1.14E-03  693
 CAN FAT/LUNG             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG             0-80.5 km  0.6957  6.85E-02  4.31E-03  8.17E-02  1.71E-01  1.47E+00  2.26E+00  4.72E+00  1.14E-03  693
 CAN FAT/LUNG              0-161 km  0.6957  6.85E-02  4.31E-03  8.17E-02  1.71E-01  1.47E+00  2.26E+00  4.72E+00  1.14E-03  693
 CAN FAT/LUNG             0-1609 km  0.6957  6.85E-02  4.31E-03  8.17E-02  1.71E-01  1.47E+00  2.26E+00  4.72E+00  1.14E-03  693
 CAN FAT/LEUKEMIA         0-1609 km  0.6957  8.22E-02  7.15E-03  1.38E-01  2.70E-01  1.62E+00  2.50E+00  4.70E+00  1.15E-03  410
 CAN FAT/BONE             0-1609 km  0.6957  2.66E-03  2.33E-04  4.40E-03  8.73E-03  5.16E-02  6.56E-02  1.47E-01  1.15E-03  410
 CAN FAT/LIVER            0-1609 km  0.6957  7.47E-03  4.39E-04  8.69E-03  1.82E-02  1.63E-01  2.50E-01  5.16E-01  1.14E-03  693
 CAN FAT/COLON            0-1609 km  0.6957  6.90E-02  4.29E-03  8.45E-02  1.82E-01  1.38E+00  2.00E+00  4.62E+00  1.14E-03  693
 CAN FAT/RESIDUAL         0-1609 km  0.6957  1.31E-01  7.51E-03  1.56E-01  2.99E-01  2.64E+00  4.71E+00  9.00E+00  2.30E-03  693

                             PROB              QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH   CONSEQ    PROB  TRIAL
EARLY FATALITY DISTANCE (km)
 ERL FAT/TOTAL RISK > 0.000          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                             PROB              QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH   CONSEQ    PROB  TRIAL
POPULATION EXCEEDING DOSE
 EARLY dose A-RED MARR > 2.32 Sv      0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-LUNGS > 13.6 Sv         0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-STOMACH > 6.50 Sv       0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                             PROB              QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH   CONSEQ    PROB  TRIAL
POPULATION DOSE (Sv)
 L-ICRP60ED  TOT LIF      0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF      0-80.5 km  0.6957  6.92E+00  4.93E-01  9.05E+00  1.93E+01  1.38E+02  2.00E+02  4.46E+02  1.14E-03  693
 L-ICRP60ED  TOT LIF       0-161 km  0.6957  6.92E+00  4.93E-01  9.05E+00  1.93E+01  1.38E+02  2.00E+02  4.46E+02  1.14E-03  693
 L-ICRP60ED  TOT LIF      0-1609 km  0.6957  6.92E+00  4.93E-01  9.05E+00  1.93E+01  1.38E+02  2.00E+02  4.46E+02  1.14E-03  693

                             PROB              QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH   CONSEQ    PROB  TRIAL
POPULATION WEIGHTED RISK
 CAN FAT/TOTAL            0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            0-32.2 km  0.6734  1.16E-06  7.07E-08  1.24E-06  2.90E-06  2.54E-05  4.01E-05  1.04E-04  1.15E-03  410
 CAN FAT/TOTAL            0-48.3 km  0.6957  3.79E-07  2.55E-08  4.84E-07  9.76E-07  7.54E-06  1.31E-05  2.49E-05  1.14E-03  693
 CAN FAT/TOTAL            0-64.4 km  0.6957  3.79E-07  2.55E-08  4.84E-07  9.76E-07  7.54E-06  1.31E-05  2.49E-05  1.14E-03  693
 CAN FAT/TOTAL            0-80.5 km  0.6957  3.79E-07  2.55E-08  4.84E-07  9.76E-07  7.54E-06  1.31E-05  2.49E-05  1.14E-03  693
 CAN FAT/TOTAL             0-161 km  0.6957  3.79E-07  2.55E-08  4.84E-07  9.76E-07  7.54E-06  1.31E-05  2.49E-05  1.14E-03  693
 CAN FAT/TOTAL             0-322 km  0.6957  3.79E-07  2.55E-08  4.84E-07  9.76E-07  7.54E-06  1.31E-05  2.49E-05  1.14E-03  693
 CAN FAT/TOTAL             0-805 km  0.6957  3.79E-07  2.55E-08  4.84E-07  9.76E-07  7.54E-06  1.31E-05  2.49E-05  1.14E-03  693
 CAN FAT/TOTAL            0-1609 km  0.6957  3.79E-07  2.55E-08  4.84E-07  9.76E-07  7.54E-06  1.31E-05  2.49E-05  1.14E-03  693
 CAN FAT/TOTAL         16.1-32.2 km  0.6734  1.16E-06  7.07E-08  1.24E-06  2.90E-06  2.54E-05  4.01E-05  1.04E-04  1.15E-03  410
 CAN FAT/TOTAL         32.2-48.3 km  0.4831  1.50E-07  0.00E+00  2.84E-07  4.66E-07  3.59E-06  5.74E-06  1.40E-05  1.52E-04  507
 CAN FAT/TOTAL         48.3-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL         64.4-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          80.5-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           161-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           322-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          805-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                             PROB              QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH   CONSEQ    PROB  TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)
 L-ICRP60ED                0-0.2 km  0.6392  1.26E-03  1.38E-04  2.44E-03  6.10E-03  1.89E-02  3.10E-02  4.77E-02  1.14E-03  486
 L-ICRP60ED              0.2-0.5 km  0.6329  1.20E-03  1.28E-04  2.32E-03  6.34E-03  1.77E-02  3.10E-02  4.70E-02  1.14E-03  486
 L-ICRP60ED              0.5-1.2 km  0.6249  1.07E-03  1.09E-04  2.08E-03  5.76E-03  1.57E-02  2.19E-02  3.65E-02  1.09E-03  287
 L-ICRP60ED              1.2-1.6 km  0.6128  9.89E-04  1.03E-04  1.66E-03  4.84E-03  1.53E-02  2.46E-02  3.48E-02  1.14E-03  693
 L-ICRP60ED              1.6-2.1 km  0.6594  1.11E-03  1.57E-04  1.75E-03  5.29E-03  1.76E-02  3.21E-02  4.51E-02  1.09E-03   97
 L-ICRP60ED              2.1-3.2 km  0.6465  1.09E-03  1.43E-04  1.58E-03  4.79E-03  2.05E-02  3.23E-02  4.71E-02  1.09E-03   97
 L-ICRP60ED              3.2-4.0 km  0.7523  1.14E-03  2.36E-04  1.93E-03  4.55E-03  1.76E-02  3.13E-02  4.82E-02  1.14E-03  123
 L-ICRP60ED              4.0-4.8 km  0.7472  8.95E-04  2.29E-04  1.61E-03  3.22E-03  1.50E-02  2.26E-02  4.45E-02  1.52E-04   95
 L-ICRP60ED              4.8-5.6 km  0.7447  7.84E-04  2.10E-04  1.35E-03  2.70E-03  1.12E-02  1.87E-02  4.51E-02  1.52E-04   95
 L-ICRP60ED              5.6-8.1 km  0.7132  7.53E-04  1.52E-04  1.26E-03  3.06E-03  1.23E-02  1.91E-02  3.18E-02  1.52E-04   95
 L-ICRP60ED             8.1-11.3 km  0.6943  7.63E-04  1.22E-04  1.04E-03  3.35E-03  1.38E-02  2.08E-02  3.83E-02  1.15E-03  654
 L-ICRP60ED            11.3-16.1 km  0.6574  4.20E-04  8.49E-05  6.35E-04  1.26E-03  7.00E-03  9.44E-03  3.36E-02  1.09E-03   97
 L-ICRP60ED            16.1-20.9 km  0.6393  4.07E-04  6.05E-05  4.90E-04  9.00E-04  8.04E-03  1.37E-02  3.38E-02  1.15E-03  410
 L-ICRP60ED            20.9-25.8 km  0.6229  4.10E-04  4.30E-05  4.45E-04  9.82E-04  7.05E-03  1.26E-02  4.57E-02  1.15E-03  410
 L-ICRP60ED            25.8-32.2 km  0.5594  3.52E-04  1.78E-05  3.53E-04  8.13E-04  8.00E-03  1.20E-02  3.20E-02  1.15E-03  410
 L-ICRP60ED            32.2-40.2 km  0.4768  7.66E-05  0.00E+00  1.96E-04  2.54E-04  4.87E-04  2.09E-03  5.81E-03  1.52E-04  507
 L-ICRP60ED            40.2-48.3 km  0.3801  8.88E-05  0.00E+00  1.09E-04  1.77E-04  2.51E-03  3.69E-03  7.09E-03  1.14E-03  693
 L-ICRP60ED            48.3-64.4 km  0.3472  5.41E-05  0.00E+00  8.10E-05  1.24E-04  2.08E-03  2.77E-03  4.79E-03  3.04E-04  650
 L-ICRP60ED            64.4-80.5 km  1.0000  8.56E-03  7.52E-03  9.72E-03  1.15E-02  1.80E-02  2.19E-02  3.54E-02  1.11E-03  312
 L-ICRP60ED             80.5-113 km  1.0000  6.67E-03  6.47E-03  8.40E-03  9.14E-03  1.36E-02  1.75E-02  2.17E-02  1.14E-03  518
 L-ICRP60ED              113-161 km  1.0000  4.62E-03  4.01E-03  7.34E-03  7.86E-03  9.24E-03  9.90E-03  1.33E-02  1.14E-03  518
 L-ICRP60ED              161-241 km  1.0000  2.72E-03  2.28E-03  5.02E-03  6.13E-03  7.67E-03  8.11E-03  9.33E-03  8.56E-04  456
 L-ICRP60ED              241-322 km  1.0000  1.72E-03  1.30E-03  3.17E-03  3.99E-03  6.68E-03  7.35E-03  8.41E-03  1.14E-03  513
 L-ICRP60ED              322-563 km  1.0000  8.73E-04  7.29E-04  1.56E-03  2.03E-03  3.21E-03  3.69E-03  5.03E-03  1.11E-03  312
 L-ICRP60ED              563-805 km  1.0000  4.57E-04  3.62E-04  8.76E-04  1.10E-03  1.86E-03  2.81E-03  3.66E-03  1.13E-03  334
 L-ICRP60ED             805-1609 km  1.0000  7.97E-05  2.07E-05  2.27E-04  2.90E-04  4.01E-04  4.59E-04  7.88E-04  3.04E-04  632

                             PROB              QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN      50TH      90TH      95TH      99TH     99.5TH   CONSEQ    PROB  TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   1.00E-02 Sv  0.1149  1.75E+03  0.00E+00  5.46E+01  1.20E+04  4.04E+04  5.38E+04  1.16E+05  1.13E-03  517
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 AREA (ha) THAT EXCEEDS THRESHOLD
 A-THYROID  Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

 **** Indicates that the value is outside resolution of the analysis.
 Optionally increase number of trials for better resolution.

  "ATMOS"  DESCRIPTION = OCP3 high density no spray
  "EARLY"  DESCRIPTION = OCP3 high density no spray, EARLY input
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SOURCE TERM 1 OF 1:
OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT 8 = 10-30 Special Facilities

HEALTH EFFECTS CASES

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-32.2 km	0.0613	1.41E-04	0.00E+00	0.00E+00	2.95E-05	3.32E-03	8.66E-03	2.11E-02	1.14E-03	528
CAN INJ/TOTAL	0-48.3 km	0.0637	1.92E-04	0.00E+00	0.00E+00	4.71E-05	4.54E-03	1.11E-02	3.69E-02	1.14E-03	528
CAN INJ/TOTAL	0-64.4 km	0.0637	1.92E-04	0.00E+00	0.00E+00	4.71E-05	4.54E-03	1.11E-02	3.69E-02	1.14E-03	528
CAN INJ/TOTAL	0-80.5 km	0.0637	1.92E-04	0.00E+00	0.00E+00	4.71E-05	4.54E-03	1.11E-02	3.69E-02	1.14E-03	528
CAN INJ/TOTAL	0-161 km	0.0637	1.92E-04	0.00E+00	0.00E+00	4.71E-05	4.54E-03	1.11E-02	3.69E-02	1.14E-03	528
CAN INJ/TOTAL	0-322 km	0.0637	1.92E-04	0.00E+00	0.00E+00	4.71E-05	4.54E-03	1.11E-02	3.69E-02	1.14E-03	528
CAN INJ/TOTAL	0-805 km	0.0637	1.92E-04	0.00E+00	0.00E+00	4.71E-05	4.54E-03	1.11E-02	3.69E-02	1.14E-03	528
CAN INJ/TOTAL	0-1609 km	0.0637	1.92E-04	0.00E+00	0.00E+00	4.71E-05	4.54E-03	1.11E-02	3.69E-02	1.14E-03	528
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	0.0613	7.06E-05	0.00E+00	0.00E+00	1.48E-05	1.57E-03	4.46E-03	1.05E-02	1.14E-03	528
CAN FAT/TOTAL	0-48.3 km	0.0637	9.63E-05	0.00E+00	0.00E+00	2.13E-05	2.44E-03	6.11E-03	1.84E-02	1.14E-03	528
CAN FAT/TOTAL	0-64.4 km	0.0637	9.63E-05	0.00E+00	0.00E+00	2.13E-05	2.44E-03	6.11E-03	1.84E-02	1.14E-03	528
CAN FAT/TOTAL	0-80.5 km	0.0637	9.63E-05	0.00E+00	0.00E+00	2.13E-05	2.44E-03	6.11E-03	1.84E-02	1.14E-03	528
CAN FAT/TOTAL	0-161 km	0.0637	9.63E-05	0.00E+00	0.00E+00	2.13E-05	2.44E-03	6.11E-03	1.84E-02	1.14E-03	528
CAN FAT/TOTAL	0-322 km	0.0637	9.63E-05	0.00E+00	0.00E+00	2.13E-05	2.44E-03	6.11E-03	1.84E-02	1.14E-03	528
CAN FAT/TOTAL	0-805 km	0.0637	9.63E-05	0.00E+00	0.00E+00	2.13E-05	2.44E-03	6.11E-03	1.84E-02	1.14E-03	528
CAN FAT/TOTAL	0-1609 km	0.0637	9.63E-05	0.00E+00	0.00E+00	2.13E-05	2.44E-03	6.11E-03	1.84E-02	1.14E-03	528
CAN FAT/THYROID	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/THYROID	0-80.5 km	0.0637	7.79E-07	0.00E+00	0.00E+00	2.09E-07	2.22E-05	4.63E-05	1.31E-04	1.14E-03	528
CAN FAT/THYROID	0-161 km	0.0637	7.79E-07	0.00E+00	0.00E+00	2.09E-07	2.22E-05	4.63E-05	1.31E-04	1.14E-03	528
CAN FAT/THYROID	0-1609 km	0.0637	7.79E-07	0.00E+00	0.00E+00	2.09E-07	2.22E-05	4.63E-05	1.31E-04	1.14E-03	528
CAN FAT/BREAST	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-80.5 km	0.0637	4.42E-06	0.00E+00	0.00E+00	9.85E-07	9.57E-05	2.55E-04	8.72E-04	1.14E-03	528
CAN FAT/BREAST	0-161 km	0.0637	4.42E-06	0.00E+00	0.00E+00	9.85E-07	9.57E-05	2.55E-04	8.72E-04	1.14E-03	528
CAN FAT/BREAST	0-1609 km	0.0637	4.42E-06	0.00E+00	0.00E+00	9.85E-07	9.57E-05	2.55E-04	8.72E-04	1.14E-03	528
CAN FAT/LUNG	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-80.5 km	0.0637	1.61E-05	0.00E+00	0.00E+00	4.02E-06	4.52E-04	1.06E-03	2.95E-03	1.14E-03	528
CAN FAT/LUNG	0-161 km	0.0637	1.61E-05	0.00E+00	0.00E+00	4.02E-06	4.52E-04	1.06E-03	2.95E-03	1.14E-03	528
CAN FAT/LUNG	0-1609 km	0.0637	1.61E-05	0.00E+00	0.00E+00	4.02E-06	4.52E-04	1.06E-03	2.95E-03	1.14E-03	528
CAN FAT/LEUKEMIA	0-1609 km	0.0637	2.77E-05	0.00E+00	0.00E+00	6.90E-06	6.55E-04	1.80E-03	5.24E-03	1.14E-03	528
CAN FAT/BONE	0-1609 km	0.0637	9.61E-07	0.00E+00	0.00E+00	2.16E-07	2.44E-05	6.11E-05	1.81E-04	1.14E-03	528
CAN FAT/LIVER	0-1609 km	0.0637	1.62E-06	0.00E+00	0.00E+00	3.21E-07	3.85E-05	9.24E-05	3.19E-04	1.14E-03	528
CAN FAT/COLON	0-1609 km	0.0637	1.63E-05	0.00E+00	0.00E+00	3.25E-06	4.50E-04	1.11E-03	3.15E-03	1.14E-03	528
CAN FAT/RESIDUAL	0-1609 km	0.0637	2.84E-05	0.00E+00	0.00E+00	6.63E-06	6.55E-04	1.80E-03	5.58E-03	1.14E-03	528

EARLY FATALITY DISTANCE (km)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL RISK > 0.000	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION EXCEEDING DOSE

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
EARLY dose A-RED MARR > 2.32 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-LUNGS > 13.6 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-STOMACH > 6.50 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION DOSE (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED TOT LIF	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED TOT LIF	0-80.5 km	0.0637	1.82E-03	0.00E+00	0.00E+00	4.60E-04	4.52E-02	1.11E-01	3.47E-01	1.14E-03	528
L-ICRP60ED TOT LIF	0-161 km	0.0637	1.82E-03	0.00E+00	0.00E+00	4.60E-04	4.52E-02	1.11E-01	3.47E-01	1.14E-03	528
L-ICRP60ED TOT LIF	0-1609 km	0.0637	1.82E-03	0.00E+00	0.00E+00	4.60E-04	4.52E-02	1.11E-01	3.47E-01	1.14E-03	528

POPULATION WEIGHTED RISK

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	0.0613	3.97E-09	0.00E+00	0.00E+00	8.73E-10	9.26E-08	2.55E-07	5.92E-07	1.14E-03	528
CAN FAT/TOTAL	0-48.3 km	0.0637	1.68E-09	0.00E+00	0.00E+00	3.59E-10	4.52E-08	1.11E-07	3.21E-07	1.14E-03	528
CAN FAT/TOTAL	0-64.4 km	0.0637	1.68E-09	0.00E+00	0.00E+00	3.59E-10	4.52E-08	1.11E-07	3.21E-07	1.14E-03	528
CAN FAT/TOTAL	0-80.5 km	0.0637	1.68E-09	0.00E+00	0.00E+00	3.59E-10	4.52E-08	1.11E-07	3.21E-07	1.14E-03	528
CAN FAT/TOTAL	0-161 km	0.0637	1.68E-09	0.00E+00	0.00E+00	3.59E-10	4.52E-08	1.11E-07	3.21E-07	1.14E-03	528
CAN FAT/TOTAL	0-322 km	0.0637	1.68E-09	0.00E+00	0.00E+00	3.59E-10	4.52E-08	1.11E-07	3.21E-07	1.14E-03	528
CAN FAT/TOTAL	0-805 km	0.0637	1.68E-09	0.00E+00	0.00E+00	3.59E-10	4.52E-08	1.11E-07	3.21E-07	1.14E-03	528
CAN FAT/TOTAL	0-1609 km	0.0637	1.68E-09	0.00E+00	0.00E+00	3.59E-10	4.52E-08	1.11E-07	3.21E-07	1.14E-03	528
CAN FAT/TOTAL	16.1-32.2 km	0.0613	3.97E-09	0.00E+00	0.00E+00	8.73E-10	9.26E-08	2.55E-07	5.92E-07	1.14E-03	528
CAN FAT/TOTAL	32.2-48.3 km	0.0225	6.48E-10	0.00E+00	0.00E+00	0.00E+00	2.34E-08	3.55E-08	2.00E-07	1.14E-03	528
CAN FAT/TOTAL	48.3-64.4 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	64.4-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	80.5-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	161-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	322-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	805-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	0.2102	3.50E-05	0.00E+00	8.51E-05	2.02E-04	5.95E-04	8.54E-04	2.35E-03	3.23E-04	91
L-ICRP60ED	0.2-0.5 km	0.2021	3.04E-05	0.00E+00	7.78E-05	1.71E-04	4.45E-04	6.89E-04	2.09E-03	3.23E-04	91
L-ICRP60ED	0.5-1.2 km	0.1794	2.34E-05	0.00E+00	5.02E-05	1.37E-04	4.15E-04	5.54E-04	2.08E-03	1.14E-03	715
L-ICRP60ED	1.2-1.6 km	0.1695	2.09E-05	0.00E+00	4.00E-05	1.24E-04	4.50E-04	5.59E-04	1.05E-03	3.23E-04	91
L-ICRP60ED	1.6-2.1 km	0.1970	2.63E-05	0.00E+00	4.83E-05	1.38E-04	5.87E-04	7.56E-04	1.23E-03	1.14E-03	85
L-ICRP60ED	2.1-3.2 km	0.1946	2.38E-05	0.00E+00	4.35E-05	1.05E-04	5.68E-04	9.15E-04	1.32E-03	3.71E-04	99
L-ICRP60ED	3.2-4.0 km	0.2288	2.56E-05	0.00E+00	5.05E-05	1.07E-04	5.49E-04	7.33E-04	1.53E-03	3.71E-04	99
L-ICRP60ED	4.0-4.8 km	0.2215	2.09E-05	0.00E+00	4.74E-05	1.10E-04	3.52E-04	5.28E-04	1.35E-03	1.15E-03	478
L-ICRP60ED	4.8-5.6 km	0.2224	1.68E-05	0.00E+00	4.68E-05	9.46E-05	2.89E-04	3.86E-04	6.20E-04	3.71E-04	487
L-ICRP60ED	5.6-8.1 km	0.1594	1.33E-05	0.00E+00	2.36E-05	5.72E-05	3.22E-04	4.54E-04	9.02E-04	1.15E-03	694
L-ICRP60ED	8.1-11.3 km	0.1087	7.00E-06	0.00E+00	2.78E-06	2.49E-05	2.19E-04	3.60E-04	5.41E-04	1.15E-03	694
L-ICRP60ED	11.3-16.1 km	0.0722	2.73E-06	0.00E+00	0.00E+00	9.32E-06	6.32E-05	1.58E-04	2.86E-04	3.23E-04	489
L-ICRP60ED	16.1-20.9 km	0.0408	1.29E-06	0.00E+00	0.00E+00	0.00E+00	4.57E-05	7.84E-05	1.21E-04	1.52E-04	95
L-ICRP60ED	20.9-25.8 km	0.0427	2.48E-06	0.00E+00	0.00E+00	0.00E+00	5.49E-05	1.07E-04	5.33E-04	1.15E-03	694
L-ICRP60ED	25.8-32.2 km	0.0298	1.57E-06	0.00E+00	0.00E+00	0.00E+00	4.12E-05	8.79E-05	4.12E-04	1.14E-03	528
L-ICRP60ED	32.2-40.2 km	0.0212	4.18E-07	0.00E+00	0.00E+00	0.00E+00	1.63E-05	2.61E-05	6.23E-05	3.23E-04	96
L-ICRP60ED	40.2-48.3 km	0.0057	2.68E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.05E-06	1.12E-04	1.15E-03	694

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 L-ICRP60ED            48.3-64.4 km  0.0194  4.42E-07  0.00E+00  0.00E+00  0.00E+00  1.32E-05  3.44E-05  7.39E-05  1.13E-03  687
 L-ICRP60ED            64.4-80.5 km  1.0000  6.34E-03  5.96E-03  8.50E-03  9.40E-03  1.40E-02  1.70E-02  2.50E-02  1.11E-03  312
 L-ICRP60ED             80.5-113 km  1.0000  4.39E-03  3.72E-03  7.25E-03  8.04E-03  1.02E-02  1.17E-02  1.53E-02  1.14E-03  518
 L-ICRP60ED              113-161 km  1.0000  2.78E-03  2.30E-03  4.76E-03  6.16E-03  7.93E-03  8.49E-03  9.83E-03  1.14E-03  513
 L-ICRP60ED              161-241 km  1.0000  1.61E-03  1.24E-03  2.88E-03  3.55E-03  5.27E-03  5.77E-03  7.15E-03  1.15E-03  330
 L-ICRP60ED              241-322 km  1.0000  1.01E-03  8.29E-04  1.85E-03  2.41E-03  3.52E-03  3.94E-03  5.36E-03  1.14E-03  513
 L-ICRP60ED              322-563 km  1.0000  5.10E-04  4.16E-04  9.15E-04  1.16E-03  1.97E-03  2.11E-03  2.68E-03  3.04E-04  635
 L-ICRP60ED              563-805 km  1.0000  2.40E-04  2.03E-04  4.58E-04  5.85E-04  1.02E-03  1.21E-03  1.77E-03  1.13E-03  334
 L-ICRP60ED             805-1609 km  1.0000  4.34E-05  1.08E-05  1.19E-04  1.48E-04  2.33E-04  2.72E-04  4.59E-04  3.04E-04  632

                               PROB                   QUANTILES                        PEAK      PEAK  PEAK
                             NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   1.00E-02 Sv  0.0262  4.23E+02  0.00E+00  0.00E+00  0.00E+00  1.52E+04  3.15E+04  4.20E+04  2.27E-03  517
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 AREA (ha) THAT EXCEEDS THRESHOLD
 A-THYROID  Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

 **** Indicates that the value is outside resolution of the analysis.
 Optionally increase number of trials for better resolution.

  "ATMOS"  DESCRIPTION = OCP3 high density no spray
  "EARLY"  DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM  1 OF  1:
   OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

   COHORT 9 = 30-40 Shadow

                               PROB                   QUANTILES                        PEAK      PEAK  PEAK
                             NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
HEALTH EFFECTS CASES
 ERL FAT/TOTAL              0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL              0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL               0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL               0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL               0-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL               0-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL              0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL               0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL               0-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL               0-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID            0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID            0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID             0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID            0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST             0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST              0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST             0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG               0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG               0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG                0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG               0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LEUKEMIA           0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BONE               0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LIVER              0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/COLON              0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/RESIDUAL           0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                               PROB                   QUANTILES                        PEAK      PEAK  PEAK
                             NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
EARLY FATALITY DISTANCE (km)
 ERL FAT/TOTAL RISK > 0.000             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                               PROB                   QUANTILES                        PEAK      PEAK  PEAK
                             NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
POPULATION EXCEEDING DOSE
 EARLY dose A-RED MARR > 2.32 Sv        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-LUNGS > 13.6 Sv           0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-STOMACH > 6.50 Sv         0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                               PROB                   QUANTILES                        PEAK      PEAK  PEAK
                             NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
POPULATION DOSE (Sv)
 L-ICRP60ED  TOT LIF        0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF        0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF         0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF        0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                               PROB                   QUANTILES                        PEAK      PEAK  PEAK
                             NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
POPULATION WEIGHTED RISK
 CAN FAT/TOTAL              0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL               0-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL               0-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL               0-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL              0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           16.1-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           32.2-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           48.3-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
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 CAN FAT/TOTAL          64.4-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           80.5-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            161-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            322-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           805-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                             PROB                  QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)
 L-ICRP60ED                 0-0.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               0.2-0.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               0.5-1.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               1.2-1.6 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               1.6-2.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               2.1-3.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               3.2-4.0 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               4.0-4.8 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               4.8-5.6 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED               5.6-8.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED              8.1-11.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED             11.3-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED             16.1-20.9 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED             20.9-25.8 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED             25.8-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED             32.2-40.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED             40.2-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED             48.3-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED             64.4-80.5 km  1.0000  8.56E-03  7.52E-03  9.72E-03  1.15E-02  1.80E-02  2.19E-02  3.54E-02  1.11E-03  312
 L-ICRP60ED              80.5-113 km  1.0000  6.67E-03  6.47E-03  8.40E-03  9.14E-03  1.36E-02  1.75E-02  2.17E-02  1.14E-03  518
 L-ICRP60ED               113-161 km  1.0000  4.62E-03  4.01E-03  7.34E-03  7.86E-03  9.24E-03  9.90E-03  1.33E-02  1.14E-03  518
 L-ICRP60ED               161-241 km  1.0000  2.72E-03  2.28E-03  5.02E-03  6.13E-03  7.67E-03  8.11E-03  9.33E-03  8.56E-04  456
 L-ICRP60ED               241-322 km  1.0000  1.72E-03  1.30E-03  3.17E-03  3.99E-03  6.68E-03  7.35E-03  8.41E-03  1.14E-03  513
 L-ICRP60ED               322-563 km  1.0000  8.73E-04  7.29E-04  1.56E-03  2.03E-03  3.21E-03  3.69E-03  5.03E-03  1.11E-03  312
 L-ICRP60ED               563-805 km  1.0000  4.57E-04  3.62E-04  8.76E-04  1.10E-03  1.86E-03  2.81E-03  3.66E-03  1.13E-03  334
 L-ICRP60ED              805-1609 km  1.0000  7.97E-05  2.07E-05  2.27E-04  2.90E-04  4.01E-04  4.59E-04  7.88E-04  3.04E-04  632

                             PROB                  QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   1.00E-02 Sv  0.0654  1.67E+03  0.00E+00  0.00E+00  1.17E+04  4.04E+04  5.38E+04  1.16E+05  1.13E-03  517
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 AREA (ha) THAT EXCEEDS THRESHOLD
 A-THYROID  Area exceeds   5.00E-02 Sv  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

 **** Indicates that the value is outside resolution of the analysis.
 Optionally increase number of trials for better resolution.

  "ATMOS"  DESCRIPTION = OCP3 high density no spray
  "EARLY"  DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM  1 OF  1:
   OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

   COHORT10 = 10-30 Tail

                             PROB                  QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
HEALTH EFFECTS CASES
 ERL FAT/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL             0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL             0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-32.2 km  0.9563  1.24E+00  2.56E-01  3.65E+00  6.12E+00  1.21E+01  1.68E+01  2.61E+01  1.15E-03  410
 CAN INJ/TOTAL             0-48.3 km  0.9563  1.79E+00  3.83E-01  5.20E+00  8.14E+00  1.75E+01  2.17E+01  3.35E+01  1.15E-03  410
 CAN INJ/TOTAL             0-64.4 km  0.9563  1.79E+00  3.83E-01  5.20E+00  8.14E+00  1.75E+01  2.17E+01  3.35E+01  1.15E-03  410
 CAN INJ/TOTAL             0-80.5 km  0.9563  1.79E+00  3.83E-01  5.20E+00  8.14E+00  1.75E+01  2.17E+01  3.35E+01  1.15E-03  410
 CAN INJ/TOTAL              0-161 km  0.9563  1.79E+00  3.83E-01  5.20E+00  8.14E+00  1.75E+01  2.17E+01  3.35E+01  1.15E-03  410
 CAN INJ/TOTAL              0-322 km  0.9563  1.79E+00  3.83E-01  5.20E+00  8.14E+00  1.75E+01  2.17E+01  3.35E+01  1.15E-03  410
 CAN INJ/TOTAL              0-805 km  0.9563  1.79E+00  3.83E-01  5.20E+00  8.14E+00  1.75E+01  2.17E+01  3.35E+01  1.15E-03  410
 CAN INJ/TOTAL             0-1609 km  0.9563  1.79E+00  3.83E-01  5.20E+00  8.14E+00  1.75E+01  2.17E+01  3.35E+01  1.15E-03  410
 CAN FAT/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-32.2 km  0.9563  5.78E-01  1.27E-01  1.78E+00  2.79E+00  5.17E+00  6.42E+00  1.22E+01  1.15E-03  410
 CAN FAT/TOTAL             0-48.3 km  0.9563  8.36E-01  1.90E-01  2.47E+00  3.99E+00  7.80E+00  9.95E+00  1.56E+01  1.15E-03  410
 CAN FAT/TOTAL             0-64.4 km  0.9563  8.36E-01  1.90E-01  2.47E+00  3.99E+00  7.80E+00  9.95E+00  1.56E+01  1.15E-03  410
 CAN FAT/TOTAL             0-80.5 km  0.9563  8.36E-01  1.90E-01  2.47E+00  3.99E+00  7.80E+00  9.95E+00  1.56E+01  1.15E-03  410
 CAN FAT/TOTAL              0-161 km  0.9563  8.36E-01  1.90E-01  2.47E+00  3.99E+00  7.80E+00  9.95E+00  1.56E+01  1.15E-03  410
 CAN FAT/TOTAL              0-322 km  0.9563  8.36E-01  1.90E-01  2.47E+00  3.99E+00  7.80E+00  9.95E+00  1.56E+01  1.15E-03  410
 CAN FAT/TOTAL              0-805 km  0.9563  8.36E-01  1.90E-01  2.47E+00  3.99E+00  7.80E+00  9.95E+00  1.56E+01  1.15E-03  410
 CAN FAT/TOTAL             0-1609 km  0.9563  8.36E-01  1.90E-01  2.47E+00  3.99E+00  7.80E+00  9.95E+00  1.56E+01  1.15E-03  410
 CAN FAT/THYROID           0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/THYROID           0-80.5 km  0.9563  5.12E-03  1.23E-03  1.39E-02  2.30E-02  4.95E-02  6.21E-02  9.32E-02  1.15E-03  410
 CAN FAT/THYROID            0-161 km  0.9563  5.12E-03  1.23E-03  1.39E-02  2.30E-02  4.95E-02  6.21E-02  9.32E-02  1.15E-03  410
 CAN FAT/THYROID           0-1609 km  0.9563  5.12E-03  1.23E-03  1.39E-02  2.30E-02  4.95E-02  6.21E-02  9.32E-02  1.15E-03  410
 CAN FAT/BREAST            0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/BREAST            0-80.5 km  0.9563  4.66E-02  9.46E-03  1.36E-01  2.20E-01  4.51E-01  5.37E-01  8.74E-01  1.15E-03  410
 CAN FAT/BREAST             0-161 km  0.9563  4.66E-02  9.46E-03  1.36E-01  2.20E-01  4.51E-01  5.37E-01  8.74E-01  1.15E-03  410
 CAN FAT/BREAST            0-1609 km  0.9563  4.66E-02  9.46E-03  1.36E-01  2.20E-01  4.51E-01  5.37E-01  8.74E-01  1.15E-03  410
 CAN FAT/LUNG              0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/LUNG              0-80.5 km  0.9563  1.59E-01  3.20E-02  4.84E-01  7.66E-01  1.55E+00  1.99E+00  2.94E+00  1.15E-03  410
 CAN FAT/LUNG               0-161 km  0.9563  1.59E-01  3.20E-02  4.84E-01  7.66E-01  1.55E+00  1.99E+00  2.94E+00  1.15E-03  410
 CAN FAT/LUNG              0-1609 km  0.9563  1.59E-01  3.20E-02  4.84E-01  7.66E-01  1.55E+00  1.99E+00  2.94E+00  1.15E-03  410
 CAN FAT/LEUKEMIA          0-1609 km  0.9563  1.53E-01  4.56E-02  4.69E-01  6.74E-01  1.18E+00  1.40E+00  2.82E+00  1.15E-03  410
 CAN FAT/BONE              0-1609 km  0.9563  4.82E-03  1.51E-03  1.35E-02  2.11E-02  3.72E-02  4.51E-02  8.86E-02  1.15E-03  410
 CAN FAT/LIVER             0-1609 km  0.9563  1.70E-02  3.46E-03  5.13E-02  7.84E-02  1.56E-01  1.99E-01  3.20E-01  1.15E-03  410
 CAN FAT/COLON             0-1609 km  0.9563  1.53E-01  3.32E-02  4.69E-01  7.20E-01  1.41E+00  1.77E+00  2.87E+00  1.15E-03  410
 CAN FAT/RESIDUAL          0-1609 km  0.9563  2.97E-01  6.11E-02  9.02E-01  1.36E+00  2.96E+00  3.52E+00  5.58E+00  1.15E-03  410

                             PROB                  QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
EARLY FATALITY DISTANCE (km)
 ERL FAT/TOTAL RISK > 0.000           0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                             PROB                  QUANTILES                    PEAK     PEAK  PEAK
                           NON-ZERO   MEAN     50TH      90TH      95TH      99TH     99.5TH    CONSEQ     PROB  TRIAL
POPULATION EXCEEDING DOSE
 EARLY dose A-RED MARR > 2.32 Sv      0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
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 EARLY dose A-LUNGS > 13.6 Sv          0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 EARLY dose A-STOMACH > 6.50 Sv        0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
POPULATION DOSE (Sv)
 L-ICRP60ED  TOT LIF       0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 L-ICRP60ED  TOT LIF       0-80.5 km  0.9563  1.48E+01  3.54E+00  4.40E+01  7.06E+01  1.29E+02  1.59E+02  2.76E+02  1.15E-03  410
 L-ICRP60ED  TOT LIF        0-161 km  0.9563  1.48E+01  3.54E+00  4.40E+01  7.06E+01  1.29E+02  1.59E+02  2.76E+02  1.15E-03  410
 L-ICRP60ED  TOT LIF       0-1609 km  0.9563  1.48E+01  3.54E+00  4.40E+01  7.06E+01  1.29E+02  1.59E+02  2.76E+02  1.15E-03  410

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
POPULATION WEIGHTED RISK
 CAN FAT/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-32.2 km  0.9563  1.40E-05  3.08E-06  4.30E-05  6.73E-05  1.35E-04  1.74E-04  2.95E-04  1.15E-03  410
 CAN FAT/TOTAL             0-48.3 km  0.9563  6.27E-06  1.39E-06  1.95E-05  3.03E-05  5.71E-05  6.97E-05  1.17E-04  1.15E-03  410
 CAN FAT/TOTAL             0-64.4 km  0.9563  6.27E-06  1.39E-06  1.95E-05  3.03E-05  5.71E-05  6.97E-05  1.17E-04  1.15E-03  410
 CAN FAT/TOTAL             0-80.5 km  0.9563  6.27E-06  1.39E-06  1.95E-05  3.03E-05  5.71E-05  6.97E-05  1.17E-04  1.15E-03  410
 CAN FAT/TOTAL              0-161 km  0.9563  6.27E-06  1.39E-06  1.95E-05  3.03E-05  5.71E-05  6.97E-05  1.17E-04  1.15E-03  410
 CAN FAT/TOTAL              0-322 km  0.9563  6.27E-06  1.39E-06  1.95E-05  3.03E-05  5.71E-05  6.97E-05  1.17E-04  1.15E-03  410
 CAN FAT/TOTAL              0-805 km  0.9563  6.27E-06  1.39E-06  1.95E-05  3.03E-05  5.71E-05  6.97E-05  1.17E-04  1.15E-03  410
 CAN FAT/TOTAL             0-1609 km  0.9563  6.27E-06  1.39E-06  1.95E-05  3.03E-05  5.71E-05  6.97E-05  1.17E-04  1.15E-03  410
 CAN FAT/TOTAL          16.1-32.2 km  0.9563  1.40E-05  3.08E-06  4.30E-05  6.73E-05  1.35E-04  1.74E-04  2.95E-04  1.15E-03  410
 CAN FAT/TOTAL          32.2-48.3 km  0.8391  2.81E-06  4.65E-07  8.91E-06  1.56E-05  3.02E-05  3.43E-05  4.50E-05  1.14E-03  693
 CAN FAT/TOTAL          48.3-64.4 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL          64.4-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           80.5-161 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            161-322 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL            322-805 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL           805-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)
 L-ICRP60ED                 0-0.2 km  1.0000  1.02E-01  5.92E-02  2.37E-01  3.22E-01  5.36E-01  6.07E-01  7.65E-01  1.12E-03   83
 L-ICRP60ED               0.2-0.5 km  1.0000  9.10E-02  4.84E-02  2.21E-01  3.08E-01  5.26E-01  5.76E-01  7.37E-01  1.12E-03   83
 L-ICRP60ED               0.5-1.2 km  1.0000  7.98E-02  3.88E-02  2.11E-01  2.98E-01  4.85E-01  5.39E-01  6.49E-01  1.14E-03  114
 L-ICRP60ED               1.2-1.6 km  1.0000  7.45E-02  3.39E-02  2.08E-01  2.87E-01  4.20E-01  4.89E-01  6.88E-01  1.14E-03  411
 L-ICRP60ED               1.6-2.1 km  1.0000  7.89E-02  3.42E-02  2.19E-01  3.08E-01  4.41E-01  5.13E-01  6.87E-01  1.14E-03  411
 L-ICRP60ED               2.1-3.2 km  1.0000  7.01E-02  2.89E-02  2.02E-01  3.00E-01  3.73E-01  4.09E-01  6.88E-01  1.12E-03   83
 L-ICRP60ED               3.2-4.0 km  0.9966  6.71E-02  3.05E-02  1.85E-01  2.64E-01  3.54E-01  3.86E-01  4.64E-01  1.14E-03  411
 L-ICRP60ED               4.0-4.8 km  0.9989  5.98E-02  2.54E-02  1.70E-01  2.41E-01  3.35E-01  3.60E-01  4.21E-01  1.15E-03  181
 L-ICRP60ED               4.8-5.6 km  0.9943  5.21E-02  2.30E-02  1.43E-01  2.10E-01  3.07E-01  3.32E-01  3.91E-01  1.14E-03  293
 L-ICRP60ED               5.6-8.1 km  0.9909  3.78E-02  1.63E-02  1.03E-01  1.41E-01  2.24E-01  2.45E-01  3.03E-01  1.12E-03  163
 L-ICRP60ED              8.1-11.3 km  0.9901  2.25E-02  1.03E-02  5.92E-02  8.98E-02  1.40E-01  1.66E-01  2.32E-01  1.14E-03  891
 L-ICRP60ED             11.3-16.1 km  0.9804  1.33E-02  5.53E-03  3.47E-02  5.59E-02  9.75E-02  1.11E-01  1.43E-01  1.15E-03  237
 L-ICRP60ED             16.1-20.9 km  0.9506  8.41E-03  2.51E-03  2.31E-02  3.45E-02  6.34E-02  7.21E-02  8.29E-02  1.15E-03  410
 L-ICRP60ED             20.9-25.8 km  0.9234  5.61E-03  1.06E-03  1.65E-02  2.57E-02  4.73E-02  5.46E-02  8.39E-02  1.11E-03  868
 L-ICRP60ED             25.8-32.2 km  0.8778  3.71E-03  6.18E-04  1.14E-02  1.62E-02  3.25E-02  3.78E-02  6.95E-02  1.15E-03  410
 L-ICRP60ED             32.2-40.2 km  0.8357  2.16E-03  3.44E-04  7.43E-03  1.07E-02  1.96E-02  2.22E-02  2.84E-02  1.15E-03  253
 L-ICRP60ED             40.2-48.3 km  0.7477  1.06E-03  1.48E-04  3.52E-03  5.88E-03  1.18E-02  1.52E-02  3.28E-02  1.15E-03  410
 L-ICRP60ED             48.3-64.4 km  0.6606  7.24E-04  6.86E-05  2.23E-03  4.62E-03  9.54E-03  1.20E-02  1.93E-02  1.15E-03  410
 L-ICRP60ED             64.4-80.5 km  1.0000  8.56E-03  7.52E-03  9.72E-03  1.15E-02  1.80E-02  2.19E-02  3.54E-02  1.11E-03  312
 L-ICRP60ED              80.5-113 km  1.0000  6.67E-03  6.47E-03  8.40E-03  9.14E-03  1.36E-02  1.75E-02  2.17E-02  1.14E-03  518
 L-ICRP60ED               113-161 km  1.0000  4.62E-03  4.01E-03  7.34E-03  7.86E-03  9.24E-03  9.90E-03  1.33E-02  1.14E-03  518
 L-ICRP60ED               161-241 km  1.0000  2.72E-03  2.28E-03  5.02E-03  6.13E-03  7.67E-03  8.11E-03  9.33E-03  8.56E-04  456
 L-ICRP60ED               241-322 km  1.0000  1.72E-03  1.30E-03  3.17E-03  3.99E-03  6.68E-03  7.35E-03  8.41E-03  1.14E-03  513
 L-ICRP60ED               322-563 km  1.0000  8.73E-04  7.29E-04  1.56E-03  2.03E-03  3.21E-03  3.69E-03  5.03E-03  1.11E-03  312
 L-ICRP60ED               563-805 km  1.0000  4.57E-04  3.62E-04  8.76E-04  1.10E-03  1.86E-03  2.81E-03  3.66E-03  1.13E-03  334
 L-ICRP60ED              805-1609 km  1.0000  7.97E-05  2.07E-05  2.27E-04  2.90E-04  4.01E-04  4.59E-04  7.88E-04  3.04E-04  632

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   1.00E-02 Sv  0.9862  6.15E+03  2.57E+03  1.53E+04  2.20E+04  4.16E+04  5.38E+04  1.17E+05  1.13E-03  517
 AREA (ha) THAT EXCEEDS THRESHOLD
 L-ICRP60ED Area exceeds   5.00E-02 Sv  0.5980  5.28E+02  4.44E+00  1.58E+03  2.94E+03  5.63E+03  6.84E+03  9.01E+03  1.14E-03  486
 AREA (ha) THAT EXCEEDS THRESHOLD
 A-THYROID  Area exceeds   5.00E-02 Sv  0.0295  1.05E+00  0.00E+00  0.00E+00  0.00E+00  3.36E+01  5.65E+01  1.83E+02  1.09E-03   97

 **** Indicates that the value is outside resolution of the analysis.
 Optionally increase number of trials for better resolution.

  "ATMOS"  DESCRIPTION = OCP3 high density no spray
  "EARLY"  DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM  1 OF  1:
   OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

   COHORT11 = 30-40 Shelter in Place

                          PROB                  QUANTILES                    PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
HEALTH EFFECTS CASES
 ERL FAT/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL             0-80.5 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 ERL FAT/TOTAL             0-1609 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN INJ/TOTAL             0-64.4 km  1.0000  1.65E+02  1.28E+02  3.11E+02  3.71E+02  5.52E+02  6.43E+02  7.61E+02  1.14E-03  390
 CAN INJ/TOTAL             0-80.5 km  1.0000  1.65E+02  1.28E+02  3.11E+02  3.71E+02  5.52E+02  6.43E+02  7.61E+02  1.14E-03  390
 CAN INJ/TOTAL              0-161 km  1.0000  1.65E+02  1.28E+02  3.11E+02  3.71E+02  5.52E+02  6.43E+02  7.61E+02  1.14E-03  390
 CAN INJ/TOTAL              0-322 km  1.0000  1.65E+02  1.28E+02  3.11E+02  3.71E+02  5.52E+02  6.43E+02  7.61E+02  1.14E-03  390
 CAN INJ/TOTAL              0-805 km  1.0000  1.65E+02  1.28E+02  3.11E+02  3.71E+02  5.52E+02  6.43E+02  7.61E+02  1.14E-03  390
 CAN INJ/TOTAL             0-1609 km  1.0000  1.65E+02  1.28E+02  3.11E+02  3.71E+02  5.52E+02  6.43E+02  7.61E+02  1.14E-03  390
 CAN FAT/TOTAL             0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-32.2 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-48.3 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 CAN FAT/TOTAL             0-64.4 km  1.0000  7.40E+01  6.14E+01  1.33E+02  1.68E+02  2.61E+02  3.03E+02  3.40E+02  1.14E-03  390
 CAN FAT/TOTAL             0-80.5 km  1.0000  7.40E+01  6.14E+01  1.33E+02  1.68E+02  2.61E+02  3.03E+02  3.40E+02  1.14E-03  390
 CAN FAT/TOTAL              0-161 km  1.0000  7.40E+01  6.14E+01  1.33E+02  1.68E+02  2.61E+02  3.03E+02  3.40E+02  1.14E-03  390
 CAN FAT/TOTAL              0-322 km  1.0000  7.40E+01  6.14E+01  1.33E+02  1.68E+02  2.61E+02  3.03E+02  3.40E+02  1.14E-03  390
 CAN FAT/TOTAL              0-805 km  1.0000  7.40E+01  6.14E+01  1.33E+02  1.68E+02  2.61E+02  3.03E+02  3.40E+02  1.14E-03  390
 CAN FAT/TOTAL             0-1609 km  1.0000  7.40E+01  6.14E+01  1.33E+02  1.68E+02  2.61E+02  3.03E+02  3.40E+02  1.14E-03  390
 CAN FAT/THYROID           0-16.1 km  0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
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CAN FAT/THYROID	0-80.5 km	1.0000	4.87E-01	3.97E-01	9.18E-01	1.12E+00	1.65E+00	1.95E+00	2.24E+00	1.14E-03	310
CAN FAT/THYROID	0-161 km	1.0000	4.87E-01	3.97E-01	9.18E-01	1.12E+00	1.65E+00	1.95E+00	2.24E+00	1.14E-03	310
CAN FAT/THYROID	0-1609 km	1.0000	4.87E-01	3.97E-01	9.18E-01	1.12E+00	1.65E+00	1.95E+00	2.24E+00	1.14E-03	310
CAN FAT/BREAST	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/BREAST	0-80.5 km	1.0000	5.07E+00	4.11E+00	9.77E+00	1.17E+01	1.74E+01	2.02E+01	2.34E+01	1.14E-03	390
CAN FAT/BREAST	0-161 km	1.0000	5.07E+00	4.11E+00	9.77E+00	1.17E+01	1.74E+01	2.02E+01	2.34E+01	1.14E-03	390
CAN FAT/BREAST	0-1609 km	1.0000	5.07E+00	4.11E+00	9.77E+00	1.17E+01	1.74E+01	2.02E+01	2.34E+01	1.14E-03	390
CAN FAT/LUNG	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/LUNG	0-80.5 km	1.0000	1.70E+01	1.32E+01	3.16E+01	3.85E+01	5.71E+01	6.51E+01	7.93E+01	1.14E-03	310
CAN FAT/LUNG	0-161 km	1.0000	1.70E+01	1.32E+01	3.16E+01	3.85E+01	5.71E+01	6.51E+01	7.93E+01	1.14E-03	310
CAN FAT/LUNG	0-1609 km	1.0000	1.70E+01	1.32E+01	3.16E+01	3.85E+01	5.71E+01	6.51E+01	7.93E+01	1.14E-03	310
CAN FAT/LEUKEMIA	0-1609 km	1.0000	7.16E+00	5.90E+00	1.29E+01	1.60E+01	2.55E+01	3.01E+01	3.29E+01	1.14E-03	390
CAN FAT/BONE	0-1609 km	1.0000	1.63E-01	1.28E-01	3.06E-01	3.71E-01	5.84E-01	7.06E-01	8.49E-01	8.56E-04	303
CAN FAT/LIVER	0-1609 km	1.0000	1.70E+00	1.30E+00	3.18E+00	3.85E+00	5.67E+00	6.49E+00	7.83E+00	1.14E-03	390
CAN FAT/COLON	0-1609 km	1.0000	1.38E+01	1.11E+01	2.62E+01	3.22E+01	4.46E+01	5.10E+01	6.38E+01	1.14E-03	390
CAN FAT/RESIDUAL	0-1609 km	1.0000	2.86E+01	2.37E+01	5.43E+01	6.90E+01	9.10E+01	1.02E+02	1.32E+02	1.14E-03	390

EARLY FATALITY DISTANCE (km)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL RISK > 0.000	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION EXCEEDING DOSE

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
EARLY dose A-RED MARR > 2.32 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-LUNGS > 13.6 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-STOMACH > 6.50 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION DOSE (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED TOT LIF	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
L-ICRP60ED TOT LIF	0-80.5 km	1.0000	1.28E+03	1.05E+03	2.40E+03	3.06E+03	4.40E+03	5.06E+03	5.88E+03	1.14E-03	390
L-ICRP60ED TOT LIF	0-161 km	1.0000	1.28E+03	1.05E+03	2.40E+03	3.06E+03	4.40E+03	5.06E+03	5.88E+03	1.14E-03	390
L-ICRP60ED TOT LIF	0-1609 km	1.0000	1.28E+03	1.05E+03	2.40E+03	3.06E+03	4.40E+03	5.06E+03	5.88E+03	1.14E-03	390

POPULATION WEIGHTED RISK

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	0-64.4 km	1.0000	4.53E-05	3.69E-05	8.51E-05	1.06E-04	1.47E-04	1.70E-04	2.08E-04	1.14E-03	390
CAN FAT/TOTAL	0-80.5 km	1.0000	4.53E-05	3.69E-05	8.51E-05	1.06E-04	1.47E-04	1.70E-04	2.08E-04	1.14E-03	390
CAN FAT/TOTAL	0-161 km	1.0000	4.53E-05	3.69E-05	8.51E-05	1.06E-04	1.47E-04	1.70E-04	2.08E-04	1.14E-03	390
CAN FAT/TOTAL	0-322 km	1.0000	4.53E-05	3.69E-05	8.51E-05	1.06E-04	1.47E-04	1.70E-04	2.08E-04	1.14E-03	390
CAN FAT/TOTAL	0-805 km	1.0000	4.53E-05	3.69E-05	8.51E-05	1.06E-04	1.47E-04	1.70E-04	2.08E-04	1.14E-03	390
CAN FAT/TOTAL	0-1609 km	1.0000	4.53E-05	3.69E-05	8.51E-05	1.06E-04	1.47E-04	1.70E-04	2.08E-04	1.14E-03	390
CAN FAT/TOTAL	16.1-32.2 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	32.2-48.3 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	48.3-64.4 km	1.0000	4.53E-05	3.69E-05	8.51E-05	1.06E-04	1.47E-04	1.70E-04	2.08E-04	1.14E-03	390
CAN FAT/TOTAL	64.4-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	80.5-161 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	161-322 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	322-805 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN FAT/TOTAL	805-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	1.0000	1.26E-01	1.05E-01	2.09E-01	2.47E-01	3.66E-01	4.34E-01	6.35E-01	1.11E-03	392
L-ICRP60ED	0.2-0.5 km	1.0000	5.38E-02	4.30E-02	8.99E-02	1.08E-01	1.43E-01	1.61E-01	2.21E-01	1.43E-04	158
L-ICRP60ED	0.5-1.2 km	1.0000	3.03E-02	2.46E-02	5.17E-02	7.17E-02	9.94E-02	1.04E-01	1.15E-01	1.13E-03	314
L-ICRP60ED	1.2-1.6 km	1.0000	2.34E-02	2.09E-02	3.92E-02	5.21E-02	7.35E-02	7.77E-02	8.74E-02	1.13E-03	314
L-ICRP60ED	1.6-2.1 km	1.0000	2.07E-02	1.94E-02	3.39E-02	4.62E-02	5.92E-02	6.43E-02	7.38E-02	1.13E-03	314
L-ICRP60ED	2.1-3.2 km	1.0000	1.77E-02	1.54E-02	2.67E-02	3.11E-02	4.56E-02	5.09E-02	5.53E-02	1.13E-03	314
L-ICRP60ED	3.2-4.0 km	1.0000	1.64E-02	1.42E-02	2.42E-02	2.68E-02	3.27E-02	3.50E-02	4.05E-02	1.13E-03	314
L-ICRP60ED	4.0-4.8 km	1.0000	1.58E-02	1.38E-02	2.26E-02	2.42E-02	2.84E-02	3.05E-02	3.64E-02	1.15E-03	552
L-ICRP60ED	4.8-5.6 km	1.0000	1.62E-02	1.42E-02	2.26E-02	2.42E-02	2.84E-02	3.04E-02	3.55E-02	1.12E-03	760
L-ICRP60ED	5.6-8.1 km	1.0000	1.66E-02	1.48E-02	2.27E-02	2.43E-02	2.84E-02	3.08E-02	4.16E-02	1.14E-03	830
L-ICRP60ED	8.1-11.3 km	1.0000	1.84E-02	1.72E-02	2.55E-02	2.86E-02	4.13E-02	4.88E-02	8.08E-02	1.12E-03	607
L-ICRP60ED	11.3-16.1 km	1.0000	1.97E-02	1.96E-02	2.94E-02	3.26E-02	4.07E-02	4.48E-02	6.24E-02	1.12E-03	607
L-ICRP60ED	16.1-20.9 km	1.0000	2.01E-02	2.00E-02	3.10E-02	3.31E-02	3.85E-02	4.10E-02	4.72E-02	1.13E-03	606
L-ICRP60ED	20.9-25.8 km	1.0000	1.94E-02	1.85E-02	3.07E-02	3.30E-02	3.89E-02	4.17E-02	4.86E-02	1.14E-03	829
L-ICRP60ED	25.8-32.2 km	1.0000	1.69E-02	1.38E-02	2.70E-02	3.06E-02	3.40E-02	3.56E-02	3.94E-02	1.12E-03	545
L-ICRP60ED	32.2-40.2 km	1.0000	1.34E-02	1.05E-02	2.14E-02	2.42E-02	3.07E-02	3.19E-02	3.46E-02	1.15E-03	953
L-ICRP60ED	40.2-48.3 km	1.0000	1.08E-02	8.71E-03	1.48E-02	1.85E-02	2.59E-02	2.95E-02	3.43E-02	1.13E-03	517
L-ICRP60ED	48.3-64.4 km	1.0000	8.74E-03	7.66E-03	1.04E-02	1.24E-02	1.85E-02	2.18E-02	3.56E-02	1.14E-03	723
L-ICRP60ED	64.4-80.5 km	1.0000	7.02E-03	7.00E-03	8.67E-03	9.50E-03	1.41E-02	1.71E-02	2.55E-02	1.11E-03	312
L-ICRP60ED	80.5-113 km	1.0000	5.02E-03	4.38E-03	7.67E-03	8.38E-03	1.04E-02	1.19E-02	1.56E-02	1.14E-03	518
L-ICRP60ED	113-161 km	1.0000	3.22E-03	2.76E-03	5.52E-03	6.81E-03	8.09E-03	8.64E-03	9.95E-03	1.14E-03	332
L-ICRP60ED	161-241 km	1.0000	1.86E-03	1.43E-03	3.31E-03	4.17E-03	6.60E-03	7.19E-03	7.81E-03	1.15E-03	330
L-ICRP60ED	241-322 km	1.0000	1.16E-03	9.53E-04	2.13E-03	2.82E-03	4.76E-03	5.21E-03	5.83E-03	1.14E-03	513
L-ICRP60ED	322-563 km	1.0000	5.87E-04	5.01E-04	1.05E-03	1.33E-03	2.17E-03	2.46E-03	3.22E-03	1.11E-03	312
L-ICRP60ED	563-805 km	1.0000	2.97E-04	2.45E-04	5.68E-04	6.88E-04	1.32E-03	1.90E-03	2.32E-03	1.13E-03	334
L-ICRP60ED	805-1609 km	1.0000	5.24E-05	1.34E-05	1.39E-04	1.86E-04	2.60E-04	2.96E-04	5.29E-04	3.04E-04	632

DOSE FOUND AT ALL LOCATIONS (Sv)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
AREA (ha) THAT EXCEEDS THRESHOLD										
L-ICRP60ED Area exceeds 1.00E-02 Sv	1.0000	1.40E+04	1.09E+04	2.79E+04	3.38E+04	5.06E+04	6.74E+04	9.84E+04	1.14E-03	518
AREA (ha) THAT EXCEEDS THRESHOLD										
L-ICRP60ED Area exceeds 5.00E-02 Sv	0.5067	4.75E+00	5.31E-02	4.93E+00	8.68E+00	4.64E+01	3.21E+02	9.35E+02	1.12E-03	607
AREA (ha) THAT EXCEEDS THRESHOLD										
A-THYROID Area exceeds 5.00E-02 Sv	0.0027	8.50E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.99E-01	1.56E-03	158

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 high density no spray
"EARLY" DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 high density no spray

RESULTS FOR A SINGLE EMERGENCY RESPONSE COHORT WITHOUT ANY WEIGHTING FRACTIONS BEING APPLIED

COHORT12 = Nonevacuees

HEALTH EFFECTS CASES

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL	0-16.1 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-80.5 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
ERL FAT/TOTAL	0-1609 km	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CAN INJ/TOTAL	0-16.1 km	1.0000	2.87E-02	2.22E-02	5.87E-02	7.34E-02	1.05E-01	1.21E-01	1.66E-01	1.11E-03	392
CAN INJ/TOTAL	0-32.2 km	1.0000	4.44E-01	3.73E-01	7.94E-01	9.43E-01	1.14E+00	1.22E+00	1.76E+00	1.14E-04	34
CAN INJ/TOTAL	0-48.3 km	1.0000	1.25E+00	1.08E+00	2.02E+00	2.26E+00	2.94E+00	3.15E+00	3.58E+00	1.15E-03	101
CAN INJ/TOTAL	0-64.4 km	1.0000	2.54E+00	2.24E+00	4.22E+00	5.10E+00	6.02E+00	6.46E+00	8.17E+00	1.14E-03	389
CAN INJ/TOTAL	0-80.5 km	1.0000	2.04E+02	1.88E+02	3.32E+02	3.78E+02	5.08E+02	5.53E+02	6.64E+02	1.14E-03	388
CAN INJ/TOTAL	0-161 km	1.0000	1.11E+03	8.36E+02	2.26E+03	2.77E+03	3.64E+03	4.03E+03	5.02E+03	1.14E-03	388
CAN INJ/TOTAL	0-322 km	1.0000	2.11E+03	1.32E+03	4.73E+03	6.87E+03	9.13E+03	1.02E+04	1.25E+04	1.14E-03	718
CAN INJ/TOTAL	0-805 km	1.0000	2.44E+03	1.64E+03	5.19E+03	7.26E+03	1.03E+04	1.11E+04	1.31E+04	1.14E-03	718
CAN INJ/TOTAL	0-1609 km	1.0000	2.47E+03	1.70E+03	5.19E+03	7.26E+03	1.03E+04	1.11E+04	1.31E+04	1.14E-03	718
CAN FAT/TOTAL	0-16.1 km	1.0000	1.30E-02	1.03E-02	2.65E-02	3.28E-02	4.74E-02	5.41E-02	7.48E-02	1.11E-03	392
CAN FAT/TOTAL	0-32.2 km	1.0000	1.99E-01	1.72E-01	3.50E-01	4.17E-01	5.75E-01	6.40E-01	7.96E-01	1.14E-04	34
CAN FAT/TOTAL	0-48.3 km	1.0000	5.58E-01	5.14E-01	9.10E-01	1.04E+00	1.25E+00	1.35E+00	1.60E+00	1.15E-03	101
CAN FAT/TOTAL	0-64.4 km	1.0000	1.14E+00	1.03E+00	1.90E+00	2.19E+00	2.82E+00	3.10E+00	3.65E+00	1.14E-03	389
CAN FAT/TOTAL	0-80.5 km	1.0000	9.11E+01	8.33E+01	1.45E+02	1.75E+02	2.29E+02	2.49E+02	2.98E+02	1.14E-03	388
CAN FAT/TOTAL	0-161 km	1.0000	4.96E+02	3.65E+02	1.01E+03	1.12E+03	1.43E+03	1.59E+03	2.24E+03	1.14E-03	388
CAN FAT/TOTAL	0-322 km	1.0000	9.46E+02	6.04E+02	2.11E+03	3.02E+03	3.77E+03	4.14E+03	5.60E+03	1.14E-03	718
CAN FAT/TOTAL	0-805 km	1.0000	1.09E+03	7.62E+02	2.32E+03	3.12E+03	4.00E+03	4.46E+03	5.87E+03	1.14E-03	718
CAN FAT/TOTAL	0-1609 km	1.0000	1.10E+03	7.90E+02	2.32E+03	3.12E+03	4.00E+03	4.46E+03	5.87E+03	1.14E-03	718
CAN FAT/THYROID	0-16.1 km	1.0000	8.57E-05	6.77E-05	1.68E-04	2.14E-04	3.01E-04	3.49E-04	4.80E-04	1.11E-03	392
CAN FAT/THYROID	0-80.5 km	1.0000	5.97E-01	5.52E-01	1.01E+00	1.12E+00	1.42E+00	1.57E+00	1.96E+00	1.14E-03	388
CAN FAT/THYROID	0-161 km	1.0000	3.24E+00	2.44E+00	6.61E+00	7.99E+00	1.10E+01	1.20E+01	1.46E+01	1.14E-03	388
CAN FAT/THYROID	0-1609 km	1.0000	7.16E+00	5.07E+00	1.54E+01	2.09E+01	2.95E+01	3.23E+01	3.84E+01	1.14E-03	718
CAN FAT/BREAST	0-16.1 km	1.0000	8.23E-04	6.54E-04	1.63E-03	2.11E-03	3.01E-03	3.47E-03	4.70E-03	1.11E-03	392
CAN FAT/BREAST	0-80.5 km	1.0000	6.37E+00	5.87E+00	1.04E+01	1.15E+01	1.46E+01	1.61E+01	2.04E+01	1.14E-03	388
CAN FAT/BREAST	0-161 km	1.0000	3.47E+01	2.62E+01	7.15E+01	8.39E+01	1.13E+02	1.25E+02	1.54E+02	1.14E-03	388
CAN FAT/BREAST	0-1609 km	1.0000	7.72E+01	5.59E+01	1.60E+02	2.16E+02	3.14E+02	3.35E+02	3.85E+02	1.14E-03	718
CAN FAT/LUNG	0-16.1 km	1.0000	2.80E-03	2.15E-03	5.74E-03	7.23E-03	1.00E-02	1.16E-02	1.60E-02	1.11E-03	392
CAN FAT/LUNG	0-80.5 km	1.0000	2.07E+01	1.96E+01	3.43E+01	3.95E+01	5.30E+01	5.79E+01	6.97E+01	1.14E-03	388
CAN FAT/LUNG	0-161 km	1.0000	1.13E+02	8.60E+01	2.30E+02	2.87E+02	3.69E+02	4.06E+02	5.21E+02	1.14E-03	388
CAN FAT/LUNG	0-1609 km	1.0000	2.53E+02	1.76E+02	5.39E+02	7.44E+02	1.03E+03	1.13E+03	1.37E+03	1.14E-03	718
CAN FAT/LEUKEMIA	0-1609 km	1.0000	1.05E+02	7.59E+01	2.22E+02	2.99E+02	3.93E+02	4.42E+02	5.64E+02	1.14E-03	718
CAN FAT/BONE	0-1609 km	1.0000	2.36E+00	1.61E+00	5.16E+00	6.83E+00	9.35E+00	1.04E+01	1.27E+01	1.14E-03	718
CAN FAT/LIVER	0-1609 km	1.0000	2.55E+01	1.77E+01	5.28E+01	7.35E+01	1.05E+02	1.13E+02	1.33E+02	1.14E-03	718
CAN FAT/COLON	0-1609 km	1.0000	2.07E+02	1.42E+02	4.44E+02	6.00E+02	7.98E+02	8.60E+02	1.10E+03	1.14E-03	718
CAN FAT/RESIDUAL	0-1609 km	1.0000	4.28E+02	3.07E+02	9.20E+02	1.11E+03	1.52E+03	1.73E+03	2.26E+03	1.14E-03	718

EARLY FATALITY DISTANCE (km)

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
ERL FAT/TOTAL RISK > 0.000	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION EXCEEDING DOSE

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
EARLY dose A-RED MARR > 2.32 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-LUNGS > 13.6 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
EARLY dose A-STOMACH > 6.50 Sv	0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

POPULATION DOSE (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED TOT LIF	0-16.1 km	1.0000	2.27E-01	1.76E-01	4.63E-01	5.76E-01	8.43E-01	9.79E-01	1.29E+00	1.11E-03	392
L-ICRP60ED TOT LIF	0-80.5 km	1.0000	1.58E+03	1.32E+03	2.70E+03	3.13E+03	3.82E+03	4.16E+03	5.14E+03	1.14E-03	388
L-ICRP60ED TOT LIF	0-161 km	1.0000	8.61E+03	6.64E+03	1.67E+04	2.12E+04	2.87E+04	3.19E+04	3.86E+04	1.14E-03	388
L-ICRP60ED TOT LIF	0-1609 km	1.0000	1.92E+04	1.33E+04	4.14E+04	5.62E+04	7.82E+04	8.46E+04	1.01E+05	1.14E-03	718

POPULATION WEIGHTED RISK

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
CAN FAT/TOTAL	0-16.1 km	1.0000	5.99E-05	4.75E-05	1.17E-04	1.43E-04	2.15E-04	2.38E-04	3.45E-04	1.11E-03	392
CAN FAT/TOTAL	0-32.2 km	1.0000	8.73E-05	7.56E-05	1.47E-04	1.84E-04	2.54E-04	2.88E-04	3.48E-04	1.14E-04	34
CAN FAT/TOTAL	0-48.3 km	1.0000	8.11E-05	7.42E-05	1.21E-04	1.37E-04	1.84E-04	2.04E-04	2.33E-04	1.15E-03	101
CAN FAT/TOTAL	0-64.4 km	1.0000	6.63E-05	5.95E-05	1.03E-04	1.15E-04	1.45E-04	1.61E-04	2.13E-04	1.14E-03	389
CAN FAT/TOTAL	0-80.5 km	1.0000	4.58E-05	4.06E-05	7.67E-05	9.00E-05	1.15E-04	1.25E-04	1.50E-04	1.14E-03	388
CAN FAT/TOTAL	0-161 km	1.0000	3.24E-05	2.44E-05	6.60E-05	7.95E-05	1.09E-04	1.20E-04	1.47E-04	1.14E-03	388
CAN FAT/TOTAL	0-322 km	1.0000	2.14E-05	1.34E-05	4.76E-05	6.95E-05	9.36E-05	1.04E-04	1.26E-04	1.14E-03	718
CAN FAT/TOTAL	0-805 km	1.0000	1.05E-05	7.43E-06	2.21E-05	3.06E-05	3.79E-05	4.16E-05	5.64E-05	1.14E-03	718
CAN FAT/TOTAL	0-1609 km	1.0000	5.91E-06	4.15E-06	1.21E-05	1.66E-05	2.32E-05	2.53E-05	3.14E-05	1.14E-03	718
CAN FAT/TOTAL	16.1-32.2 km	1.0000	9.01E-05	7.78E-05	1.54E-04	1.93E-04	2.65E-04	3.01E-04	3.77E-04	1.14E-04	34
CAN FAT/TOTAL	32.2-48.3 km	1.0000	7.81E-05	7.22E-05	1.17E-04	1.32E-04	1.75E-04	1.97E-04	2.16E-04	1.14E-03	519
CAN FAT/TOTAL	48.3-64.4 km	1.0000	5.63E-05	4.79E-05	1.02E-04	1.15E-04	1.53E-04	1.74E-04	2.63E-04	1.14E-03	389
CAN FAT/TOTAL	64.4-80.5 km	1.0000	4.56E-05	4.06E-05	7.64E-05	8.99E-05	1.15E-04	1.25E-04	1.50E-04	1.14E-03	388
CAN FAT/TOTAL	80.5-161 km	1.0000	3.04E-05	2.17E-05	6.58E-05	8.03E-05	1.10E-04	1.21E-04	1.46E-04	1.14E-03	388
CAN FAT/TOTAL	161-322 km	1.0000	1.55E-05	6.74E-06	4.03E-05	6.18E-05	8.80E-05	9.89E-05	1.38E-04	1.14E-03	718
CAN FAT/TOTAL	322-805 km	0.9897	2.41E-06	2.06E-06	4.84E-06	5.83E-06	9.66E-06	1.08E-05	1.32E-05	1.15E-03	543
CAN FAT/TOTAL	805-1609 km	0.9165	1.85E-07	2.81E-09	6.30E-07	1.11E-06	2.08E-06	2.33E-06	4.14E-06	1.13E-03	724

PEAK DOSE FOUND ON SPATIAL GRID (Sv)

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
L-ICRP60ED	0-0.2 km	1.0000	1.76E-01	1.39E-01	3.00E-01	3.43E-01	4.64E-01	5.62E-01	8.85E-01	1.11E-03	392
L-ICRP60ED	0.2-0.5 km	1.0000	7.49E-02	6.19E-02	1.18E-01	1.48E-01	2.11E-01	2.24E-01	3.09E-01	1.43E-04	158
L-ICRP60ED	0.5-1.2 km	1.0000	4.05E-02	3.31E-02	7.25E-02	1.01E-01	1.23E-01	1.34E-01	1.61E-01	1.13E-03	314
L-ICRP60ED	1.2-1.6 km	1.0000	2.95E-02	2.38E-02	5.41E-02	7.23E-02	1.03E-01	1.09E-01	1.22E-01	1.13E-03	314
L-ICRP60ED	1.6-2.1 km	1.0000	2.47E-02	2.12E-02	4.38E-02	6.22E-02	8.05E-02	8.63E-02	1.03E-01	1.13E-03	314
L-ICRP60ED	2.1-3.2 km	1.0000	1.96E-02	1.70E-02	3.12E-02	4.27E-02	6.47E-02	7.12E-02	7.70E-02	1.13E-03	314
L-ICRP60ED	3.2-4.0 km	1.0000	1.61E-02	1.33E-02	2.57E-02	3.02E-02	4.54E-02	5.10E-02	5.63E-02	1.13E-03	314
L-ICRP60ED	4.0-4.8 km	1.0000	1.48E-02	1.25E-02	2.39E-02	2.72E-02	3.42E-02	3.72E-02	4.46E-02	1.13E-03	314
L-ICRP60ED	4.8-5.6 km	1.0000	1.46E-02	1.25E-02	2.32E-02	2.58E-02	3.16E-02	3.35E-02	3.80E-02	1.15E-03	552
L-ICRP60ED	5.6-8.1 km	1.0000	1.52E-02	1.31E-02	2.21E-02	2.35E-02	2.72E-02	2.90E-02	5.78E-02	1.14E-03	830
L-ICRP60ED	8.1-11.3 km	1.0000	1.62E-02	1.36E-02	2.29E-02	2.47E-02	2.97E-02	6.02E-02	1.12E-01	1.12E-03	607
L-ICRP60ED	11.3-16.1 km	1.0000	1.75E-02	1.56E-02	2.46E-02	2.72E-02	3.82E-02	4.54E-02	8.67E-02	1.12E-03	607
L-ICRP60ED	16.1-20.9 km	1.0000	1.90E-02	1.91E-02	2.75E-02	3.14E-02	4.15E-02	4.68E-02	6.56E-02	1.13E-03	606
L-ICRP60ED	20.9-25.8 km	1.0000	1.91E-02	1.76E-02	2.87E-02	3.19E-02	3.86E-02	4.19E-02	6.75E-02	1.14E-03	829
L-ICRP60ED	25.8-32.2 km	1.0000	1.83E-02	1.63E-02	2.85E-02	3.18E-02	3.86E-02	4.19E-02	5.31E-02	1.14E-03	829
L-ICRP60ED	32.2-40.2 km	1.0000	1.58E-02	1.27E-02	2.50E-02	2.92E-02	3.38E-02	3.58E-02	4.03E-02	1.14E-03	829
L-ICRP60ED	40.2-48.3 km	1.0000	1.29E-02	1.01E-02	2.07E-02	2.29E-02	2.91E-02	3.10E-02	3.43E-02	1.14E-03	567
L-ICRP60ED	48.3-64.4 km	1.0000	1.02E-02	8.33E-03	1.32E-02	1.62E-02	2.41E-02	2.77E-02	3.35E-02	1.14E-03	723
L-ICRP60ED	64.4-80.5 km	1.0000	8.56E-03	7.52E-03	9.72E-03	1.15E-02	1.80E-02	2.19E-02	3.54E-02	1.11E-03	312
L-ICRP60ED	80.5-113 km	1.0000	6.67E-03	6.47E-03	8.40E-03	9.14E-03	1.36E-02	1.75E-02	2.17E-02	1.14E-03	518
L-ICRP60ED	113-161 km	1.0000	4.62E-03	4.01E-03	7.34E-03	7.86E-03	9.24E-03	9.90E-03	1.33E-02	1.14E-03	518
L-ICRP60ED	161-241 km	1.0000	2.72E-03	2.28E-03	5.02E-03	6.13E-03	7.67E-03	8.11E-03	9.33E-03	8.56E-04	456
L-ICRP60ED	241-322 km	1.0000	1.72E-03	1.30E-03	3.17E-03	3.99E-03	6.68E-03	7.35E-03	8.41E-03	1.14E-03	513

L-ICRP60ED	322-563 km	1.0000	8.73E-04	7.29E-04	1.56E-03	2.03E-03	3.21E-03	3.69E-03	5.03E-03	1.11E-03	312
L-ICRP60ED	563-805 km	1.0000	4.57E-04	3.62E-04	8.76E-04	1.10E-03	1.86E-03	2.81E-03	3.66E-03	1.13E-03	334
L-ICRP60ED	805-1609 km	1.0000	7.97E-05	2.07E-05	2.27E-04	2.90E-04	4.01E-04	4.59E-04	7.88E-04	3.04E-04	632

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
DOSE FOUND AT ALL LOCATIONS (Sv)											
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	1.00E-02 Sv	1.0000	1.89E+04	1.39E+04	3.82E+04	4.98E+04	7.92E+04	9.21E+04	1.68E+05	1.14E-03	518
AREA (ha) THAT EXCEEDS THRESHOLD											
L-ICRP60ED Area exceeds	5.00E-02 Sv	0.7192	1.29E+01	1.12E+00	1.01E+01	2.47E+01	3.07E+02	7.63E+02	1.61E+03	1.12E-03	607
AREA (ha) THAT EXCEEDS THRESHOLD											
A-THYROID Area exceeds	5.00E-02 Sv	0.0112	3.74E-03	0.00E+00	0.00E+00	0.00E+00	5.57E-02	3.18E-01	1.65E+00	1.43E-04	158

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

"ATMOS" DESCRIPTION = OCP3 high density no spray
"EARLY" DESCRIPTION = OCP3 high density no spray, EARLY input
"CHRONC" DESCRIPTION = OCP3 high density no spray, EARLY input

SOURCE TERM 1 OF 1:
OCP3 high density no spray

RESULTS FROM THE "CHRONC" MODULE ALONE

COHORT13 = OCP3 high density no spray, EARLY input

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
HEALTH EFFECTS CASES											
CAN INJ/TOTAL	0-16.1 km	1.0000	2.41E+02	2.19E+02	3.32E+02	3.62E+02	4.44E+02	4.85E+02	6.44E+02	1.12E-03	464
CAN INJ/TOTAL	0-32.2 km	1.0000	1.20E+03	1.07E+03	1.66E+03	2.00E+03	2.32E+03	2.47E+03	2.84E+03	1.12E-03	391
CAN INJ/TOTAL	0-48.3 km	1.0000	2.96E+03	2.83E+03	4.18E+03	4.86E+03	5.54E+03	5.81E+03	6.44E+03	1.14E-03	7
CAN INJ/TOTAL	0-64.4 km	1.0000	6.05E+03	5.40E+03	1.00E+04	1.06E+04	1.22E+04	1.29E+04	1.47E+04	1.14E-03	390
CAN INJ/TOTAL	0-80.5 km	1.0000	8.81E+03	7.97E+03	1.15E+04	1.25E+04	1.52E+04	1.66E+04	1.98E+04	1.14E-03	693
CAN INJ/TOTAL	0-161 km	1.0000	2.54E+04	2.34E+04	3.83E+04	4.45E+04	5.37E+04	5.63E+04	6.22E+04	1.14E-03	387
CAN INJ/TOTAL	0-322 km	1.0000	5.78E+04	5.33E+04	1.02E+05	1.09E+05	1.27E+05	1.36E+05	1.57E+05	1.14E-03	388
CAN INJ/TOTAL	0-805 km	1.0000	9.05E+04	8.56E+04	1.35E+05	1.56E+05	2.08E+05	2.22E+05	2.53E+05	1.14E-03	234
CAN INJ/TOTAL	0-1609 km	1.0000	9.88E+04	9.59E+04	1.49E+05	1.78E+05	2.35E+05	2.59E+05	3.15E+05	1.13E-03	396
CAN FAT/TOTAL	0-16.1 km	1.0000	1.06E+02	1.01E+02	1.37E+02	1.56E+02	2.08E+02	2.29E+02	2.80E+02	1.12E-03	464
CAN FAT/TOTAL	0-32.2 km	1.0000	5.28E+02	5.09E+02	7.72E+02	8.53E+02	1.04E+03	1.10E+03	1.24E+03	1.12E-03	391
CAN FAT/TOTAL	0-48.3 km	1.0000	1.30E+03	1.12E+03	1.88E+03	2.08E+03	2.37E+03	2.50E+03	2.82E+03	1.14E-03	7
CAN FAT/TOTAL	0-64.4 km	1.0000	2.65E+03	2.35E+03	3.98E+03	4.74E+03	5.52E+03	5.79E+03	6.42E+03	1.14E-03	390
CAN FAT/TOTAL	0-80.5 km	1.0000	3.86E+03	3.44E+03	5.96E+03	6.94E+03	7.66E+03	7.98E+03	8.69E+03	1.14E-03	693
CAN FAT/TOTAL	0-161 km	1.0000	1.11E+04	1.05E+04	1.63E+04	1.97E+04	2.28E+04	2.41E+04	2.73E+04	1.14E-03	387
CAN FAT/TOTAL	0-322 km	1.0000	2.53E+04	2.31E+04	3.97E+04	4.57E+04	5.60E+04	5.98E+04	6.89E+04	1.14E-03	388
CAN FAT/TOTAL	0-805 km	1.0000	3.95E+04	3.61E+04	6.60E+04	7.42E+04	9.12E+04	9.97E+04	1.10E+05	1.14E-03	234
CAN FAT/TOTAL	0-1609 km	1.0000	4.31E+04	3.97E+04	7.19E+04	8.14E+04	1.05E+05	1.14E+05	1.35E+05	1.13E-03	396
CAN FAT/THYROID	0-16.1 km	1.0000	5.79E-01	5.37E-01	8.10E-01	9.01E-01	1.12E+00	1.21E+00	1.45E+00	1.12E-03	464
CAN FAT/THYROID	0-80.5 km	1.0000	2.38E+01	2.16E+01	3.44E+01	3.84E+01	4.93E+01	5.11E+01	5.37E+01	1.14E-03	693
CAN FAT/THYROID	0-161 km	1.0000	6.89E+01	7.01E+01	1.04E+02	1.12E+02	1.34E+02	1.44E+02	1.69E+02	1.14E-03	387
CAN FAT/THYROID	0-1609 km	1.0000	2.64E+02	2.50E+02	4.15E+02	4.89E+02	6.32E+02	7.01E+02	7.94E+02	5.99E-04	397
CAN FAT/BREAST	0-16.1 km	1.0000	7.97E+00	7.41E+00	1.10E+01	1.19E+01	1.45E+01	1.57E+01	1.88E+01	1.12E-03	464
CAN FAT/BREAST	0-80.5 km	1.0000	3.70E+02	3.32E+02	5.71E+02	6.52E+02	7.48E+02	7.76E+02	8.39E+02	1.14E-03	693
CAN FAT/BREAST	0-161 km	1.0000	1.07E+03	1.04E+03	1.55E+03	1.85E+03	2.22E+03	2.35E+03	2.64E+03	1.14E-03	387
CAN FAT/BREAST	0-1609 km	1.0000	4.03E+03	3.72E+03	6.64E+03	7.51E+03	9.56E+03	1.02E+04	1.12E+04	5.99E-04	397
CAN FAT/LUNG	0-16.1 km	1.0000	1.70E+01	1.44E+01	2.36E+01	2.63E+01	3.29E+01	3.57E+01	4.26E+01	1.12E-03	464
CAN FAT/LUNG	0-80.5 km	1.0000	6.96E+02	6.32E+02	1.05E+03	1.12E+03	1.29E+03	1.37E+03	1.57E+03	1.14E-03	693
CAN FAT/LUNG	0-161 km	1.0000	2.01E+03	2.02E+03	3.08E+03	3.31E+03	3.93E+03	4.23E+03	4.94E+03	1.14E-03	387
CAN FAT/LUNG	0-1609 km	1.0000	7.68E+03	7.43E+03	1.20E+04	1.35E+04	1.78E+04	2.00E+04	2.26E+04	1.13E-03	396
CAN FAT/LEUKEMIA	0-1609 km	1.0000	4.43E+03	4.10E+03	7.35E+03	8.36E+03	1.07E+04	1.14E+04	1.33E+04	1.13E-03	396
CAN FAT/BONE	0-1609 km	1.0000	1.07E+02	1.02E+02	1.57E+02	1.89E+02	2.28E+02	2.44E+02	2.98E+02	5.99E-04	397
CAN FAT/LIVER	0-1609 km	1.0000	1.13E+03	1.04E+03	1.76E+03	2.10E+03	2.70E+03	3.01E+03	3.48E+03	5.99E-04	397
CAN FAT/COLON	0-1609 km	1.0000	8.12E+03	7.75E+03	1.27E+04	1.47E+04	2.02E+04	2.21E+04	2.69E+04	1.13E-03	396
CAN FAT/RESIDUAL	0-1609 km	1.0000	1.74E+04	1.54E+04	2.84E+04	3.28E+04	4.42E+04	5.01E+04	5.70E+04	1.13E-03	396

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
POPULATION DOSE (Sv)											
L-ICRP60ED TOT LIF	0-16.1 km	1.0000	5.41E+03	5.16E+03	7.00E+03	7.59E+03	9.14E+03	9.91E+03	1.11E+04	1.14E-03	465
L-ICRP60ED TOT LIF	0-80.5 km	1.0000	7.68E+04	7.13E+04	1.08E+05	1.16E+05	1.36E+05	1.46E+05	1.69E+05	1.14E-03	693
L-ICRP60ED TOT LIF	0-161 km	1.0000	2.15E+05	2.09E+05	3.21E+05	3.55E+05	4.47E+05	4.93E+05	5.22E+05	1.15E-03	694
L-ICRP60ED TOT LIF	0-1609 km	1.0000	8.36E+05	7.93E+05	1.27E+06	1.44E+06	1.94E+06	2.09E+06	2.48E+06	5.99E-04	397

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
POPULATION WEIGHTED RISK											
CAN FAT/TOTAL	0-16.1 km	1.0000	5.56E-04	4.93E-04	1.00E-03	1.08E-03	1.27E-03	1.37E-03	1.59E-03	1.14E-03	776
CAN FAT/TOTAL	0-32.2 km	1.0000	8.00E-04	7.43E-04	1.17E-03	1.31E-03	1.69E-03	1.88E-03	2.18E-03	1.12E-03	391
CAN FAT/TOTAL	0-48.3 km	1.0000	7.61E-04	7.35E-04	1.08E-03	1.15E-03	1.35E-03	1.44E-03	1.67E-03	1.14E-03	944
CAN FAT/TOTAL	0-64.4 km	1.0000	6.72E-04	5.99E-04	1.06E-03	1.14E-03	1.34E-03	1.44E-03	1.68E-03	1.14E-03	390
CAN FAT/TOTAL	0-80.5 km	1.0000	6.37E-04	5.81E-04	1.02E-03	1.08E-03	1.23E-03	1.29E-03	1.46E-03	1.14E-03	693
CAN FAT/TOTAL	0-161 km	1.0000	5.54E-04	5.37E-04	8.63E-04	1.00E-03	1.15E-03	1.22E-03	1.38E-03	1.14E-03	387
CAN FAT/TOTAL	0-322 km	1.0000	4.98E-04	4.46E-04	8.33E-04	9.33E-04	1.11E-03	1.19E-03	1.36E-03	1.14E-03	388
CAN FAT/TOTAL	0-805 km	1.0000	3.22E-04	3.06E-04	5.38E-04	5.88E-04	7.18E-04	7.66E-04	8.80E-04	1.14E-03	234
CAN FAT/TOTAL	0-1609 km	1.0000	1.92E-04	1.93E-04	3.15E-04	3.38E-04	3.98E-04	4.27E-04	4.97E-04	1.14E-03	234
CAN FAT/TOTAL	16.1-32.2 km	1.0000	8.25E-04	7.61E-04	1.20E-03	1.35E-03	1.77E-03	1.99E-03	2.28E-03	1.12E-03	391
CAN FAT/TOTAL	32.2-48.3 km	1.0000	7.42E-04	7.19E-04	1.07E-03	1.15E-03	1.36E-03	1.46E-03	1.70E-03	1.14E-03	245
CAN FAT/TOTAL	48.3-64.4 km	1.0000	6.12E-04	5.17E-04	1.06E-03	1.16E-03	1.42E-03	1.55E-03	1.87E-03	1.13E-03	714
CAN FAT/TOTAL	64.4-80.5 km	1.0000	5.76E-04	5.41E-04	9.22E-04	1.03E-03	1.15E-03	1.21E-03	1.35E-03	1.15E-03	72
CAN FAT/TOTAL	80.5-161 km	1.0000	5.20E-04	5.19E-04	8.48E-04	9.90E-04	1.15E-03	1.22E-03	1.40E-03	1.14E-03	387
CAN FAT/TOTAL	161-322 km	1.0000	4.62E-04	3.46E-04	9.81E-04	1.05E-03	1.20E-03	1.27E-03	1.43E-03	1.15E-03	444
CAN FAT/TOTAL	322-805 km	0.9897	1.82E-04	1.45E-04	3.60E-04	4.35E-04	5.60E-04	6.01E-04	8.03E-04	1.13E-03	968
CAN FAT/TOTAL	805-1609 km	0.9165	2.40E-05	3.83E-07	8.32E-05	1.40E-04	3.06E-04	3.44E-04	4.39E-04	1.13E-03	877

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.5TH	PEAK CONSEQ	PEAK PROB	PEAK TRIAL
PEAK DOSE FOUND ON SPATIAL GRID (Sv)											
L-ICRP60ED	0-0.2 km	0.4373	2.86E-02	0.00E+00	7.15E-02	7.26E-02	7.52E-02	7.64E-02	7.90E-02	1.13E-03	777
L-ICRP60ED	0.2-0.5 km	0.8523	5.84E-02	7.02E-02	7.24E-02	7.34E-02	7.57E-02	7.67E-02	7.89E-02	1.13E-03	660
L-ICRP60ED	0.5-1.2 km	0.9397	6.67E-02	7.03E-02	7.25E-02	7.35E-02	7.58E-02	7.68E-02	7.89E-02	1.15E-03	405
L-ICRP60ED	1.2-1.6 km	0.9539	6.69E-02	7.03E-02	7.25E-02	7.35E-02	7.58E-02	7.68E-02	7.90E-02	1.14E-03	605
L-ICRP60ED	1.6-2.1 km	0.9574	6.63E-02	7.03E-02	7.24E-02	7.33E-02	7.55E-02	7.64E-02	7.90E-02	8.56E-04	150
L-ICRP60ED	2.1-3.2 km	0.9819	6.74E-02	7.02E-02	7.24E-02	7.34E-02	7.57E-02	7.67E-02	7.89E-02	1.13E-03	696
L-ICRP60ED	3.2-4.0 km	0.9892	6.85E-02	7.03E-02	7.25E-02	7.34E-02	7.57E-02	7.67E-02	7.89E-02	1.13E-03	686
L-ICRP60ED	4.0-4.8 km	0.9847	6.85E-02	7.03E-02	7.25E-02	7.35E-02	7.57E-02	7.68E-02	7.89E-02	1.15E-03	457
L-ICRP60ED	4.8-5.6 km	0.9973	7.05E-02	7.04E-02	7.26E-02	7.35E-02	7.58E-02	7.68E-02	7.89E-02	1.15E-03	552
L-ICRP60ED	5.6-8.1 km	0.9989	7.17E-02	7.05E-02	7.26E-02	7.35E-02	7.58E-02	7.68E-02	7.89E-02	1.15E-03	338
L-ICRP60ED	8.1-11.3 km	1.0000	7.21E-02	7.05E-02	7.26E-02	7.36E-02	7.58E-02	7.67E-02	7.88E-02	1.14E-03	852

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L-ICRP60ED           11.3-16.1 km  0.9989  7.46E-02  7.07E-02  7.28E-02  7.37E-02  7.58E-02  7.68E-02  7.88E-02  1.15E-03  826
L-ICRP60ED           16.1-20.9 km  1.0000  7.57E-02  7.07E-02  7.28E-02  7.37E-02  7.58E-02  7.68E-02  7.88E-02  1.14E-03  300
L-ICRP60ED           20.9-25.8 km  1.0000  7.62E-02  7.08E-02  7.28E-02  7.37E-02  7.59E-02  7.68E-02  7.89E-02  1.12E-03  760
L-ICRP60ED           25.8-32.2 km  1.0000  7.56E-02  7.07E-02  7.25E-02  7.33E-02  7.53E-02  7.61E-02  7.88E-02  5.99E-04  943
L-ICRP60ED           32.2-40.2 km  1.0000  7.34E-02  7.06E-02  7.26E-02  7.36E-02  7.57E-02  7.67E-02  7.88E-02  1.13E-03  618
L-ICRP60ED           40.2-48.3 km  1.0000  6.96E-02  7.00E-02  7.23E-02  7.32E-02  7.55E-02  7.66E-02  7.88E-02  1.13E-03  117
L-ICRP60ED           48.3-64.4 km  1.0000  6.26E-02  5.82E-02  7.09E-02  7.17E-02  7.36E-02  7.44E-02  7.88E-02  1.43E-04    3
L-ICRP60ED           64.4-80.5 km  1.0000  5.74E-02  5.46E-02  7.04E-02  7.16E-02  7.46E-02  7.59E-02  7.88E-02  1.13E-03  939
L-ICRP60ED            80.5-113 km  1.0000  5.39E-02  5.23E-02  6.52E-02  7.02E-02  7.25E-02  7.35E-02  7.88E-02  1.52E-04   94
L-ICRP60ED             113-161 km  1.0000  5.06E-02  4.94E-02  5.74E-02  6.10E-02  7.01E-02  7.26E-02  7.85E-02  1.14E-03  319
L-ICRP60ED             161-241 km  1.0000  4.72E-02  3.84E-02  5.28E-02  5.52E-02  6.11E-02  6.38E-02  7.04E-02  1.15E-03  457
L-ICRP60ED             241-322 km  1.0000  4.34E-02  3.50E-02  5.03E-02  5.13E-02  5.38E-02  5.49E-02  5.74E-02  1.13E-03  847
L-ICRP60ED             322-563 km  1.0000  3.83E-02  3.26E-02  4.33E-02  4.90E-02  5.28E-02  5.42E-02  5.73E-02  1.13E-03  432
L-ICRP60ED             563-805 km  1.0000  3.13E-02  3.08E-02  3.94E-02  4.38E-02  5.14E-02  5.27E-02  5.56E-02  1.13E-03  815
L-ICRP60ED            805-1609 km  1.0000  9.99E-03  2.73E-03  3.03E-02  3.32E-02  4.11E-02  4.51E-02  5.04E-02  1.13E-03  827

                       PROB              QUANTILES                  PEAK     PEAK  PEAK
                     NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
L-ICRP60ED  POP. DOSE (Sv)    0-16.1 km
 TOTAL LONG-TERM PATHWAYS DOSE         1.0000  5.41E+03  5.16E+03  7.00E+03  7.59E+03  9.14E+03  9.91E+03  1.11E+04  1.14E-03  465
 LONG-TERM DIRECT EXPOSURE PATHWAYS    1.0000  4.63E+02  4.01E+02  7.99E+02  9.44E+02  1.11E+03  1.18E+03  1.32E+03  1.14E-03  776
 TOTAL INGESTION PATHWAYS DOSE         1.0000  4.90E+03  4.40E+03  6.36E+03  7.14E+03  8.89E+03  9.77E+03  1.09E+04  1.14E-03  465
 LONG-TERM GROUNDSHINE DOSE            1.0000  4.62E+02  4.01E+02  7.99E+02  9.44E+02  1.11E+03  1.18E+03  1.32E+03  1.14E-03  776
 LONG-TERM RESUSPENSION DOSE           1.0000  6.77E-01  5.84E-01  1.16E+00  1.33E+00  1.84E+00  2.07E+00  2.52E+00  1.15E-03  896
 WATER INGESTION DOSE                  1.0000  4.87E+03  4.34E+03  6.33E+03  7.11E+03  8.88E+03  9.77E+03  1.09E+04  1.14E-03  465
 POP.-DEPENDENT DECONTAMINATION DOSE   1.0000  4.16E+01  3.44E+01  7.77E+01  9.37E+01  1.16E+02  1.24E+02  1.46E+02  1.14E-03  930
 FARM-DEPENDENT DECONTAMINATION DOSE   1.0000  2.75E+00  2.35E+00  5.26E+00  6.23E+00  7.64E+00  8.05E+00  9.02E+00  1.14E-03  390
 INGESTION OF GRAINS                   1.0000  1.08E+00  8.62E-01  2.17E+00  2.66E+00  3.89E+00  4.52E+00  6.01E+00  3.71E-04  279
 INGESTION OF LEAF VEG                 1.0000  5.52E+00  4.51E+00  1.06E+01  1.18E+01  1.53E+01  1.71E+01  2.27E+01  3.23E-04  299
 INGESTION OF ROOT CROPS               1.0000  3.58E+00  2.91E+00  7.41E+00  9.12E+00  1.12E+01  1.19E+01  1.51E+01  3.23E-04  299
 INGESTION OF FRUITS                   1.0000  5.54E+00  5.03E+00  1.01E+01  1.09E+01  1.31E+01  1.42E+01  1.68E+01  1.15E-03  894
 INGESTION OF LEGUMES                  1.0000  6.36E+00  5.19E+00  1.25E+01  1.55E+01  2.15E+01  2.29E+01  2.63E+01  1.15E-03  894
 INGESTION OF BEEF                     1.0000  5.29E+00  3.92E+00  1.07E+01  1.34E+01  2.22E+01  2.70E+01  3.35E+01  1.13E-03  781
 INGESTION OF MILK                     1.0000  5.44E+00  3.75E+00  1.10E+01  1.45E+01  2.72E+01  3.37E+01  4.98E+01  1.14E-03  383
 INGESTION OF POULTRY                  1.0000  1.72E+00  1.21E+00  3.34E+00  4.58E+00  8.27E+00  9.70E+00  1.35E+01  1.14E-03  319
 INGESTION OF OTHER MEAT CROPS         1.0000  3.36E-01  2.65E-01  6.56E-01  8.41E-01  1.29E+00  1.52E+00  2.12E+00  3.71E-04  279

                       PROB              QUANTILES                  PEAK     PEAK  PEAK
                     NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
L-ICRP60ED  POP. DOSE (Sv)    0-80.5 km
 TOTAL LONG-TERM PATHWAYS DOSE         1.0000  7.68E+04  7.13E+04  1.08E+05  1.16E+05  1.36E+05  1.46E+05  1.69E+05  1.14E-03  693
 LONG-TERM DIRECT EXPOSURE PATHWAYS    1.0000  6.57E+04  5.98E+04  1.03E+05  1.09E+05  1.25E+05  1.33E+05  1.50E+05  1.14E-03  388
 TOTAL INGESTION PATHWAYS DOSE         1.0000  7.68E+03  7.28E+03  1.02E+04  1.10E+04  1.31E+04  1.41E+04  1.66E+04  1.14E-03  328
 LONG-TERM GROUNDSHINE DOSE            1.0000  6.56E+04  5.98E+04  1.03E+05  1.09E+05  1.25E+05  1.33E+05  1.50E+05  1.14E-03  388
 LONG-TERM RESUSPENSION DOSE           1.0000  1.38E+02  1.18E+02  2.22E+02  2.47E+02  3.06E+02  3.18E+02  3.46E+02  1.14E-03  235
 WATER INGESTION DOSE                  1.0000  6.76E+03  6.17E+03  9.01E+03  1.02E+04  1.24E+04  1.35E+04  1.62E+04  1.14E-03  328
 POP.-DEPENDENT DECONTAMINATION DOSE   1.0000  3.32E+03  3.02E+03  5.53E+03  6.88E+03  9.39E+03  1.06E+04  1.36E+04  1.13E-03  714
 FARM-DEPENDENT DECONTAMINATION DOSE   1.0000  5.09E+01  4.65E+01  7.45E+01  8.49E+01  1.07E+02  1.14E+02  1.47E+02  3.23E-04    6
 INGESTION OF GRAINS                   1.0000  2.97E+01  2.26E+01  5.75E+01  7.09E+01  9.86E+01  1.10E+02  1.37E+02  1.14E-03  263
 INGESTION OF LEAF VEG                 1.0000  1.10E+02  1.01E+02  1.60E+02  1.95E+02  2.63E+02  2.98E+02  3.48E+02  1.13E-03  379
 INGESTION OF ROOT CROPS               1.0000  7.25E+01  6.51E+01  1.11E+02  1.24E+02  1.61E+02  1.79E+02  2.20E+02  1.14E-03  263
 INGESTION OF FRUITS                   1.0000  1.45E+02  1.21E+02  2.16E+02  2.41E+02  3.05E+02  3.24E+02  3.68E+02  1.14E-03  141
 INGESTION OF LEGUMES                  1.0000  1.20E+02  1.07E+02  1.73E+02  2.03E+02  2.24E+02  2.33E+02  2.81E+02  2.38E-04  306
 INGESTION OF BEEF                     1.0000  1.95E+02  1.53E+02  3.57E+02  4.36E+02  5.91E+02  6.54E+02  8.41E+02  1.15E-03  787
 INGESTION OF MILK                     1.0000  1.83E+02  1.33E+02  3.47E+02  4.80E+02  7.99E+02  9.23E+02  1.19E+03  1.14E-03  355
 INGESTION OF POULTRY                  1.0000  6.29E+01  5.18E+01  1.14E+02  1.50E+02  2.33E+02  2.62E+02  3.23E+02  1.13E-03  379
 INGESTION OF OTHER MEAT CROPS         1.0000  1.03E+01  8.48E+00  1.78E+01  2.30E+01  3.46E+01  3.89E+01  5.27E+01  1.13E-03  379

                       PROB              QUANTILES                  PEAK     PEAK  PEAK
                     NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
L-ICRP60ED  POP. DOSE (Sv)    0-161 km
 TOTAL LONG-TERM PATHWAYS DOSE         1.0000  2.15E+05  2.09E+05  3.21E+05  3.55E+05  4.47E+05  4.93E+05  5.22E+05  1.15E-03  694
 LONG-TERM DIRECT EXPOSURE PATHWAYS    1.0000  1.98E+05  2.01E+05  3.07E+05  3.30E+05  3.91E+05  4.21E+05  4.93E+05  1.14E-03  387
 TOTAL INGESTION PATHWAYS DOSE         1.0000  1.15E+04  1.05E+04  1.48E+04  1.71E+04  2.16E+04  2.30E+04  2.64E+04  1.13E-03  117
 LONG-TERM GROUNDSHINE DOSE            1.0000  1.97E+05  2.01E+05  3.06E+05  3.30E+05  3.91E+05  4.21E+05  4.92E+05  1.14E-03  387
 LONG-TERM RESUSPENSION DOSE           1.0000  5.10E+02  4.95E+02  7.97E+02  8.96E+02  1.13E+03  1.24E+03  1.51E+03  1.14E-03  930
 WATER INGESTION DOSE                  1.0000  8.11E+03  7.47E+03  1.10E+04  1.23E+04  1.58E+04  1.77E+04  2.16E+04  1.13E-03  286
 POP.-DEPENDENT DECONTAMINATION DOSE   1.0000  5.61E+03  4.49E+03  1.04E+04  1.26E+04  1.95E+04  2.21E+04  2.84E+04  1.14E-03  885
 FARM-DEPENDENT DECONTAMINATION DOSE   1.0000  7.92E+01  6.56E+01  1.29E+02  1.63E+02  2.17E+02  2.29E+02  2.58E+02  1.15E-03   72
 INGESTION OF GRAINS                   1.0000  1.06E+02  5.72E+01  2.46E+02  3.16E+02  4.58E+02  5.26E+02  6.68E+02  1.13E-03  117
 INGESTION OF LEAF VEG                 1.0000  2.88E+02  2.25E+02  5.37E+02  7.02E+02  1.03E+03  1.17E+03  1.56E+03  1.14E-03  393
 INGESTION OF ROOT CROPS               1.0000  2.02E+02  1.48E+02  3.89E+02  4.95E+02  6.89E+02  7.67E+02  9.56E+02  1.13E-03  117
 INGESTION OF FRUITS                   1.0000  3.79E+02  3.12E+02  6.68E+02  8.09E+02  1.12E+03  1.24E+03  1.52E+03  1.13E-03  117
 INGESTION OF LEGUMES                  1.0000  2.82E+02  2.45E+02  4.54E+02  5.39E+02  7.25E+02  7.92E+02  9.56E+02  1.13E-03  117
 INGESTION OF BEEF                     1.0000  9.17E+02  7.56E+02  1.78E+03  2.25E+03  3.24E+03  3.57E+03  4.40E+03  1.14E-03  774
 INGESTION OF MILK                     1.0000  9.18E+02  6.04E+02  1.87E+03  2.64E+03  5.43E+03  6.01E+03  9.42E+03  5.99E-04  397
 INGESTION OF POULTRY                  1.0000  2.90E+02  2.21E+02  5.77E+02  7.75E+02  1.18E+03  1.35E+03  1.80E+03  1.14E-03  393
 INGESTION OF OTHER MEAT CROPS         1.0000  4.36E+01  3.37E+01  8.37E+01  1.11E+02  1.78E+02  2.11E+02  2.71E+02  1.14E-03  393

                       PROB              QUANTILES                  PEAK     PEAK  PEAK
                     NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
L-ICRP60ED  POP. DOSE (Sv)    0-1609 km
 TOTAL LONG-TERM PATHWAYS DOSE         1.0000  8.36E+05  7.93E+05  1.27E+06  1.44E+06  1.94E+06  2.09E+06  2.48E+06  5.99E-04  397
 LONG-TERM DIRECT EXPOSURE PATHWAYS    1.0000  6.98E+05  7.04E+05  1.10E+06  1.19E+06  1.42E+06  1.53E+06  1.80E+06  1.14E-03  234
 TOTAL INGESTION PATHWAYS DOSE         1.0000  1.30E+05  8.75E+04  2.44E+05  3.43E+05  6.81E+05  8.43E+05  1.20E+06  1.13E-03  396
 LONG-TERM GROUNDSHINE DOSE            1.0000  6.96E+05  7.02E+05  1.10E+06  1.18E+06  1.41E+06  1.53E+06  1.80E+06  1.14E-03  234
 LONG-TERM RESUSPENSION DOSE           1.0000  2.44E+03  2.28E+03  3.90E+03  4.53E+03  5.62E+03  6.03E+03  7.18E+03  1.14E-03  234
 WATER INGESTION DOSE                  1.0000  3.73E+04  3.28E+04  5.91E+04  6.68E+04  7.34E+04  7.52E+04  7.92E+04  1.13E-03  761
 POP.-DEPENDENT DECONTAMINATION DOSE   1.0000  7.36E+03  5.52E+03  1.38E+04  1.91E+04  3.10E+04  3.43E+04  4.25E+04  1.14E-03  576
 FARM-DEPENDENT DECONTAMINATION DOSE   1.0000  9.70E+01  7.74E+01  1.72E+02  2.20E+02  3.17E+02  3.50E+02  4.33E+02  1.15E-03  457
 INGESTION OF GRAINS                   1.0000  3.43E+03  4.75E+02  1.01E+04  1.39E+04  2.76E+04  3.30E+04  4.46E+04  1.14E-03  944
 INGESTION OF LEAF VEG                 1.0000  5.91E+03  9.92E+02  1.42E+04  2.30E+04  6.00E+04  1.01E+05  1.49E+05  5.99E-04  397
 INGESTION OF ROOT CROPS               1.0000  4.67E+03  7.04E+02  1.30E+04  1.88E+04  3.43E+04  3.87E+04  6.06E+04  1.14E-03  944
 INGESTION OF FRUITS                   1.0000  7.15E+03  1.72E+03  2.02E+04  2.81E+04  5.54E+04  6.31E+04  8.94E+04  1.14E-03  944
 INGESTION OF LEGUMES                  1.0000  4.22E+03  1.07E+03  1.15E+04  1.61E+04  3.32E+04  3.79E+04  5.16E+04  1.14E-03  944
 INGESTION OF BEEF                     1.0000  3.02E+04  1.66E+04  7.30E+04  1.01E+05  1.76E+05  2.14E+05  2.94E+05  1.14E-03  944
 INGESTION OF MILK                     1.0000  2.58E+04  1.39E+04  6.00E+04  8.97E+04  1.77E+05  2.45E+05  5.27E+05  1.13E-03  396
 INGESTION OF POULTRY                  1.0000  1.02E+04  5.34E+03  2.13E+04  3.30E+04  8.76E+04  1.12E+05  2.15E+05  5.99E-04  397
 INGESTION OF OTHER MEAT CROPS         1.0000  1.43E+03  7.55E+02  3.05E+03  4.60E+03  1.31E+04  2.01E+04  3.01E+04  5.99E-04  397

                       PROB              QUANTILES                  PEAK     PEAK  PEAK
                     NON-ZERO   MEAN     50TH      90TH      95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
ECONOMIC COST MEASURES ($)    0-1609 km
 TOTAL ECONOMIC COSTS                  1.0000  8.46E+11  6.45E+11  1.70E+12  2.06E+12  2.36E+12  2.51E+12  2.85E+12  1.14E-03  547
  POP.-DEPENDENT COSTS                 1.0000  8.25E+11  6.12E+11  1.66E+12  2.05E+12  2.35E+12  2.50E+12  2.84E+12  1.14E-03  547
  FARM-DEPENDENT COSTS                 1.0000  2.11E+10  1.45E+10  4.35E+10  5.58E+10  8.40E+10  9.79E+10  1.28E+11  1.13E-03  396
  POP.-DEPENDENT DECONTAMINATION COST  1.0000  1.03E+11  7.71E+10  2.15E+11  2.69E+11  3.50E+11  3.82E+11  4.59E+11  1.11E-03  290
  FARM-DEPENDENT DECONTAMINATION COST  1.0000  3.12E+09  2.42E+09  5.92E+09  7.14E+09  9.17E+09  1.02E+10  1.27E+10  1.14E-03  568
  POP.-DEPENDENT INTERDICTION COST     1.0000  7.02E+11  5.23E+11  1.32E+12  1.64E+12  2.13E+12  2.22E+12  2.44E+12  1.14E-03  547
  FARM-DEPENDENT INTERDICTION COST     1.0000  1.11E+10  6.10E+09  2.70E+10  3.50E+10  5.59E+10  6.81E+10  9.08E+10  1.13E-03  859
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 POP.-DEPENDENT CONDEMNATION COST    1.0000  1.93E+10  1.28E+10  3.84E+10  5.59E+10  9.75E+10  1.22E+11  2.13E+11  1.14E-03  886
 FARM-DEPENDENT CONDEMNATION COST    1.0000  5.05E+08  4.58E+08  7.99E+08  9.83E+08  1.20E+09  1.30E+09  1.96E+09  1.43E-04   76
 EMERGENCY PHASE COST                1.0000  4.70E+08  3.63E+08  8.53E+08  1.11E+09  1.86E+09  2.20E+09  3.04E+09  1.15E-03  711
 INTERMEDIATE PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL COST                  1.0000  2.39E+08  1.30E+08  6.16E+08  8.14E+08  1.23E+09  1.44E+09  2.02E+09  1.14E-03  860
 CROP DISPOSAL COST                  1.0000  6.14E+09  4.62E+09  1.19E+10  1.47E+10  2.18E+10  2.42E+10  3.84E+10  1.13E-03  396

                      PROB                QUANTILES                 PEAK     PEAK  PEAK
                    NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH  CONSEQ    PROB  TRIAL
ECONOMIC COST MEASURES ($)    0-16.1 km
 TOTAL ECONOMIC COSTS                1.0000  4.54E+09  4.08E+09  6.40E+09  7.12E+09  8.02E+09  8.45E+09  9.44E+09  1.11E-03  392
 POP.-DEPENDENT COSTS                1.0000  4.25E+09  3.76E+09  5.92E+09  6.62E+09  7.65E+09  8.02E+09  8.91E+09  1.11E-03  392
 FARM-DEPENDENT COSTS                1.0000  2.94E+08  3.00E+08  3.91E+08  4.38E+08  5.25E+08  5.48E+08  6.00E+08  1.12E-03  577
 POP.-DEPENDENT DECONTAMINATION COST 1.0000  1.55E+08  1.27E+08  2.81E+08  3.18E+08  3.82E+08  4.14E+08  4.91E+08  1.14E-03  776
 FARM-DEPENDENT DECONTAMINATION COST 1.0000  1.70E+07  1.38E+07  3.04E+07  3.33E+07  4.12E+07  4.51E+07  5.26E+07  1.14E-03  355
 POP.-DEPENDENT INTERDICTION COST    1.0000  8.72E+08  7.74E+08  1.39E+09  1.66E+09  2.13E+09  2.24E+09  2.49E+09  1.14E-03  930
 FARM-DEPENDENT INTERDICTION COST    1.0000  2.11E+07  1.91E+07  3.51E+07  3.97E+07  5.22E+07  5.73E+07  7.36E+07  1.15E-03  896
 POP.-DEPENDENT CONDEMNATION COST    1.0000  3.20E+09  3.06E+09  4.56E+09  5.14E+09  5.89E+09  6.24E+09  7.33E+09  1.43E-04    8
 FARM-DEPENDENT CONDEMNATION COST    1.0000  2.03E+08  2.04E+08  3.07E+08  3.28E+08  3.83E+08  4.10E+08  4.74E+08  1.12E-03  577
 EMERGENCY PHASE COST                1.0000  1.91E+07  1.69E+07  2.82E+07  3.07E+07  3.37E+07  3.50E+07  4.28E+07  1.43E-04    8
 INTERMEDIATE PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL COST                  1.0000  4.54E+06  4.44E+06  7.13E+06  7.29E+06  7.68E+06  7.85E+06  8.24E+06  1.11E-03  139
 CROP DISPOSAL COST                  1.0000  4.81E+07  4.71E+07  7.16E+07  7.36E+07  7.85E+07  8.07E+07  8.56E+07  1.14E-03  936

                      PROB                QUANTILES                 PEAK     PEAK  PEAK
                    NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH  CONSEQ    PROB  TRIAL
ECONOMIC COST MEASURES ($)  16.1-32.2 km
 TOTAL ECONOMIC COSTS                1.0000  2.43E+10  2.28E+10  3.37E+10  3.66E+10  4.41E+10  4.78E+10  5.73E+10  1.14E-03    7
 POP.-DEPENDENT COSTS                1.0000  2.38E+10  2.22E+10  3.34E+10  3.63E+10  4.40E+10  4.78E+10  5.63E+10  1.14E-03    7
 FARM-DEPENDENT COSTS                1.0000  5.22E+08  5.18E+08  7.52E+08  8.00E+08  9.22E+08  9.81E+08  1.25E+09  1.43E-04   76
 POP.-DEPENDENT DECONTAMINATION COST 1.0000  2.35E+09  2.14E+09  3.83E+09  4.49E+09  5.57E+09  5.94E+09  6.83E+09  1.12E-03  391
 FARM-DEPENDENT DECONTAMINATION COST 1.0000  8.24E+07  7.60E+07  1.21E+08  1.36E+08  1.76E+08  1.97E+08  2.21E+08  1.14E-03  387
 POP.-DEPENDENT INTERDICTION COST    1.0000  1.20E+10  1.07E+10  1.94E+10  2.17E+10  2.70E+10  2.97E+10  3.48E+10  1.14E-03  244
 FARM-DEPENDENT INTERDICTION COST    1.0000  9.63E+07  9.10E+07  1.35E+08  1.55E+08  2.05E+08  2.14E+08  2.36E+08  1.14E-03  387
 POP.-DEPENDENT CONDEMNATION COST    0.9989  9.29E+09  7.31E+09  1.76E+10  2.20E+10  3.11E+10  3.37E+10  4.00E+10  1.15E-03  380
 FARM-DEPENDENT CONDEMNATION COST    1.0000  2.09E+08  1.95E+08  3.45E+08  3.95E+08  5.30E+08  5.87E+08  9.07E+08  1.43E-04   76
 EMERGENCY PHASE COST                1.0000  1.13E+08  1.04E+08  1.69E+08  2.03E+08  2.40E+08  2.57E+08  3.03E+08  1.14E-03    7
 INTERMEDIATE PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL COST                  1.0000  9.72E+06  9.32E+06  1.42E+07  1.66E+07  2.03E+07  2.06E+07  2.12E+07  1.14E-03    2
 CROP DISPOSAL COST                  1.0000  1.24E+08  1.10E+08  2.02E+08  2.08E+08  2.23E+08  2.30E+08  2.45E+08  1.14E-03  930

                      PROB                QUANTILES                 PEAK     PEAK  PEAK
                    NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH  CONSEQ    PROB  TRIAL
ECONOMIC COST MEASURES ($)  32.2-48.3 km
 TOTAL ECONOMIC COSTS                1.0000  3.42E+10  3.18E+10  5.17E+10  5.60E+10  6.75E+10  7.41E+10  9.31E+10  1.14E-03    7
 POP.-DEPENDENT COSTS                1.0000  3.37E+10  3.17E+10  5.11E+10  5.52E+10  6.61E+10  7.21E+10  9.18E+10  1.14E-03    7
 FARM-DEPENDENT COSTS                1.0000  4.91E+08  4.95E+08  7.04E+08  7.62E+08  9.14E+08  9.88E+08  1.28E+09  1.14E-03    7
 POP.-DEPENDENT DECONTAMINATION COST 1.0000  4.99E+09  4.80E+09  7.58E+09  8.48E+09  1.05E+10  1.11E+10  1.25E+10  1.14E-03  944
 FARM-DEPENDENT DECONTAMINATION COST 1.0000  1.29E+08  1.12E+08  1.94E+08  2.11E+08  2.45E+08  2.61E+08  2.99E+08  1.12E-03  391
 POP.-DEPENDENT INTERDICTION COST    1.0000  2.43E+10  2.25E+10  3.51E+10  3.88E+10  4.91E+10  5.14E+10  5.52E+10  1.14E-03  245
 FARM-DEPENDENT INTERDICTION COST    1.0000  1.52E+08  1.27E+08  2.20E+08  2.39E+08  2.90E+08  3.08E+08  3.37E+08  1.14E-03  930
 POP.-DEPENDENT CONDEMNATION COST    0.6345  4.32E+09  1.03E+09  1.32E+10  2.02E+10  3.08E+10  3.37E+10  4.10E+10  1.14E-03  381
 FARM-DEPENDENT CONDEMNATION COST    0.7323  6.04E+07  3.45E+07  1.49E+08  2.11E+08  3.42E+08  3.94E+08  6.81E+08  1.43E-04   76
 EMERGENCY PHASE COST                0.9989  1.26E+08  1.12E+08  2.11E+08  2.36E+08  3.05E+08  3.31E+08  3.94E+08  1.12E-03  577
 INTERMEDIATE PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL COST                  1.0000  9.11E+06  8.63E+06  1.26E+07  1.40E+07  1.80E+07  2.00E+07  2.08E+07  1.14E-03  140
 CROP DISPOSAL COST                  1.0000  1.40E+08  1.20E+08  2.08E+08  2.19E+08  2.48E+08  2.62E+08  2.93E+08  1.14E-03  930

                      PROB                QUANTILES                 PEAK     PEAK  PEAK
                    NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH  CONSEQ    PROB  TRIAL
ECONOMIC COST MEASURES ($)  48.3-64.4 km
 TOTAL ECONOMIC COSTS                1.0000  5.53E+10  4.40E+10  1.03E+11  1.13E+11  1.42E+11  1.56E+11  1.92E+11  1.14E-03  697
 POP.-DEPENDENT COSTS                1.0000  5.49E+10  4.33E+10  1.02E+11  1.13E+11  1.41E+11  1.56E+11  1.91E+11  1.14E-03  697
 FARM-DEPENDENT COSTS                1.0000  4.48E+08  4.05E+08  6.41E+08  7.19E+08  8.54E+08  9.20E+08  1.08E+09  1.14E-03    7
 POP.-DEPENDENT DECONTAMINATION COST 1.0000  8.87E+09  7.60E+09  1.60E+10  2.04E+10  2.57E+10  2.84E+10  3.14E+10  1.14E-03  387
 FARM-DEPENDENT DECONTAMINATION COST 1.0000  1.34E+08  1.15E+08  2.02E+08  2.19E+08  2.65E+08  2.87E+08  3.14E+08  1.14E-03  387
 POP.-DEPENDENT INTERDICTION COST    1.0000  4.44E+10  3.69E+10  8.12E+10  9.58E+10  1.16E+11  1.24E+11  1.44E+11  1.13E-03  714
 FARM-DEPENDENT INTERDICTION COST    1.0000  1.71E+08  1.47E+08  2.52E+08  2.87E+08  3.29E+08  3.45E+08  4.16E+08  3.23E-04    6
 POP.-DEPENDENT CONDEMNATION COST    0.1546  1.54E+09  0.00E+00  3.28E+09  8.14E+09  3.85E+10  6.57E+10  1.06E+11  1.14E-03  886
 FARM-DEPENDENT CONDEMNATION COST    0.1829  1.48E+07  0.00E+00  5.80E+07  1.01E+08  1.53E+08  1.83E+08  2.27E+08  1.43E-04   10
 EMERGENCY PHASE COST                0.8759  1.05E+08  5.60E+07  2.74E+08  3.70E+08  5.96E+08  6.86E+08  1.05E+09  1.13E-03  714
 INTERMEDIATE PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL COST                  1.0000  5.37E+06  5.14E+06  1.02E+07  1.06E+07  1.17E+07  1.22E+07  1.38E+07  5.99E-04  943
 CROP DISPOSAL COST                  1.0000  1.23E+08  1.08E+08  1.63E+08  1.94E+08  2.26E+08  2.40E+08  2.71E+08  1.14E-03  930

                      PROB                QUANTILES                 PEAK     PEAK  PEAK
                    NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH  CONSEQ    PROB  TRIAL
ECONOMIC COST MEASURES ($)  64.4-80.5 km
 TOTAL ECONOMIC COSTS                1.0000  4.77E+10  4.40E+10  7.54E+10  8.57E+10  1.05E+11  1.09E+11  1.19E+11  1.14E-03  387
 POP.-DEPENDENT COSTS                1.0000  4.72E+10  4.35E+10  7.48E+10  8.46E+10  1.04E+11  1.08E+11  1.18E+11  1.14E-03  387
 FARM-DEPENDENT COSTS                1.0000  5.24E+08  5.04E+08  7.52E+08  8.30E+08  1.02E+09  1.06E+09  1.16E+09  1.14E-03    2
 POP.-DEPENDENT DECONTAMINATION COST 1.0000  7.14E+09  7.06E+09  1.10E+10  1.23E+10  1.59E+10  1.77E+10  2.07E+10  1.14E-03  100
 FARM-DEPENDENT DECONTAMINATION COST 1.0000  1.47E+08  1.24E+08  2.18E+08  2.44E+08  3.07E+08  3.22E+08  3.57E+08  1.13E-03    1
 POP.-DEPENDENT INTERDICTION COST    1.0000  3.92E+10  3.57E+10  6.39E+10  7.20E+10  8.28E+10  8.80E+10  1.00E+11  1.14E-03  387
 FARM-DEPENDENT INTERDICTION COST    1.0000  2.08E+08  1.97E+08  3.15E+08  3.38E+08  3.99E+08  4.29E+08  5.32E+08  1.14E-03  930
 POP.-DEPENDENT CONDEMNATION COST    0.0960  8.38E+08  0.00E+00  0.00E+00  7.17E+09  1.85E+10  2.35E+10  4.02E+10  1.14E-03  886
 FARM-DEPENDENT CONDEMNATION COST    0.1132  9.39E+06  0.00E+00  3.51E+07  7.49E+07  1.58E+08  2.02E+08  2.56E+08  1.11E-03   75
 EMERGENCY PHASE COST                0.5737  4.83E+07  1.00E+07  1.44E+08  2.01E+08  2.71E+08  3.10E+08  4.43E+08  1.14E-03  601
 INTERMEDIATE PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL COST                  1.0000  6.48E+06  5.94E+06  1.11E+07  1.19E+07  1.41E+07  1.52E+07  1.78E+07  1.14E-03  944
 CROP DISPOSAL COST                  1.0000  1.52E+08  1.27E+08  2.17E+08  2.37E+08  2.90E+08  3.12E+08  3.54E+08  1.14E-03  930

                      PROB                QUANTILES                 PEAK     PEAK  PEAK
                    NON-ZERO  MEAN     50TH     90TH     95TH     99TH    99.5TH  CONSEQ    PROB  TRIAL
ECONOMIC COST MEASURES ($)  80.5-161 km
 TOTAL ECONOMIC COSTS                1.0000  2.46E+11  2.14E+11  4.07E+11  4.77E+11  6.20E+11  6.90E+11  8.19E+11  1.14E-03  387
 POP.-DEPENDENT COSTS                1.0000  2.43E+11  2.13E+11  4.06E+11  4.77E+11  6.20E+11  6.90E+11  8.11E+11  1.14E-03  387
 FARM-DEPENDENT COSTS                1.0000  3.25E+09  3.05E+09  4.72E+09  5.29E+09  6.38E+09  6.92E+09  7.83E+09  1.14E-03  930
 POP.-DEPENDENT DECONTAMINATION COST 1.0000  3.27E+10  2.58E+10  6.85E+10  7.54E+10  9.12E+10  9.89E+10  1.22E+11  1.14E-03  884
 FARM-DEPENDENT DECONTAMINATION COST 1.0000  7.60E+08  6.77E+08  1.10E+09  1.21E+09  1.50E+09  1.64E+09  2.27E+09  1.13E-03    1
 POP.-DEPENDENT INTERDICTION COST    1.0000  2.10E+11  1.93E+11  3.67E+11  4.15E+11  5.36E+11  5.84E+11  7.29E+11  1.14E-03  387
 FARM-DEPENDENT INTERDICTION COST    1.0000  1.40E+09  1.18E+09  2.21E+09  2.50E+09  3.19E+09  3.43E+09  4.02E+09  1.14E-03  930
 POP.-DEPENDENT CONDEMNATION COST    0.0258  1.13E+08  0.00E+00  0.00E+00  0.00E+00  4.53E+09  5.83E+09  2.80E+10  2.38E-04  413
 FARM-DEPENDENT CONDEMNATION COST    0.0368  7.63E+06  0.00E+00  0.00E+00  0.00E+00  2.38E+08  2.96E+08  5.47E+08  1.15E-03  462
 EMERGENCY PHASE COST                0.2494  5.72E+07  0.00E+00  1.13E+08  2.92E+08  1.20E+09  1.53E+09  2.34E+09  1.15E-03  711
 INTERMEDIATE PHASE COST             0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
 MILK DISPOSAL COST                  1.0000  4.25E+07  3.44E+07  8.43E+07  9.97E+07  1.16E+08  1.24E+08  1.43E+08  1.14E-03  930
 CROP DISPOSAL COST                  1.0000  1.04E+09  1.01E+09  1.39E+09  1.60E+09  2.09E+09  2.22E+09  2.54E+09  1.14E-03  930

                      PROB                QUANTILES                 PEAK     PEAK  PEAK
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                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH     99.5TH    CONSEQ     PROB  TRIAL
ECONOMIC COST MEASURES ($)    161-322 km
 TOTAL ECONOMIC COSTS                0.9906  3.30E+11  1.14E+11  1.02E+12  1.10E+12  1.31E+12  1.41E+12  1.66E+12  1.14E-03  402
  POP.-DEPENDENT COSTS               0.9800  3.25E+11  1.09E+11  1.02E+12  1.10E+12  1.31E+12  1.41E+12  1.66E+12  1.14E-03  402
  FARM-DEPENDENT COSTS               0.9906  4.81E+09  3.94E+09  8.84E+09  1.06E+10  1.36E+10  1.52E+10  1.92E+10  1.14E-03  981
  POP.-DEPENDENT DECONTAMINATION COST  0.9800  3.71E+10  1.17E+10  1.08E+11  1.42E+11  2.35E+11  2.75E+11  3.29E+11  1.11E-03  290
  FARM-DEPENDENT DECONTAMINATION COST  0.9800  8.18E+08  5.87E+08  1.71E+09  2.38E+09  4.35E+09  5.14E+09  5.84E+09  1.13E-03  480
  POP.-DEPENDENT INTERDICTION COST    0.9800  2.88E+11  1.01E+11  9.02E+11  1.02E+12  1.11E+12  1.14E+12  1.35E+12  1.52E-04  137
  FARM-DEPENDENT INTERDICTION COST    0.9906  2.37E+09  1.75E+09  4.99E+09  6.47E+09  8.56E+09  9.46E+09  1.15E+10  1.14E-03  981
  POP.-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  EMERGENCY PHASE COST               0.0134  1.91E+06  0.00E+00  0.00E+00  0.00E+00  2.47E+07  5.97E+07  6.19E+08  2.26E-03  545
  INTERMEDIATE PHASE COST            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  MILK DISPOSAL COST                 0.9814  4.16E+07  1.88E+07  1.06E+08  1.17E+08  1.47E+08  1.62E+08  2.05E+08  1.14E-03  905
  CROP DISPOSAL COST                 0.9906  1.58E+09  1.32E+09  2.91E+09  3.26E+09  4.07E+09  4.48E+09  5.50E+09  1.14E-03  387

                          PROB                  QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH     99.5TH    CONSEQ     PROB  TRIAL
ECONOMIC COST MEASURES ($)    322-805 km
 TOTAL ECONOMIC COSTS                0.6999  1.03E+11  3.35E+10  2.90E+11  4.22E+11  6.84E+11  7.72E+11  8.22E+11  3.43E-03  591
  POP.-DEPENDENT COSTS               0.5774  9.29E+10  2.48E+10  2.67E+11  4.03E+11  6.64E+11  7.61E+11  8.13E+11  3.43E-03  591
  FARM-DEPENDENT COSTS               0.6999  9.77E+09  4.64E+09  2.65E+10  3.42E+10  5.27E+10  5.77E+10  7.25E+10  1.15E-03  911
  POP.-DEPENDENT DECONTAMINATION COST  0.5774  9.49E+09  2.48E+09  2.73E+10  4.11E+10  6.84E+10  9.30E+10  9.74E+10  4.54E-03  546
  FARM-DEPENDENT DECONTAMINATION COST  0.5774  1.01E+09  2.36E+08  3.19E+09  4.16E+09  6.26E+09  7.17E+09  9.41E+09  1.14E-03  568
  POP.-DEPENDENT INTERDICTION COST    0.5774  8.34E+10  2.16E+10  2.50E+11  3.55E+11  5.55E+11  6.19E+11  7.32E+11  2.28E-03  591
  FARM-DEPENDENT INTERDICTION COST    0.6999  5.98E+09  2.16E+09  1.68E+10  2.33E+10  3.49E+10  3.92E+10  5.23E+10  1.15E-03  911
  POP.-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  EMERGENCY PHASE COST               0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  INTERMEDIATE PHASE COST            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  MILK DISPOSAL COST                 0.6932  1.11E+08  2.14E+07  3.74E+08  5.63E+08  7.57E+08  7.99E+08  8.98E+08  1.13E-03  859
  CROP DISPOSAL COST                 0.6999  2.67E+09  1.43E+09  7.31E+09  9.12E+09  1.27E+10  1.43E+10  1.86E+10  1.15E-03  911

                          PROB                  QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH     99.5TH    CONSEQ     PROB  TRIAL
ECONOMIC COST MEASURES ($)   805-1609 km
 TOTAL ECONOMIC COSTS                0.0464  1.78E+09  0.00E+00  0.00E+00  0.00E+00  3.86E+10  6.01E+10  4.35E+11  1.13E-03  724
  POP.-DEPENDENT COSTS               0.0060  8.20E+08  0.00E+00  0.00E+00  0.00E+00  0.00E+00  5.33E+09  4.12E+11  1.13E-03  724
  FARM-DEPENDENT COSTS               0.0464  9.61E+08  0.00E+00  0.00E+00  0.00E+00  3.41E+10  5.30E+10  8.47E+10  1.13E-03  396
  POP.-DEPENDENT DECONTAMINATION COST  0.0060  8.25E+07  0.00E+00  0.00E+00  0.00E+00  0.00E+00  5.33E+08  4.15E+10  1.13E-03  724
  FARM-DEPENDENT DECONTAMINATION COST  0.0060  1.51E+07  0.00E+00  0.00E+00  0.00E+00  0.00E+00  1.14E+09  6.32E+09  1.13E-03  724
  POP.-DEPENDENT INTERDICTION COST    0.0060  7.37E+08  0.00E+00  0.00E+00  0.00E+00  0.00E+00  3.31E+09  3.71E+11  1.13E-03  724
  FARM-DEPENDENT INTERDICTION COST    0.0464  6.76E+08  0.00E+00  0.00E+00  0.00E+00  2.40E+10  3.28E+10  5.83E+10  1.13E-03  396
  POP.-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  FARM-DEPENDENT CONDEMNATION COST    0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  EMERGENCY PHASE COST               0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  INTERMEDIATE PHASE COST            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
  MILK DISPOSAL COST                 0.0413  8.53E+06  0.00E+00  0.00E+00  0.00E+00  2.52E+08  5.81E+08  8.98E+08  1.14E-03  860
  CROP DISPOSAL COST                 0.0464  2.61E+08  0.00E+00  0.00E+00  0.00E+00  1.06E+10  1.42E+10  2.65E+10  1.13E-03  396

                          PROB                  QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH     99.5TH    CONSEQ     PROB  TRIAL
MAXIMUM LONG-TERM ACTION DISTANCE (km)
 FARM-DEPENDENT DECONTAMINATION DIST.   1.0000  5.40E+02  5.55E+02  7.82E+02  8.31E+02  9.57E+02    ****    1.61E+03  6.02E-03  724
 POP.-DEPENDENT DECONTAMINATION DIST.   1.0000  5.40E+02  5.55E+02  7.82E+02  8.31E+02  9.57E+02    ****    1.61E+03  6.02E-03  724
 FARM-DEPENDENT INTERDICTION DIST.     1.0000  6.47E+02  7.04E+02  8.93E+02  9.89E+02    ****      ****    1.61E+03  4.64E-02  394
 POP.-DEPENDENT INTERDICTION DIST.     1.0000  5.40E+02  5.55E+02  7.82E+02  8.31E+02  9.57E+02    ****    1.61E+03  6.02E-03  724
 FARM-DEPENDENT CONDEMNATION DIST.     1.0000  4.74E+01  3.69E+01  7.35E+01  9.10E+01  1.25E+02  1.41E+02  1.61E+02  2.28E-03  319
 POP.-DEPENDENT CONDEMNATION DIST.     1.0000  4.44E+01  3.59E+01  6.93E+01  8.38E+01    ****      ****    1.13E+02  2.58E-02   71
 MILK DISPOSAL DIST.                 1.0000  6.35E+02  6.94E+02  8.81E+02  9.73E+02    ****      ****    1.61E+03  4.13E-02  724
 CROP DISPOSAL DIST.                 1.0000  6.47E+02  7.04E+02  8.93E+02  9.89E+02    ****      ****    1.61E+03  4.64E-02  394

                          PROB                  QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH     99.5TH    CONSEQ     PROB  TRIAL
AFFECTED AREA/POPULATION      0-1609 km
 FARM DECONTAMINATION (ha)             1.0000  1.98E+06  1.44E+06  3.79E+06  4.88E+06  6.25E+06  6.91E+06  9.41E+06  1.14E-03  568
 POP. DECONTAMINATION (INDIVIDUALS)    1.0000  1.09E+07  7.96E+06  2.27E+07  2.70E+07  3.27E+07  3.44E+07  3.84E+07  1.14E-03  258
 POP. DECONTAMINATION AREA (ha)        1.0000  4.66E+06  3.78E+06  8.53E+06  1.01E+07  1.30E+07  1.45E+07  1.83E+07  1.14E-03  235
 FARM INTERDICTION (ha)               1.0000  4.67E+06  2.49E+06  1.09E+07  1.40E+07  2.41E+07  2.92E+07  5.99E+07  1.13E-03  396
 POP. INTERDICTION (INDIVIDUALS)       1.0000  1.09E+07  7.96E+06  2.27E+07  2.70E+07  3.27E+07  3.44E+07  3.84E+07  1.14E-03  258
 POP. INTERDICTION AREA (ha)          1.0000  4.66E+06  3.78E+06  8.53E+06  1.01E+07  1.30E+07  1.45E+07  1.83E+07  1.14E-03  235
 FARM CONDEMNATION (ha)               1.0000  2.82E+04  2.36E+04  4.66E+04  5.98E+04  9.55E+04  1.07E+05  1.30E+05  1.15E-03  462
 POP. CONDEMNATION (INDIVIDUALS)       1.0000  7.17E+04  4.62E+04  1.47E+05  2.10E+05  3.47E+05  4.11E+05  6.00E+05  1.14E-03  886
 POP. CONDEMNATION AREA (ha)          1.0000  2.88E+04  2.28E+04  5.00E+04  7.09E+04  1.04E+05  1.13E+05  1.35E+05  1.14E-03  463
 MILK DISPOSAL AREA (ha)              1.0000  4.08E+06  2.29E+06  1.01E+07  1.27E+07  2.15E+07  2.66E+07  3.62E+07  1.13E-03  859
 CROP DISPOSAL AREA (ha)              1.0000  4.70E+06  2.54E+06  1.09E+07  1.41E+07  2.41E+07  2.92E+07  6.00E+07  1.13E-03  396

                          PROB                  QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH     99.5TH    CONSEQ     PROB  TRIAL
AFFECTED AREA/POPULATION      0-16.1 km
 FARM DECONTAMINATION (ha)             1.0000  6.76E+03  6.09E+03  1.13E+04  1.25E+04  1.60E+04  1.78E+04  2.11E+04  1.15E-03  896
 POP. DECONTAMINATION (INDIVIDUALS)    1.0000  9.69E+03  8.52E+03  1.61E+04  2.01E+04  2.31E+04  2.45E+04  2.79E+04  1.14E-03  776
 POP. DECONTAMINATION AREA (ha)        1.0000  8.26E+03  7.37E+03  1.28E+04  1.47E+04  2.03E+04  2.18E+04  2.55E+04  1.14E-03  910
 FARM INTERDICTION (ha)               1.0000  7.23E+03  6.60E+03  1.18E+04  1.33E+04  1.75E+04  1.97E+04  2.68E+04  1.15E-03  896
 POP. INTERDICTION (INDIVIDUALS)       1.0000  9.69E+03  8.52E+03  1.61E+04  2.01E+04  2.31E+04  2.45E+04  2.79E+04  1.14E-03  776
 POP. INTERDICTION AREA (ha)          1.0000  8.26E+03  7.37E+03  1.28E+04  1.47E+04  2.03E+04  2.18E+04  2.55E+04  1.14E-03  910
 FARM CONDEMNATION (ha)               1.0000  1.08E+04  1.03E+04  1.34E+04  1.50E+04  1.95E+04  2.11E+04  2.44E+04  1.12E-03  577
 POP. CONDEMNATION (INDIVIDUALS)       1.0000  1.19E+04  1.07E+04  1.61E+04  1.93E+04  2.17E+04  2.25E+04  2.73E+04  1.43E-04    8
 POP. CONDEMNATION AREA (ha)          1.0000  1.04E+04  9.81E+03  1.36E+04  1.55E+04  2.02E+04  2.08E+04  2.40E+04  1.43E-04    8
 MILK DISPOSAL AREA (ha)              1.0000  1.80E+04  1.62E+04  2.48E+04  2.76E+04  3.12E+04  3.20E+04  3.38E+04  1.14E-03  936
 CROP DISPOSAL AREA (ha)              1.0000  1.81E+04  1.63E+04  2.51E+04  2.82E+04  3.13E+04  3.21E+04  3.38E+04  1.14E-03  936

                          PROB                  QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH     99.5TH    CONSEQ     PROB  TRIAL
AFFECTED AREA/POPULATION    16.1-32.2 km
 FARM DECONTAMINATION (ha)             1.0000  3.19E+04  3.02E+04  5.28E+04  5.79E+04  7.12E+04  7.56E+04  8.61E+04  1.14E-03  387
 POP. DECONTAMINATION (INDIVIDUALS)    1.0000  1.42E+05  1.20E+05  2.33E+05  2.67E+05  3.22E+05  3.39E+05  3.76E+05  1.14E-03  944
 POP. DECONTAMINATION AREA (ha)        1.0000  4.30E+04  3.78E+04  7.12E+04  7.77E+04  9.52E+04  1.03E+05  1.16E+05  1.14E-03  387
 FARM INTERDICTION (ha)               1.0000  3.40E+04  3.13E+04  5.69E+04  6.47E+04  7.57E+04  7.92E+04  8.73E+04  1.14E-03  387
 POP. INTERDICTION (INDIVIDUALS)       1.0000  1.42E+05  1.20E+05  2.33E+05  2.67E+05  3.22E+05  3.39E+05  3.76E+05  1.14E-03  944
 POP. INTERDICTION AREA (ha)          1.0000  4.30E+04  3.78E+04  7.12E+04  7.77E+04  9.52E+04  1.03E+05  1.16E+05  1.14E-03  387
 FARM CONDEMNATION (ha)               1.0000  1.10E+04  1.01E+04  1.74E+04  2.09E+04  2.69E+04  3.00E+04  4.27E+04  1.43E-04   76
 POP. CONDEMNATION (INDIVIDUALS)       0.9989  3.40E+04  2.61E+04  7.02E+04  8.24E+04  1.06E+05  1.11E+05  1.46E+05  1.14E-04  327
 POP. CONDEMNATION AREA (ha)          0.9989  1.15E+04  1.04E+04  1.70E+04  2.09E+04  3.07E+04  3.29E+04  4.69E+04  1.43E-04    8
 MILK DISPOSAL AREA (ha)              1.0000  4.46E+04  4.18E+04  6.92E+04  7.29E+04  8.08E+04  8.45E+04  9.29E+04  1.14E-03  936
 CROP DISPOSAL AREA (ha)              1.0000  4.50E+04  4.21E+04  7.01E+04  7.32E+04  8.10E+04  8.46E+04  9.29E+04  1.14E-03  936

                          PROB                  QUANTILES                  PEAK     PEAK  PEAK
                        NON-ZERO   MEAN     50TH     90TH     95TH     99TH     99.5TH    CONSEQ     PROB  TRIAL
AFFECTED AREA/POPULATION    32.2-48.3 km
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FARM DECONTAMINATION (ha)          1.0000  5.34E+04  5.13E+04  7.98E+04  8.84E+04  1.05E+05  1.09E+05  1.23E+05  5.99E-04  943
POP. DECONTAMINATION (INDIVIDUALS)     1.0000  3.34E+05  3.12E+05  5.28E+05  5.81E+05  7.12E+05  7.45E+05  8.20E+05  1.14E-03  930
POP. DECONTAMINATION AREA (ha)       1.0000  8.86E+04  8.29E+04  1.16E+05  1.26E+05  1.54E+05  1.67E+05  2.08E+05  1.14E-03  387
FARM INTERDICTION (ha)            1.0000  5.69E+04  5.38E+04  8.71E+04  1.00E+05  1.13E+05  1.19E+05  1.34E+05  1.14E-03  930
POP. INTERDICTION (INDIVIDUALS)      1.0000  3.34E+05  3.12E+05  5.28E+05  5.81E+05  7.12E+05  7.45E+05  8.20E+05  1.14E-03  930
POP. INTERDICTION AREA (ha)         1.0000  8.86E+04  8.29E+04  1.16E+05  1.26E+05  1.54E+05  1.67E+05  2.08E+05  1.14E-03  387
FARM CONDEMNATION (ha)            0.7323  3.67E+03  1.83E+03  9.99E+03  1.26E+04  2.05E+04  2.18E+04  3.26E+04  1.43E-04   76
POP. CONDEMNATION (INDIVIDUALS)      0.6345  1.65E+04  3.40E+03  5.16E+04  7.58E+04  1.16E+05  1.26E+05  1.53E+05  1.14E-03   85
POP. CONDEMNATION AREA (ha)         0.6345  3.84E+03  1.48E+03  1.07E+04  1.46E+04  2.41E+04  2.78E+04  3.66E+04  1.14E-03  886
MILK DISPOSAL AREA (ha)            1.0000  5.99E+04  5.67E+04  8.84E+04  1.00E+05  1.13E+05  1.19E+05  1.34E+05  1.14E-03  930
CROP DISPOSAL AREA (ha)            1.0000  6.05E+04  5.71E+04  8.99E+04  1.01E+05  1.14E+05  1.20E+05  1.34E+05  1.14E-03  930

                         PROB                 QUANTILES                   PEAK     PEAK   PEAK
                       NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION    48.3-64.4 km
FARM DECONTAMINATION (ha)          1.0000  6.10E+04  5.68E+04  9.22E+04  1.03E+05  1.18E+05  1.25E+05  1.41E+05  1.14E-03  387
POP. DECONTAMINATION (INDIVIDUALS)     1.0000  6.70E+05  5.64E+05  1.10E+06  1.20E+06  1.48E+06  1.61E+06  1.95E+06  1.14E-03  387
POP. DECONTAMINATION AREA (ha)       1.0000  1.24E+05  1.10E+05  1.83E+05  2.08E+05  2.42E+05  2.58E+05  2.97E+05  1.14E-03  387
FARM INTERDICTION (ha)            1.0000  6.69E+04  6.24E+04  1.03E+05  1.11E+05  1.31E+05  1.40E+05  1.63E+05  1.14E-03  930
POP. INTERDICTION (INDIVIDUALS)      1.0000  6.70E+05  5.64E+05  1.10E+06  1.20E+06  1.48E+06  1.61E+06  1.95E+06  1.14E-03  387
POP. INTERDICTION AREA (ha)         1.0000  1.24E+05  1.10E+05  1.83E+05  2.08E+05  2.42E+05  2.58E+05  2.97E+05  1.14E-03  387
FARM CONDEMNATION (ha)            0.1829  1.26E+03  0.00E+00  4.95E+03  8.81E+03  1.25E+04  1.39E+04  1.71E+04  1.38E-03   85
POP. CONDEMNATION (INDIVIDUALS)      0.1546  5.47E+03  0.00E+00  1.24E+04  3.24E+04  1.16E+05  1.86E+05  2.84E+05  1.14E-03  886
POP. CONDEMNATION AREA (ha)         0.1546  1.63E+03  0.00E+00  5.82E+03  1.22E+04  2.50E+04    ****    2.66E+04  7.69E-03   93
MILK DISPOSAL AREA (ha)            1.0000  6.71E+04  6.32E+04  1.03E+05  1.11E+05  1.30E+05  1.40E+05  1.63E+05  1.14E-03  930
CROP DISPOSAL AREA (ha)            1.0000  6.81E+04  6.40E+04  1.04E+05  1.11E+05  1.31E+05  1.40E+05  1.63E+05  1.14E-03  930

                         PROB                 QUANTILES                   PEAK     PEAK   PEAK
                       NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION    64.4-80.5 km
FARM DECONTAMINATION (ha)          1.0000  7.25E+04  7.06E+04  1.06E+05  1.13E+05  1.31E+05  1.40E+05  1.61E+05  1.14E-03  387
POP. DECONTAMINATION (INDIVIDUALS)     1.0000  6.00E+05  5.65E+05  9.84E+05  1.07E+06  1.26E+06  1.35E+06  1.58E+06  1.14E-03  387
POP. DECONTAMINATION AREA (ha)       1.0000  1.47E+05  1.22E+05  2.09E+05  2.27E+05  2.74E+05  2.97E+05  3.39E+05  1.14E-03  387
FARM INTERDICTION (ha)            1.0000  8.26E+04  7.63E+04  1.14E+05  1.25E+05  1.53E+05  1.66E+05  2.12E+05  1.14E-03  930
POP. INTERDICTION (INDIVIDUALS)      1.0000  6.00E+05  5.65E+05  9.84E+05  1.07E+06  1.26E+06  1.35E+06  1.58E+06  1.14E-03  387
POP. INTERDICTION AREA (ha)         1.0000  1.47E+05  1.22E+05  2.09E+05  2.27E+05  2.74E+05  2.97E+05  3.39E+05  1.14E-03  387
FARM CONDEMNATION (ha)            0.1132  8.08E+02  0.00E+00  3.20E+03  6.22E+03  1.22E+04  1.43E+04  2.29E+04  1.11E-03   75
POP. CONDEMNATION (INDIVIDUALS)      0.0960  3.34E+03  0.00E+00  0.00E+00  2.81E+04  6.87E+04  1.05E+05  1.22E+05  1.09E-03  287
POP. CONDEMNATION AREA (ha)         0.0960  9.66E+02  0.00E+00  0.00E+00  7.72E+03  2.01E+04  2.28E+04  3.00E+04  1.11E-03   75
MILK DISPOSAL AREA (ha)            1.0000  8.17E+04  7.55E+04  1.14E+05  1.24E+05  1.52E+05  1.66E+05  2.12E+05  1.14E-03  930
CROP DISPOSAL AREA (ha)            1.0000  8.34E+04  7.67E+04  1.15E+05  1.25E+05  1.53E+05  1.66E+05  2.12E+05  1.14E-03  930

                         PROB                 QUANTILES                   PEAK     PEAK   PEAK
                       NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION    80.5-161 km
FARM DECONTAMINATION (ha)          1.0000  4.40E+05  3.93E+05  6.68E+05  7.36E+05  8.63E+05  9.25E+05  1.19E+06  1.13E-03    1
POP. DECONTAMINATION (INDIVIDUALS)     1.0000  3.28E+06  3.03E+06  5.84E+06  6.54E+06  8.45E+06  9.42E+06  1.15E+07  1.14E-03  387
POP. DECONTAMINATION AREA (ha)       1.0000  8.82E+05  8.22E+05  1.16E+06  1.26E+06  1.53E+06  1.67E+06  2.42E+06  1.14E-03  387
FARM INTERDICTION (ha)            1.0000  5.66E+05  5.23E+05  8.91E+05  1.02E+06  1.24E+06  1.34E+06  1.60E+06  1.14E-03  930
POP. INTERDICTION (INDIVIDUALS)      1.0000  3.28E+06  3.03E+06  5.84E+06  6.54E+06  8.45E+06  9.42E+06  1.15E+07  1.14E-03  387
POP. INTERDICTION AREA (ha)         1.0000  8.82E+05  8.22E+05  1.16E+06  1.26E+06  1.53E+06  1.67E+06  2.42E+06  1.14E-03  387
FARM CONDEMNATION (ha)            0.0368  6.53E+02  0.00E+00  0.00E+00  0.00E+00  2.16E+04  2.71E+04  5.02E+04  1.15E-03  462
POP. CONDEMNATION (INDIVIDUALS)      0.0258  4.64E+02  0.00E+00  0.00E+00  0.00E+00  1.75E+04  2.41E+04  8.07E+04  2.38E-04  413
POP. CONDEMNATION AREA (ha)         0.0258  5.32E+02  0.00E+00  0.00E+00  0.00E+00  2.09E+04    ****    3.48E+04  5.10E-03   84
MILK DISPOSAL AREA (ha)            1.0000  5.43E+05  5.06E+05  8.40E+05  9.66E+05  1.21E+06  1.32E+06  1.60E+06  1.14E-03  930
CROP DISPOSAL AREA (ha)            1.0000  5.67E+05  5.23E+05  8.91E+05  1.02E+06  1.24E+06  1.34E+06  1.60E+06  1.14E-03  930

                         PROB                 QUANTILES                   PEAK     PEAK   PEAK
                       NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION     161-322 km
FARM DECONTAMINATION (ha)          0.9800  5.49E+05  3.98E+05  1.10E+06  1.49E+06  2.34E+06  2.63E+06  3.32E+06  1.13E-03  480
POP. DECONTAMINATION (INDIVIDUALS)     0.9800  4.53E+06  1.51E+06  1.20E+07  1.47E+07  2.01E+07  2.03E+07  2.13E+07  1.52E-04  137
POP. DECONTAMINATION AREA (ha)       0.9800  1.41E+06  1.14E+06  2.52E+06  3.10E+06  4.28E+06  4.91E+06  6.00E+06  1.14E-03  378
FARM INTERDICTION (ha)            0.9906  9.84E+05  7.13E+05  2.09E+06  2.61E+06  3.48E+06  3.80E+06  4.59E+06  1.14E-03  981
POP. INTERDICTION (INDIVIDUALS)      0.9800  4.53E+06  1.51E+06  1.20E+07  1.47E+07  2.01E+07  2.03E+07  2.13E+07  1.52E-04  137
POP. INTERDICTION AREA (ha)         0.9800  1.41E+06  1.14E+06  2.52E+06  3.10E+06  4.28E+06  4.91E+06  6.00E+06  1.14E-03  378
FARM CONDEMNATION (ha)            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION (INDIVIDUALS)      0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION AREA (ha)         0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
MILK DISPOSAL AREA (ha)            0.9906  8.89E+05  6.30E+05  1.91E+06  2.38E+06  3.36E+06  3.71E+06  4.59E+06  1.14E-03  981
CROP DISPOSAL AREA (ha)            0.9906  9.84E+05  7.13E+05  2.09E+06  2.61E+06  3.48E+06  3.80E+06  4.59E+06  1.14E-03  981

                         PROB                 QUANTILES                   PEAK     PEAK   PEAK
                       NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION     322-805 km
FARM DECONTAMINATION (ha)          0.5774  7.50E+05  1.73E+05  2.31E+06  3.14E+06  4.57E+06  5.30E+06  7.07E+06  1.14E-03  568
POP. DECONTAMINATION (INDIVIDUALS)     0.5774  1.31E+06  3.57E+05  3.85E+06  5.86E+06  9.46E+06  1.03E+07  1.15E+07  1.15E-03  591
POP. DECONTAMINATION AREA (ha)       0.5774  1.91E+06  8.52E+05  5.47E+06  6.92E+06  8.88E+06  9.86E+06  1.40E+07  1.14E-03  235
FARM INTERDICTION (ha)            0.6999  2.57E+06  8.64E+05  7.35E+06  1.00E+07  1.35E+07  1.53E+07  2.08E+07  1.15E-03  911
POP. INTERDICTION (INDIVIDUALS)      0.5774  1.31E+06  3.57E+05  3.85E+06  5.86E+06  9.46E+06  1.03E+07  1.15E+07  1.15E-03  591
POP. INTERDICTION AREA (ha)         0.5774  1.91E+06  8.52E+05  5.47E+06  6.92E+06  8.88E+06  9.86E+06  1.40E+07  1.14E-03  235
FARM CONDEMNATION (ha)            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION (INDIVIDUALS)      0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION AREA (ha)         0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
MILK DISPOSAL AREA (ha)            0.6932  2.14E+06  6.90E+05  6.47E+06  8.81E+06  1.30E+07  1.49E+07  2.08E+07  1.15E-03  911
CROP DISPOSAL AREA (ha)            0.6999  2.57E+06  8.64E+05  7.35E+06  1.00E+07  1.35E+07  1.53E+07  2.08E+07  1.15E-03  911

                         PROB                 QUANTILES                   PEAK     PEAK   PEAK
                       NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
AFFECTED AREA/POPULATION    805-1609 km
FARM DECONTAMINATION (ha)          0.0060  1.14E+04  0.00E+00  0.00E+00  0.00E+00  0.00E+00  7.49E+05  4.75E+06  1.13E-03  724
POP. DECONTAMINATION (INDIVIDUALS)     0.0060  1.16E+04  0.00E+00  0.00E+00  0.00E+00  0.00E+00  7.49E+04  5.84E+06  1.13E-03  724
POP. DECONTAMINATION AREA (ha)       0.0060  4.08E+04  0.00E+00  0.00E+00  0.00E+00  0.00E+00  4.74E+06  7.89E+06  3.75E-03  785
FARM INTERDICTION (ha)            0.0464  2.99E+05  0.00E+00  0.00E+00  0.00E+00  1.01E+07  1.25E+07  4.21E+07  1.13E-03  396
POP. INTERDICTION (INDIVIDUALS)      0.0060  1.16E+04  0.00E+00  0.00E+00  0.00E+00  0.00E+00  7.49E+04  5.84E+06  1.13E-03  724
POP. INTERDICTION AREA (ha)         0.0060  4.08E+04  0.00E+00  0.00E+00  0.00E+00  0.00E+00  4.74E+06  7.89E+06  3.75E-03  785
FARM CONDEMNATION (ha)            0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION (INDIVIDUALS)      0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
POP. CONDEMNATION AREA (ha)         0.0000  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00  0.00E+00    0
MILK DISPOSAL AREA (ha)            0.0413  2.33E+05  0.00E+00  0.00E+00  0.00E+00  9.21E+06  1.19E+07  1.81E+07  1.14E-03  860
CROP DISPOSAL AREA (ha)            0.0464  2.99E+05  0.00E+00  0.00E+00  0.00E+00  1.01E+07  1.25E+07  4.21E+07  1.13E-03  396

                         PROB                 QUANTILES                   PEAK     PEAK   PEAK
                       NON-ZERO   MEAN     50TH     90TH     95TH     99TH    99.5TH   CONSEQ    PROB  TRIAL
MAXIMUM ANNUAL FOOD DOSE (EFFECTIVE)
PROJECTED FOR INDIVIDUAL 11.3-16.1 km  0.9989  8.32E-03  4.50E-03  2.02E-02  2.37E-02  3.04E-02  3.08E-02  3.18E-02  1.14E-03  180
PROJECTED FOR INDIVIDUAL 25.8-32.2 km  1.0000  1.04E-02  8.11E-03  2.09E-02  2.37E-02  3.02E-02  3.08E-02  3.22E-02  8.56E-04  150
PROJECTED FOR INDIVIDUAL 40.2-48.3 km  1.0000  1.16E-02  9.73E-03  2.18E-02  2.47E-02  3.04E-02  3.09E-02  3.20E-02  1.15E-03  181
PROJECTED FOR INDIVIDUAL 48.3-64.4 km  1.0000  1.13E-02  9.33E-03  2.11E-02  2.32E-02  2.91E-02  3.03E-02  3.24E-02  1.43E-04  160
PROJECTED FOR INDIVIDUAL 64.4-80.5 km  1.0000  1.19E-02  1.02E-02  2.16E-02  2.42E-02  3.02E-02  3.05E-02  3.24E-02  1.52E-04  137
```

PROJECTED FOR INDIVIDUAL 113-161 km	1.0000	1.37E-02	1.12E-02	2.32E-02	2.67E-02	3.05E-02	3.08E-02	3.24E-02	1.52E-04	188
PROJECTED FOR INDIVIDUAL 241-322 km	1.0000	1.53E-02	1.26E-02	2.63E-02	3.01E-02	3.11E-02	3.15E-02	3.24E-02	1.12E-03	163
PROJECTED FOR INDIVIDUAL 563-805 km	1.0000	1.31E-02	1.08E-02	2.36E-02	2.69E-02	3.05E-02	3.08E-02	3.24E-02	1.43E-04	159
PROJECTED FOR INDIVIDUAL 805-1609 km	1.0000	4.27E-03	1.50E-03	1.13E-02	1.33E-02	1.93E-02	2.22E-02	2.95E-02	1.14E-03	395

	PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	PEAK 99.5TH	PEAK CONSEQ	PEAK PROB	TRIAL
MAXIMUM ANNUAL FOOD DOSE (THYROID)										
PROJECTED FOR INDIVIDUAL 11.3-16.1 km	0.9989	8.30E-03	4.23E-03	2.03E-02	2.38E-02	3.07E-02	3.16E-02	3.35E-02	1.15E-03	227
PROJECTED FOR INDIVIDUAL 25.8-32.2 km	1.0000	1.03E-02	8.10E-03	2.11E-02	2.39E-02	3.04E-02	3.12E-02	3.32E-02	8.56E-04	150
PROJECTED FOR INDIVIDUAL 40.2-48.3 km	1.0000	1.15E-02	9.93E-03	2.17E-02	2.44E-02	3.04E-02	3.12E-02	3.28E-02	1.14E-03	179
PROJECTED FOR INDIVIDUAL 48.3-64.4 km	1.0000	1.13E-02	9.46E-03	2.14E-02	2.39E-02	3.01E-02	3.06E-02	3.33E-02	1.43E-04	160
PROJECTED FOR INDIVIDUAL 64.4-80.5 km	1.0000	1.20E-02	1.03E-02	2.21E-02	2.51E-02	3.05E-02	3.10E-02	3.38E-02	1.52E-04	137
PROJECTED FOR INDIVIDUAL 113-161 km	1.0000	1.39E-02	1.13E-02	2.37E-02	2.75E-02	3.12E-02	3.18E-02	3.34E-02	1.15E-03	226
PROJECTED FOR INDIVIDUAL 241-322 km	1.0000	1.54E-02	1.26E-02	2.67E-02	3.02E-02	3.15E-02	3.20E-02	3.32E-02	1.12E-03	163
PROJECTED FOR INDIVIDUAL 563-805 km	1.0000	1.31E-02	1.07E-02	2.36E-02	2.67E-02	3.05E-02	3.09E-02	3.30E-02	1.43E-04	159
PROJECTED FOR INDIVIDUAL 805-1609 km	1.0000	4.20E-03	1.48E-03	1.13E-02	1.33E-02	1.93E-02	2.21E-02	2.88E-02	1.14E-03	395

**** Indicates that the value is outside resolution of the analysis.
Optionally increase number of trials for better resolution.

Successful completion of MACCS2 was achieved!
This job required a total of 7518.594 CPU seconds

Input processing required 0.469 CPU seconds
Simulation required 7515.969 CPU seconds
Output processing required 2.156 CPU seconds

Attachment 7

NRC Staff SECY-13-0112, Consequence Study of a Beyond Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boil Water Reactor
October 9, 2013 (ML13256A339) (posted on public ADAMS on October 22, 2013)

and

[Final] Consequence Study of a Beyond Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boil Water Reactor
October 2013 (ML13256A342) (posted on public ADAMS on October 23, 2013)

POLICY ISSUE
(Information)

October 9, 2013 SECY-13-0112

FOR: The Commissioners

FROM: Mark A. Satorius
Executive Director for Operations

SUBJECT: CONSEQUENCE STUDY OF A BEYOND-DESIGN-BASIS EARTHQUAKE AFFECTING THE SPENT FUEL POOL FOR A U.S. MARK I BOILING-WATER REACTOR

PURPOSE:

The purpose of this paper is to provide to the Commission the final report of the enclosed study entitled, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor." This paper completes the action directed by the Staff Requirements Memorandum (SRM) dated July 16, 2012 (ADAMS Accession No.: ML121980043). This paper does not address any new commitments or resource implications.

BACKGROUND AND SUMMARY OF STUDY RESULTS:

The March 11, 2011, Tohoku earthquake and subsequent tsunami in Japan resulted in significant damage to the Fukushima Dai-ichi nuclear power station. Although the spent fuel pools (SFPs) and the spent fuel assemblies stored in the pools remained intact, the event led to questions about the safe storage of spent fuel and whether the U.S. Nuclear Regulatory Commission (NRC) should require expedited transfer of spent fuel from pools to dry cask storage at U.S. nuclear power plants.

The purpose of the enclosed study is to help the agency determine if accelerated transfer of spent fuel from the spent fuel pool to dry cask storage significantly reduces risks to public health and safety. The study provides consequence estimates of a hypothetical spent fuel pool

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accident initiated by a low-likelihood seismic event at a reference plant based on the Peach Bottom BWR Mark I spent fuel pool. The study compares high-density and low-density loading conditions and assesses the benefits of post 9/11 mitigation measures.

This study shows the likelihood of a radiological release from the spent fuel after the analyzed severe earthquake at the reference plant to be very low (about 1 time in 10 million years or lower). For the hypothetical releases studied, the study predicted no early fatalities attributable to radiation exposure and individual latent cancer fatality risks are low; however, the study suggested that extensive protective actions may be needed for both the high and low density pool loadings.

DISCUSSION:

Various risk studies (most recently NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," issued February 2001 [ADAMS Accession No. ML010430066]) have shown that storage of spent fuel in a high-density configuration in SFPs is safe and that the risk to public health and safety from a SFP accident is very low. In response to the events of September 11, 2001, the NRC undertook security assessments of spent fuel storage in pools and casks. Moreover, in conjunction with these post-9/11 security assessments, the NRC issued an order—later codified in Title 10 of the *Code of Federal Regulations*, Section 50.54(hh)(2)—that requires reactor licensees to develop and implement guidance and strategies intended, in part, to maintain or restore SFP cooling capabilities following certain beyond-design-basis events (including explosions and fires). The agency also reviewed 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, Revision 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants - Draft Report for Comment," issued July 2009 (the final NUREG is expected to be published soon).

The staff performed the enclosed consequence study to continue NRC's examination of the risks and consequences of postulated spent fuel pool accidents. This study presents detailed analyses using state-of-the-art, validated, deterministic methods and assumptions as well as probabilistic insights where practical. Previous studies have shown that earthquakes present the dominant risk for spent fuel pools. Therefore, this analysis considered a severe earthquake that would be expected to occur once in 60,000 years with ground motion stronger than the maximum earthquake used for the design basis for the reference plant. The staff considers the ground motion used in this study more challenging for the spent fuel pool structure than that experienced at the Fukushima Dai-ichi nuclear power plant from the earthquake that occurred off the coast of Japan on March 11, 2011. That earthquake did not result in any spent fuel pool leaks. This study's results for the specific reference plant and earthquake analyzed are consistent with past studies' conclusions that spent fuel pools are likely to withstand severe earthquakes without leaking. The regulatory analysis included in the study indicates that expediting movement of spent fuel from the pool does not provide a substantial safety enhancement for the reference plant.

The staff presented the report to the Advisory Committee on Reactor Safeguards (ACRS) on July 9, 2013. The ACRS subsequently submitted a letter to the Commission on July 18, 2013 (ML13198A433), and the NRC provided a response on August 14, 2013 (ML13205A242). The NRC released a draft report to the public in a press release on June 24, 2013 (ML13175A104), and published a *Federal Register* notice (FRN) on July 2, 2013, announcing a 30-day public comment period. The report received responses from 14 different public commenters.

Appendix E of the enclosed report contains those comments and the staff's responses. None of the comments or responses has necessitated making substantial changes to the report.

One of the objectives of the current study is to inform the NRC's Japan Lessons Learned Tier 3 activities. The staff will use the results of the study to inform a broader regulatory analysis of the spent fuel pools at U.S. operating nuclear reactors.

Please note that the staff intends to proceed with making the attached report public, and the report will be subsequently published as a NUREG.

COORDINATION:

The Office of the General Counsel has reviewed the draft study and has no legal objections. The Office of the Chief Financial Officer reviewed this paper for resource implications and has no objections.

/RA/

Mark A. Satorius
Executive Director
for Operations

Enclosure:
As stated

cc: SECY
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/RA/

Mark A. Satorius
Executive Director
for Operations

Enclosure:
As stated

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Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor

October 2013

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US Nuclear Regulatory Commission

FOREWORD

U.S. nuclear power plants are required to be designed with appropriate consideration of the most severe natural phenomena (e.g. floods, earthquakes, tornadoes) historically reported for their location and surrounding regions, with sufficient margin, to ensure that important safety functions can be performed. As part of our mission to protect public health and safety, the U.S. Nuclear Regulatory Commission (NRC) uses advanced computer modeling and other techniques to study more severe, and highly unlikely, events that go beyond what the plant was designed to withstand to estimate risk to the public and to explore and ensure safety margins.

On March 11, 2011, the Tohoku earthquake and subsequent tsunami in Japan resulted in significant damage to the site of the Fukushima Dai-ichi nuclear power station. Although the spent fuel pools and the used fuel assemblies stored in the pools remained intact at the plant, the event led to questions about the safe storage of spent fuel and whether the NRC should require the expedited transfer of spent fuel from pools to dry cask storage containers at U.S. nuclear power plants.

This report documents the Office of Nuclear Regulatory Research's consequence study that continues our examination of the risks and consequences of postulated spent fuel pool accidents. A spent fuel pool's robust concrete structure and stainless steel liner keep more than 20 feet of water above the spent fuel stored within it ensuring ample cooling for the spent fuel and adequate radiation shielding for plant personnel. About every two years, some used fuel is removed from the reactor and placed into the spent fuel pool. The used fuel most recently removed from a reactor is radiologically and thermally "hot". The hot fuel is distributed throughout the pool and is surrounded by older, cooler used fuel. After used fuel has cooled in the spent fuel pool for more than about five years, it has radiologically decayed such that it can be moved to dry storage casks for longer term storage.

This study compared potential accident consequences from a pool nearly filled with spent fuel and a pool in which fuel that has cooled sufficiently has been removed. The staff first evaluated whether a severe, though unlikely, earthquake would damage the spent fuel pool to the point of leaking. In order to assess the consequences that might result from a spent fuel pool leak, the study assumed seismic forces greater than the maximum earthquake reasonably expected to occur at the reference plant location. The NRC expects that the ground motion used in this study is more challenging for the spent fuel pool structure than that experienced at the Fukushima Daiichi nuclear power plant from the earthquake that occurred off the coast of Japan on March 11, 2011. That earthquake did not result in any spent fuel pool leaks. In the small likelihood that such an extreme earthquake caused a leak, the staff then analyzed where the leak would be expected, the size of the leak, and how the spent fuel could overheat and potentially release radioactive material into the environment. Finally, the staff analyzed what the public health and environmental effects of a radiological release would be in the area surrounding the plant. In order to estimate the hypothetical consequences, the staff analyzed scenarios where some preplanned and improvised mitigative actions by the emergency response organization were either not successful or not implemented.

The study results for the specific reference plant and earthquake analyzed are consistent with past studies' conclusions that spent fuel pools are likely to withstand severe earthquakes without leaking. Past studies considered a wider range of earthquakes than this study. In the unlikely situation that a leak occurs, this study shows that for the scenarios and spent fuel pool studied, spent fuel is only susceptible to a radiological release within a few months after the fuel is moved from the reactor into the spent fuel pool. After that time, the spent fuel is coolable by air for at least 72 hours. This study shows the likelihood of a radiological release from the spent fuel after the analyzed severe earthquake at the reference plant to be about one time in 10 million years or lower. If a leak and radiological release were to occur, this study shows that the individual cancer fatality risk for a member of the public is several orders of magnitude lower than the Commission's Quantitative Health Objective of two in one million ($2x10^{-6}$/year). For such a radiological release, this study shows public and environmental effects are generally the same or smaller than earlier studies.

The Office of Nuclear Reactor Regulation's regulatory analysis for this study indicates that expediting movement of spent fuel from the pool does not provide a substantial safety enhancement for the reference plant. The NRC plans to use the insights from this analysis to inform a broader regulatory analysis of the spent fuel pools at all U.S. operating nuclear reactors as part of its Japan Lessons-learned Tier 3 plan. The NRC continues to believe, based on this study and previous studies that spent fuel pools continue to provide adequate protection of public health and safety.

ABSTRACT

The U.S. Nuclear Regulatory Commission performed this consequence study to continue its examination of the risks and consequences of postulated spent fuel pool accidents. The study provides publicly available consequence estimates of a hypothetical spent fuel pool accident initiated by a low likelihood seismic event at a specific reference plant. The study compares high-density and low-density loading conditions and assesses the benefits of post 9/11 mitigation measures. Past risk studies have shown that storage of spent fuel in a high-density configuration is safe and risk of a large release due to an accident is very low. This study's results are consistent with earlier research conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking. The NRC continues to believe, based on this study and previous studies that high density storage of spent fuel in pools protects public health and safety. The study's results will inform a broader regulatory analysis of the spent fuel pools at U.S. nuclear reactors as part of the Japan Lessons-learned Tier 3 plan.

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Sam Hansell	Eric Powell	Bret Tegeler	

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

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission performed this consequence study to continue its examination of the risks and consequences of postulated spent fuel pool accidents. Pertinent research conducted over the last several decades is summarized in NUREG-1353, “Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools, April 1989; in NUREG/CR-6451, “A Safety and Regulatory Assessment of Generic BWR [boiling water reactor] and PWR [pressurized water reactor] Permanently Shutdown Nuclear Power Plants,” April 1997 and in NUREG-1738, “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants,” February 2001. The purpose of this consequence study was to determine if accelerated transfer of older, colder spent fuel from the spent fuel pool at a reference plant to dry cask storage significantly reduces risks to public health and safety. The specific reference plant used for this study is a GE Type 4 BWR with a Mark I containment.

The study’s results will help inform a broader regulatory analysis of the spent fuel pools at U.S. nuclear reactors as part of the Japan Lessons-learned Tier 3 plan. This study’s results are consistent with earlier research studies’ conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking cooling water and potentially uncovering the spent fuel. The study shows the likelihood of a radiological release from the spent fuel after the analyzed severe earthquake at the reference plant to be about one time in 10 million years or lower. In addition, the regulatory analysis included with this study does not support accelerated spent fuel transfer to casks for the reference plant. .

This study presents detailed analyses using state-of-the-art, validated, deterministic methods and assumptions, as well as probabilistic insights where practical. Previous studies have shown that earthquakes present the dominant risk for spent fuel pools, so this analysis considered a severe earthquake with ground motion stronger than the maximum earthquake reasonably expected to occur for the reference plant. The NRC expects that the ground motion used in this study is more challenging for the spent fuel pool structure than that experienced at the Fukushima Daiichi nuclear power plant from the earthquake that occurred off the coast of Japan on March 11, 2011. That earthquake did not result in any spent fuel pool leaks. This beyond-design-basis earthquake severity was selected to challenge the spent fuel pool integrity. The study considered two spent fuel configurations:

- A relatively full pool where the hottest spent fuel assemblies are surrounded by four cooler fuel assemblies in a 1×4 pattern throughout the pool (referred to as the high-density loading scenario), and;

- A minimally loaded pool where all spent fuel with at least 5 years of pool cooling has been removed so the hottest fuel assemblies are surrounded by additional water (referred to as the low-density loading scenario).

Limited sensitivity analyses of a 1x8 spent fuel configuration and a uniform configuration were also performed to better understand the potential effect of spent fuel configurations on the results.

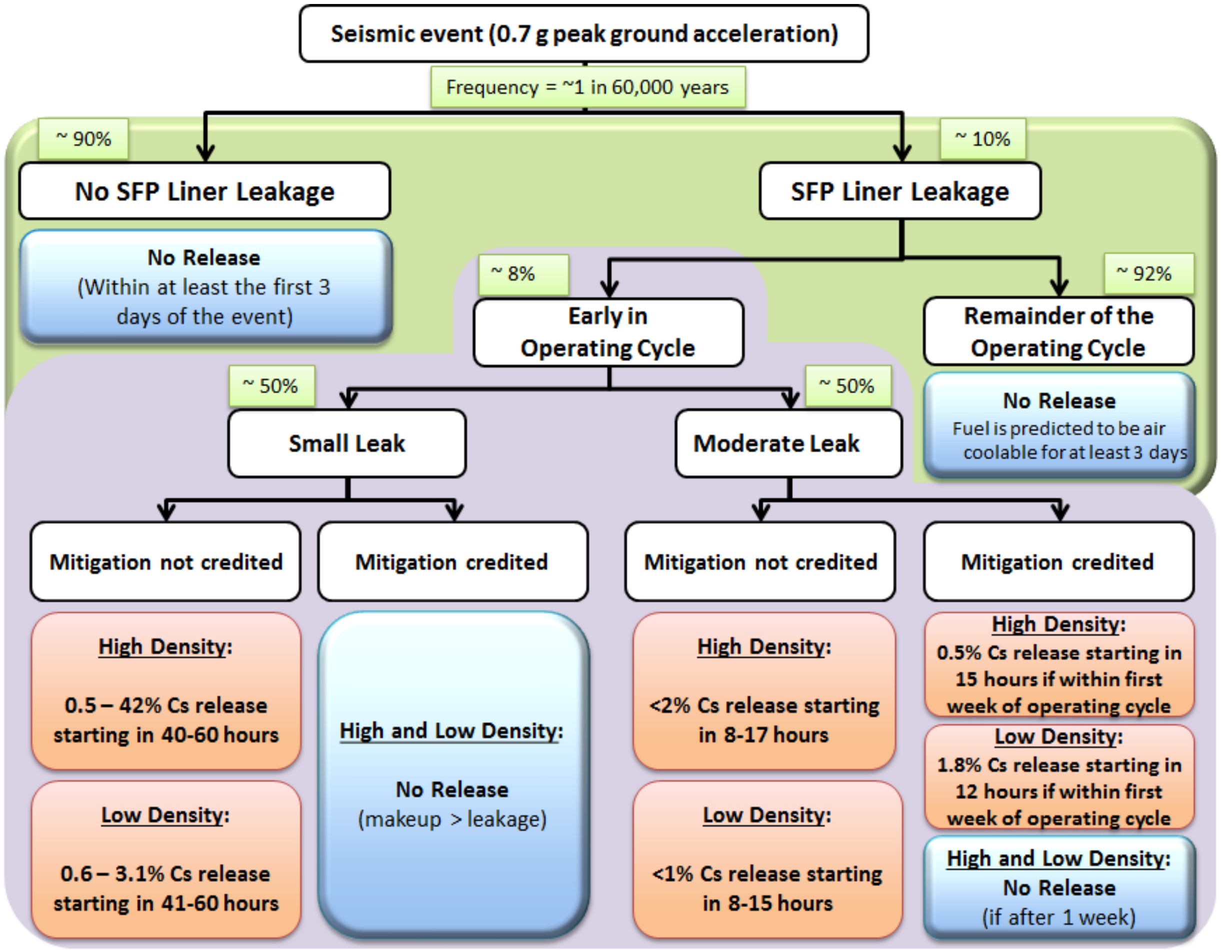
Additionally, the study evaluated the potential benefits of strategies required in Title 10, Code of Federal Regulations (10 CFR), Part 50.54 (hh)(2) following the September 11, 2001, attacks.

These “mitigation measures” are intended to maintain spent fuel pool cooling in the event of a loss of large areas of the plant due to explosions or fire.

The study evaluated 10 CFR 50.54(hh)(2) mitigation measures by analyzing each scenario twice – with and without credit for mitigation. The study shows that successful mitigation reduces the likelihood of a release. The likelihood of a spent fuel pool release was equally low for both high- and low-density fuel loading. This is because high- and low-density fuel loading contains the same amount of new, hotter spent fuel recently moved from the reactor to the spent fuel pool. In the unlikely event of an earthquake-induced spent fuel pool leak, the likelihood of fuel heatup leading to a release was more strongly affected by the fuel loading pattern rather than the total amount of fuel in the pool. In other words, the use of favorable fuel patterns such as the 1x4 pattern promotes natural circulation air coolability and reduces the likelihood of a release from a completely drained pool. Analysis also shows that for the scenarios and spent fuel pool studied, spent fuel is only susceptible to a radiological release within a few months after the fuel is moved from the reactor into the spent fuel pool. After that time, the spent fuel is coolable by air for at least 72 hours.

The study considered scenarios where some preplanned and improvised mitigative actions were either not successful or not implemented before three days, at which time the analysis was terminated. In addition to the 10 CFR 50.54(hh)(2) mitigation measures, the site emergency response organization would request support from the offsite response organizations to implement improvised additional mitigative measures, such as pumping water into the spent fuel pool using a fire truck. Analysis of these additional mitigative measures was beyond the scope of this study. Additionally, this study does not consider the post-Fukushima mitigation required by NRC in Orders EA-12-051 and EA-12-049 and currently being implemented by all operating U.S. nuclear power plants which should serve to further reduce spent fuel pool accident risk by increasing the capability of nuclear power plants to mitigate beyond-design-basis external events.

Figure ES-1 illustrates the study results in terms of the likelihood of a leak and magnitude of release from the spent fuel pool (SFP) for the severe, low likelihood earthquake considered in this study.



Note: The low-density pool has about 1/3 of Cs-137 inventory compared to high-density pool. Early in the operating cycle refers to early time after shutdown.

Figure ES-1: Likelihood of a leak and magnitude of releases from beyond design basis earthquake

This study considered a severe earthquake expected to occur once in 60,000 years; the pool is expected to remain intact during more likely, less severe earthquakes. The structural analysis of the pool shows the spent fuel pool in this study has a 90% probability of surviving the severe earthquake with no liner leakage (or conversely, a 10% probability of damaging the liner at the SFP wall/floor junction such that leakage will occur). The specific conditions for liner failure vary according to site conditions and spent fuel pool design. NUREG-1353 predicted the likelihood of liner failure from all potential earthquakes to be between about two and six times in a million years. NUREG-1738 predicted the likelihood of liner failure from all potential earthquakes to be between about two times in a million years and two times in 10 million years. This study considered an earthquake with ground motion roughly four to eight times stronger than that used in the plant design and predicted a liner failure likelihood of about two times in a million years.

The study examined how an accident is expected to proceed if the pool liner is damaged, concluding that pool leaks are somewhat less likely to release radioactive material to the environment than in previous studies. Depending on the size of the pool liner leak, releases could start anywhere from eight hours to several days after the leak starts assuming 10 CFR 50.54(hh)(2) mitigation measures are unsuccessful. In the event of an earthquake, releases are considered very unlikely for several reasons:

- The study finds liner damage is the only way to cause a radiological release in less than 3 days for the scenarios and spent fuel pool studied. Other possible outcomes provide time to prevent a release by taking emergency actions. This is consistent with earlier studies.

- The time period of susceptibility for a release of radioactive materials during the operating cycle is short. This study's detailed accident progression modeling differs from earlier work in showing that for the severe earthquake analyzed, draining the pool after liner failure is less likely to lead to a release. Because spent fuel can be effectively cooled by water, steam, or air, the likelihood of fuel overheating to the point of radiological release depends on several factors: how much residual heat the fuel generates, the fuel loading pattern, and the timing, location, and size of the liner leakage. If 10 CFR 50.54(hh)(2) mitigation measures aren't successful, releases could occur the first few months after the fuel came out of the reactor (or 8% of the reactor's two-year operating cycle). If 10 CFR 50.54(hh)(2) mitigation measures are successful, releases could only occur the first several days after the fuel came out of the reactor (a factor of twenty reduction in the likelihood of release).

In the unlikely event an earthquake induced liner failure does occur, this study predicts the largest releases would come from high-density loading cases without 10 CFR 50.54(hh)(2) mitigation measures. However, for each high-density loading release case, the corresponding low-density loading case also resulted in a release. The low-density cases generally resulted in a smaller release due to the smaller inventory of radioactive materials and the lower potential for hydrogen combustion. For the high-density cases, the releases are limited to a few percent of the cesium inventory, except for a few cases that predicted hydrogen combustion and resulted in releases of one to two orders of magnitude higher than the other cases. In these cases, the spent fuel heats up in a steam environment leading to oxidation of zirconium and releasing hydrogen gas into the reactor building. The mixing and reaction of hydrogen and oxygen leads to a hydrogen combustion and substantially damages the reactor building. That damage could breach structures that would retain radioactive material, along with allowing more oxygen into the building, potentially increasing the severity of the spent fuel fire. The study included a sensitivity analysis for a 1x8 loading pattern (hotter fuel surrounded by 8 cooler assemblies in a repeating pattern) which also resulted in smaller radioactive releases because the hotter assembly transfers its heat to the cooler assemblies resulting in lower peak fuel temperatures

Following the evaluation of successful and unsuccessful mitigation cases, a limited-scope human reliability analysis was performed to estimate the likelihood of successful operator actions implementing 10 CFR 50.54(hh)(2) mitigation measures to prevent fuel damage. Assumptions included post-earthquake on-site portable mitigation equipment required by 10 CFR 50.54(hh)(2) is available, minimum plant staffing are available for implementing spent fuel pool mitigation, and the work area is accessible to perform mitigation. The structural and accident progression analyses show that at least 99% of the time, the earthquake would not result in spent fuel overheating even without mitigative actions for the first seven days following the accident. For the remaining times, mitigative actions are needed to prevent fuel damage and the calculated mitigation success rates range from about 25% to 95% depending on plant conditions and assuming that the refueling floor is accessible. There are two exceptions where mitigation will be ineffective under the moderate leak scenarios: (1) if the earthquake occurs at the beginning of a refueling outage when the spent fuel is too hot for the assumed mitigation, and (2) if the earthquake occurs when spent fuel is relatively hot and the reactor and spent fuel pool are hydraulically disconnected resulting in insufficient time to deploy mitigation and natural cooling mechanisms cannot prevent fuel damage.

The study's analyses shows that a release from a spent fuel pool accident after the severe earthquake at the reference plant could occur about one time in 10 million years or lower. The factors leading to this low likelihood, as discussed above, are summarized in Figure ES-2.

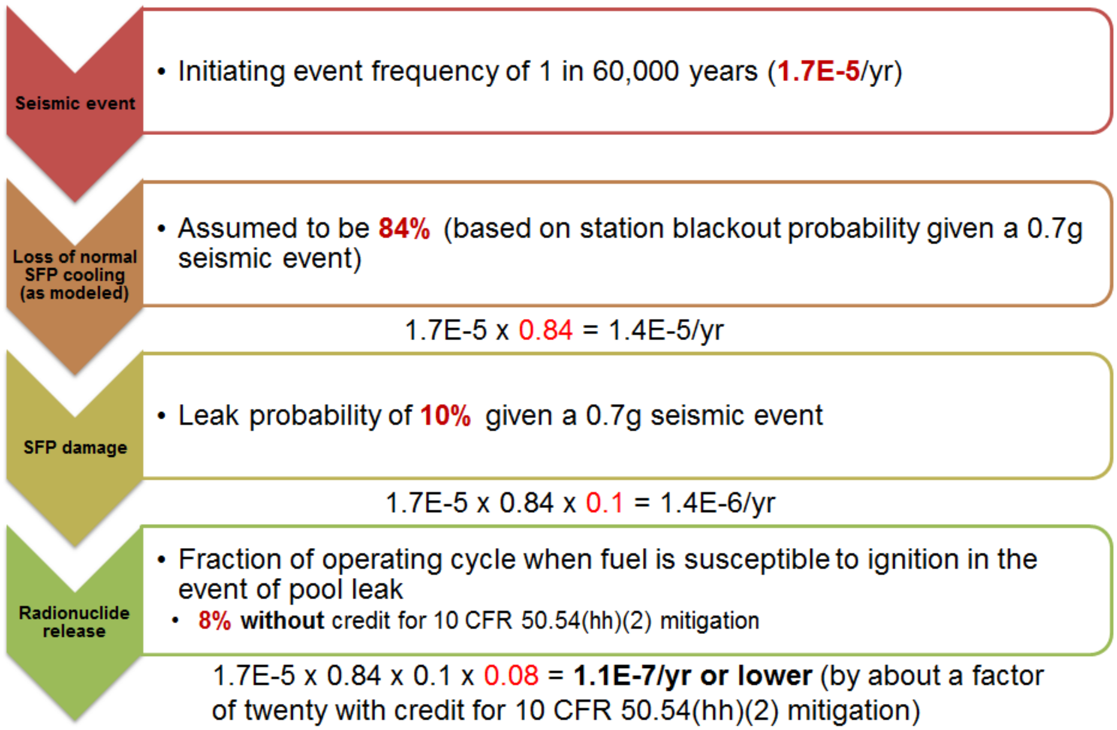


Figure ES-2: Factors Affecting Likelihood of SFP Release from a Severe Seismic Event

The study then estimated consequences to the public of a low likelihood spent fuel pool accident release. The releases of radioactive material are generally comparable to past studies. Despite the fairly large releases for certain predicted accident progressions, consequence analysis of all scenarios indicated zero early fatalities from acute radiation effects because protective actions were modeled to be effective in limiting doses to the public. The study also showed that the risk of an individual dying from cancer from the radioactive release is very low. When including the very low likelihood of a release, the risk in the analyzed scenarios that an average individual within 10 miles receives a fatal latent cancer is between about two in a trillion and five in a hundred billion per year. The risks are similar between different loading or mitigation scenarios because of modeled offsite protective actions that include evacuation, sheltering, relocation, and decontamination. Additionally, these individual risks are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough for the areas to be considered habitable.

In order to do a regulatory analysis to inform whether low density loading should be required at the reference plant, cost estimates of potential protective measures are considered along with other parameters in a cost-benefit analysis. The study shows that, while public health effects from these low likelihood spent fuel pool releases are expected to be very low for all the scenarios studied, offsite protective measures in the form of population relocation and land interdiction may be extensive. High-density loading releases without 10 CFR 50.54(hh)(2) mitigation measures are calculated to result in release frequency-weighted land interdiction values of 0.001 mi^2 per year and 0.5 displaced individuals per year which are arrived at by multiplying the estimated frequency and the estimated consequence. While the amount of land interdiction can be large, the fraction expected to be permanently interdicted is small if a release were to occur. For low-density loading or with successful deployment of 10 CFR 50.54(hh)(2) mitigation measures, considerably less land interdiction and displaced individuals are predicted.

Comparisons of the calculated individual latent cancer fatality (LCF) risk within 10 miles to the NRC Safety Goal are provided in Figure ES-3 to give context that may help the reader to understand the contribution to cancer risks from the accident scenarios that were studied. The NRC Safety Goal for latent cancer fatality risk from nuclear power plant operation (i.e., 2×10^{-6} or two in one million per year) is set 1,000 times lower than the sum of cancer fatality risks resulting from all other causes (i.e., $\sim 2 \times 10^{-3}$ or two in one thousand per year).

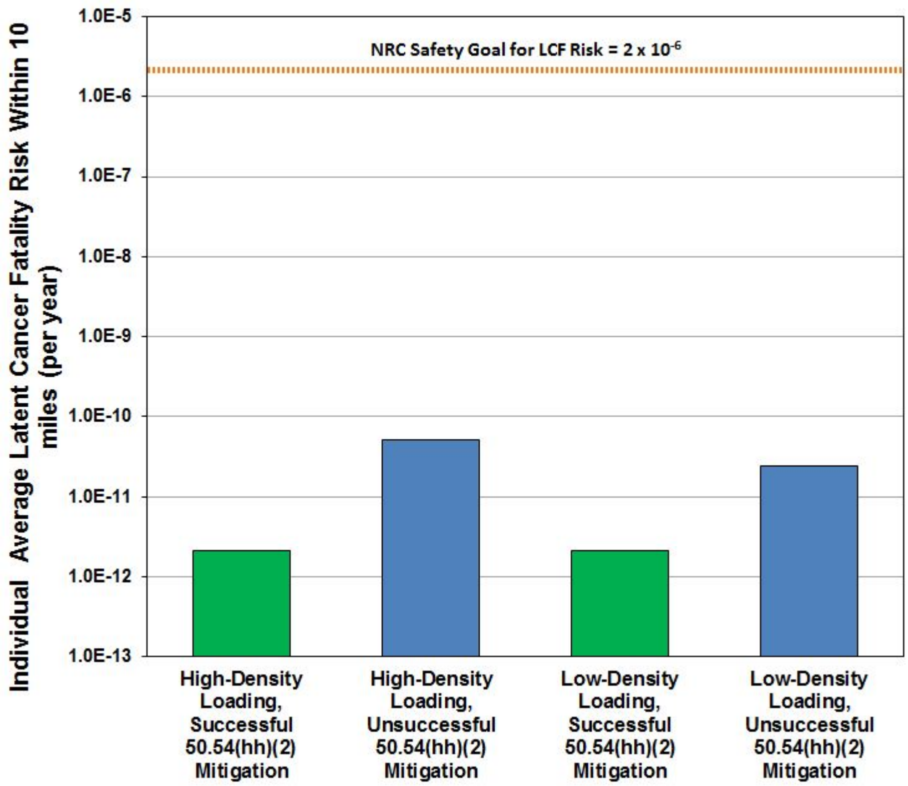


Figure ES-3: Comparison of Population-Weighted Average Individual Latent Cancer Fatality Risk Results for this Study to the NRC Safety Goal (plotted on logarithmic scale)

Comparing the study results to the NRC Safety Goal does involve important limitations. First, the safety goal is intended to encompass all accident scenarios on a nuclear power plant site, including both reactors and spent fuel. This study does not examine all scenarios that would need to be considered in a probabilistic risk assessment for a spent fuel pool, although seismic contributors are considered the most important contributors to spent fuel pool risk. Also, this study represents a mix of limited probabilistic considerations with a deterministic treatment of mitigating features. All analytical techniques, both deterministic and probabilistic, have inherent limitations of scope and method and also have uncertainty of varying degrees and types. As a result, comparison of the scenario-specific calculated individual LCF risk to the NRC Safety

Goal is incomplete. However, it is intended to show how multiple spent fuel pool scenarios' risk results in the one in a trillion (10^{-12}) to one in 10 billion (10^{-10}) per year LCF range) are low. While the results of this study are scenario-specific and related to a single spent fuel pool, staff concludes that since these risks are several orders of magnitude smaller than the $2x10^{-6}$ (two in one million) individual LCF risk that corresponds to the safety goal for latent cancer fatalities, it is unlikely that the results here would contribute significantly to a risk that would challenge the Commission's safety goal policy (NRC, 1986).

In conclusion, past SFP risk studies have shown that high-density spent fuel storage is safe and risk of a release due to an accident is low. This study is consistent with earlier research conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking. This study estimated that the likelihood of a radiological release from the spent fuel pool resulting from the selected severe seismic event analyzed in this study is on the order of one time in 10 million years or lower. For the hypothetical releases studied, no early fatalities attributable to radiation exposure were predicted and individual latent cancer fatality risks are projected to be low, but extensive protective actions may be needed.

The study results demonstrated that in a high-density loading configuration, dispersing hotter fuel throughout the pool or successful mitigation generally prevented or reduced the size of potential releases. Low-density loading reduced the size of potential releases but did not affect the likelihood of a release. When a release is predicted to occur, early and latent fatality risks for individual members of the public do not vary significantly between the scenarios studied because protective actions, including relocation of the public and land interdiction, were modeled to be effective in limiting exposure. The beneficial effects in the reduction of offsite consequences between a high-density loading scenario and a low-density loading scenario are primarily associated with the reduction in the potential extent of land contamination and associated protective actions. The regulatory analysis for this study indicates that expediting movement of spent fuel from the pool does not provide a substantial safety enhancement for the reference plant. The NRC plans to use the insights from this analysis to help inform a broader regulatory analysis of the spent fuel pools at U.S. nuclear reactors as part of its Japan Lessons-learned Tier 3 plan. The NRC continues to believe, based on this study and previous studies that high density storage of spent fuel in pools protects public health and safety.

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ABBREVIATIONS AND ACRONYMS

ac	alternating current
ACRS	Advisory Committee on Reactor Safeguards
AEF	annual exceedance frequency
ANL	Argonne National Laboratory
BEIR	biological effects of ionizing radiation
BEF	biological effectiveness factor
Bq	Becquerel
BWR	boiling-water reactor
C	Celsius
CEC	Commission of the European Communities
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
Ci	curies
Cs	cesium
CSCM	continuous surface cap model
CV	control volume
DBE	design basis earthquake
dc	direct current
DDREF	dose and dose rate effectiveness factor
DF	decontamination factor
DLTEVA	delay to evacuation
DLTSHL	delay to shelter
DOE	U.S. Department of Energy
DURBEG	duration of beginning phase
DURMID	duration of middle phase
E	East
EAL	emergency action levels
EAS	emergency alert system
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ESPEED	speed (WinMACCS input variable)
ETE	evacuation time estimate
FAQ	frequently asked questions
FEMA	Federal Emergency Management Agency
FGR	federal guidance report
FSAR	final safety analysis report
GE	General Electric
GEIS	generic environmental impact statement
GI	generic issue
GNF	Global Nuclear Fuel
gpm	gallons per minute
GSI	Generic Safety Issue
GWD	gigawatt-day
HCLPF	high confidence of low probability of failure
HEP	human error probability
hr	hour
HPS	Health Physics Society
HRA	human reliability analysis

I	iodine
ICE	inadvertent criticality event
ICRP	International Commission on Radiological Protection
INL	Idaho National Laboratory
IPEEE	individual plant evaluation for external events
ISFSI	independent spent fuel storage installation
ISRS	in-structure response spectra
K	Kelvin
KI	potassium iodide
LCF	latent cancer fatality
LLNL	Lawrence Livermore National Laboratories
LNT	linear no-threshold
LOOP	loss of offsite power
MACCS2	MELCOR Accident Consequence Code System, Version 2
MCCI	molten core-concrete interaction
MCi	megacuries
MPC	multi-purpose container
MTU	metric tons of uranium
MW	megawatts
MWD	megawatt days
N	North
NCRP	National Council on Radiation Protection and Measurements
NAS	National Academy of Sciences
NRC	Nuclear Regulatory Commission
OCP	operating cycle phase
ORNL	Oak Ridge National Laboratory
ORO	offsite response organization
OSC	operational support center
PAG	protective action guides
PBAPS	Peach Bottom Atomic Power Station
PGA	peak ground acceleration
PPG	pool performance guidelines
PWR	pressurized water reactor
PRA	probabilistic risk assessment
QHO	quantitative health objectives
RB	reactor building
REM	Roentgen Equivalent Man
RHR	residual heat removal
S	South
SAE	site area emergency
SBO	station blackout
SIP	shelter in place
SOARCA	State of the Art Reactor Consequence Analyses
SNL	Sandia National Laboratories
SFP	spent fuel pool
SFPS	Spent Fuel Pool Study
SSC	structures, systems, and components
SSE	safe shutdown earthquake
TSC	technical support center
TSG	technical support guideline
TR	technical report (EPRI technical reports)

USGS	United States Geological Survey
W	West

1. INTRODUCTION AND BACKGROUND

All operating commercial nuclear reactors in the United States are of the light-water reactor design. They utilize upright fuel assemblies (roughly 12 feet in length) with low-enriched uranium oxide fuel (less than 5-percent uranium-235). The fuel assemblies which are composed of numerous fuel rods (typically 80-100 rods for boiling-water reactor fuel and 200–300 rods for pressurized-water reactor fuel) are placed in the reactor for two to three operating cycles. Each operating cycle typically lasts 18 to 24 months. At the end of their “life,” the assemblies are placed in large pools of water near the reactor that are roughly 12 meters (m) (40 feet (ft)) deep. For facilities licensed to operate an independent spent fuel storage installation (ISFSI), the fuel assemblies are later loaded into casks and moved to the ISFSI as necessary to accommodate future core offloads. 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 (SFP)) and “dry storage” (to describe storage in casks).

SFPs in the United States were originally designed to store one to two reactor cores worth of spent fuel, so that the fuel could “cool down” (become less thermally and radioactively “hot”) before its movement to a reprocessing facility or permanent geological repository. Owing to the abandonment of spent fuel reprocessing as well as delays in the identification, licensing and construction of a repository, U.S. nuclear power plants “re-racked” their SFPs in the 1980s and 1990s to allow for the storage of larger numbers of spent nuclear fuel assemblies (i.e., roughly four reactor cores worth for the plant studied in this study). Throughout this time (including present day), 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. The basis for this position is discussed later in this section.

Stakeholders have periodically challenged the NRC’s position that SFPs provide adequate protection of public health and safety. To understand the basis for these challenges, it’s first necessary to understand two basic facts about spent nuclear fuel:

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

The list below presents some less-obvious considerations from the perspective of the benefits and disadvantages associated with transitioning from high-density storage to low-density storage. The list is subdivided into two parts—those considerations that are covered within this study and those that are not.

This study includes the following considerations:

- Removal of older fuel from the SFP will decrease the inventory of longer lived radionuclides, such as cesium-137, present in the SFP.

- Removal of older fuel will result in less radioactive material would be present in the pool if a radioactive release occurred, which would be expected to reduce potential offsite consequences.

- Removal of older fuel reduces the overall heat load in the pool while decreasing the amount of metal mass to act as a heat sink should the fuel become uncovered, which can have competing effects on accident timing depending on the type of accident (e.g., a boiloff event versus a complete draindown).

- 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-percent of the total pool volume—because most of the pool is occupied by water, not fuel).[1]

This study does not explicitly address the following considerations, though some are discussed further in APPENDIX B:

- Discharging large amounts of fuel (and thus greatly increasing the amount of fuel contained in the ISFSI) would increase the number of casks required to store the existing spent fuel inventory.

- Expedited discharging of fuel from the SFP to dry storage increases the frequency of postulated cask drops, which in turn increases the frequency of causing damage to the pool or cask 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 into casks that are not currently approved for shipping or long-term storage may require that fuel to be repackaged 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 from various stakeholders. Issues related to the existing dry cask storage infrastructure, worker dose, and economics are discussed in (NAC, 2011) and (EPRI, 2012). Section 1.6 of this report provides more information on each of these studies.

The first set of bulleted considerations is generally advantages associated with expedited fuel movement to casks, while the latter set of bulleted considerations is generally disadvantages. 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. By investigating the pros, we are informing ongoing discussions as to whether fuel movement from spent fuel pools to dry cask storage should be expedited and if any of the "pros" are more

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[1] The additional water can have a non-intuitive negative impact in certain situations. For a leak at the bottom of the SFP, the additional water at the elevation of the fuel causes it to take longer to "clear" the baseplate (i.e., for the level of the receding water to drop below the bottom of the baseplate). In situations where natural circulation of air under and up through the racks is effective for preventing fuel heatup, this actually temporarily inhibits cooling of the fuel. While this does require a specific set of conditions to be relevant, it is raised here because it does actually arise in one of the scenarios realized later in this report.

compelling than past studies suggest. If they are, then the issue can be addressed more holistically.

1.1 Project Impetus

Various risk studies (most recently NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," issued February 2001) have shown that storage of spent fuel in a high-density configuration in SFPs is safe and that the risk is 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 security assessments after the events of September 11, 2001, 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, the NRC issued a new regulation, 10 CFR 50.54(hh)(2), that requires reactor licensees to develop and implement guidance and strategies intended, in part, to maintain or restore SFP cooling capabilities following certain beyond design basis events.

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, Revision 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants - Draft Report for Comment," issued July 2009. 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 the 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.

The spent fuel pool study's (SFPS) primary objective was to determine if accelerated transfer of spent fuel from the spent fuel pool to dry cask storage provides a substantial safety enhancement for the reference plant. The insights from this analysis will inform a broader regulatory analysis of the SFPs at U.S. nuclear reactors as part of the Japan Lessons-learned Tier 3 plan. NRC analyzes low likelihood (beyond the design basis) events to estimate risk to the public and to explore and ensure safety margins. The results of the study will be used to inform the evaluation of what future regulatory actions the NRC might undertake, including whether expedited transfer of spent fuel from spent fuel pools into dry cask storage is justified. To help inform whether regulatory action needs to be taken in this area, the NRC has prepared an example of a regulatory analysis of the reference plant studied in this report (see APPENDIX D). A regulatory analysis is an analytical tool used by NRC decision-makers to help determine whether the NRC should implement a proposed regulatory action. The regulatory analysis is intended to inform NRC decision makers whether there is a substantial increase in the overall protection of the public health and safety, and whether the direct and indirect costs of implementation are justified in view of a potential substantial increase in protection. For the example regulatory analysis, the Spent Fuel Pool Study (SFPS) results are used as quantitative inputs to the safety goal screening criteria in accordance with the NRC regulatory analysis guidelines (NUREG/BR-0058), wherein the quantitative health objectives are used as a surrogate of the safety goal. The example regulatory analysis also contains estimates of benefits and costs, which are quantified when possible, together with a conclusion as to whether the proposed regulatory action is cost-beneficial. "Cost-beneficial" means that the benefits of the proposed action are equal to, or exceed, the costs of the proposed action. Accident consequences such as land interdiction and population relocation reported in this study are used to estimate the costs resulting from an accident (e.g., costs of interdiction measures, such as decontamination, cleanup, and evacuation) as part of the cost-benefit analysis.

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.

1.2 Technical Approach

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

(1) A condition representative of the following: (i) high-density loading in the SFP using a 1x4 pattern (see Figure 34 for an illustration of what is meant by this terminology), (ii) a relatively full SFP, and (iii) current regulatory requirements relating to fuel configuration and preventive/mitigative capabilities; and

(2) A condition where fuel with more than 5 years of cooling has been moved to dry cask storage (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 (e.g., seismic response, decay heat, radionuclide inventory) is based on readily available information that primarily stemmed from sources such as the study reported in NUREG-1150, "Severe Accident Risks: An Assessment of Five Nuclear Power Plants," issued December 1990; seismic information developed by the U.S. Geological Survey (USGS); the post-9/11 security assessments[2]; and the State-of-the-Art Reactor Consequence Assessment (SOARCA) described in NUREG-1935. Later in the project, the licensee provided additional information that generally corroborates the assumptions made in this study.

A BWR plant was chosen for this analysis for a mix of reasons including availability of computer models for a BWR plant, a perception of greater external stakeholder interest in elevated (relative to grade) SFPs [3], and the fact that the nuclear reactors that felt the higher tsunami waves and stronger ground motions from the March 11, 2011, Tohoku earthquake, which includes those at Fukushima Daiichi, were all BWR reactors. In the context of a seismic event, 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 BWR design is not intended to suggest that these designs are more vulnerable to

2 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.

3 SFPs at pressurized water reactor and BWR/6s (which have Mark III containments) are generally at or near grade elevation, with many being partially below grade. In the Mark I and Mark II designs, the SFP is oriented such that the top of the SFP is at the same elevation as the top of the primary containment vessel, which results in them being well above grade.

SFP accidents. In reality, there are differences between the major design types (PWRs versus BWRs) that make each more or less susceptible to SFP 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 Specificity and Familiarization

This study is intentionally based on plant-specific information for a particular site, as opposed to attempting to define a generic site that might bound a set of plants. This approach was taken because it provides the best context for examining SFP accident progression and release phenomenology in a realistic fashion, for the purpose of providing a better understanding of the factors that affect the characterization of SFP beyond-design basis accidents. The decision to proceed in this manner was deliberately made in reaction to persistent criticisms regarding the realism of past studies (due to their goal of broad applicability in order to support their intended purposes). Because this study strives to be site-specific, it does not account for the variability in design and operation across the operating fleet, but rather, represents one point within that spectrum.

In almost all situations where plant-specific design and operational information is used, it is based on Unit #3 of the Peach Bottom Atomic Power Station (PBAPS), circa 2011. Nevertheless, the SFPS makes some assumptions that are not representative of PBAPS because either (a) insufficient information was available at the time the modeling decision was made or (b) the PBAPS situation was viewed to be atypical. Regarding the former exception, the initial phase of the work was expedited to achieve early insights. Some modeling assumptions were confirmed in parallel to ongoing work, and in instances where newer information provided additional perspective on the modeling assumption, this is noted in the report. Regarding the second exception, the major example of this is that the study assumed the fuel is configured in a 1x4 pattern rather than in the 1x8 pattern used at PBAPS as discussed further in Section 5.1. In some situations, the 1x8 pattern is predicted to have a beneficial effect on the amount of radiation released (Section 9.2). Additionally, sensitivity analyses presented in Chapter 9 explore the effect of some important parameters on the study results. Due to these exceptions, the analysis contained in this report is best described as being performed for a “reference plant” which is largely based on PBAPS.

PBAPS has two General Electric (GE) Type 4 BWRs with Mark I containments, Units #2 and #3. This study uses Unit #3 when unit-specific information is required. Unit #1 is no longer in operation. Units #2 and #3 each have a dedicated SFP, 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, SFP, 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 currently under review (as of January 2013).

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 needed. The site also has an 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) that has State-specific protective action guidelines. Detailed site-specific information relevant for this study is covered in the remainder of this report, including figures that show the reactor building layout, SFP layout, etc.

1.4 Basic Scenario Development

The following key aspects of the way this study is conducted should be mentioned at this point.

- 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 (a low-density loading configuration in the pool) are analyzed. A situation in which 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 (it should be mentioned that BWR fuel is channeled, which reduces the benefit of cross-flow if the pool were to become drained).

- The study focuses on the SFP, not the reactor, though for instances in which the two are hydraulically connected, both are considered to a certain extent.

- In estimating the likelihood and consequences of radiological release, the study does not attempt to quantify the likelihood of successful deployment of mitigation, but rather treats every scenario considering both the case with successful mitigation deployment and the case with unsuccessful mitigation deployment (also referred to as mitigated and unmitigated later in the report)[4]. These results are then used to drive a human reliability analysis (Chapter 8) which provides information about what plant conditions impact mitigative reliability, and what range of likelihoods are expected.

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

1.5 Rationale for Focusing on Consequences of a Seismic Hazard

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

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, by means of the limited treatment of relative likelihood. While these elements provide some of the benefits of

[4] Note that the shorthand of "mitigated" and "unmitigated" still refers to whether mitigative actions are successfully deployed, not whether the accident itself leads to a release.

an actual risk assessment, there are several elements of a risk assessment that are specifically not performed. These include the following:

- failure modes and effects analysis (except for SSCs specifically discussed in this Chapter)
- data analysis and component reliability (e.g., consideration of random failures)
- effects of dependencies
- HRA as part of the accident progression and recovery; a limited scope HRA is performed in (Chapter 8
- 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 the following:

- 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 to put the results in context.

The inclusion of probabilistic aspects within the current study allows the study to consider some aspects of likelihood, but will not support definitive statements on risk. To elaborate, this study focuses on a specific portion of the overall risk profile, that of large seismic events between 0.5 and 1g. In comparing the results of this study to those of previous studies, one can corroborate or challenge the continued applicability of prior estimates for this piece of the risk profile. Since large seismic events have been shown in the past to be a prominent contributor to risk, this comparison helps to predict whether a comprehensive risk assessment would be expected to result in an overall decrease or increase in the estimated risk. Using this approach, the results of this study can draw supportable, but not definitive, conclusions about overall consequences and risk.

For the present study, because of (1) the relative simplicity of the SFP and its supporting infrastructure as compared to a reactor and its supporting infrastructure and (2) the much lower assembly decay heats, the majority of potential SFP accident risk is believed to emanate from either of the following two events:

(1) events that have the potential to cause a sizable leak in the SFP

(2) events that might preclude operator action to cool or inject water into the pool for an extended period of time (i.e., days)

When one considers the various possible initiators, the first criterion points to the following:

(1) very large (i.e., well beyond the design-basis) seismic events (Note that these events almost certainly initiate a loss-of-offsite power and may fail emergency on-site power.)
(2) heavy load (e.g., cask) drops
(3) inadvertent aircraft crashes

In addition to these, the second criterion also points to the following:

(4) extended loss-of-offsite power (LOOP) events caused by severe weather (e.g., severe storms, hurricanes, tornados), within design-basis seismic events or other grid upsets, with concurrent loss of emergency onsite alternating current (ac) power (either because of the same event or because of coincidental hardware failures)

(5) lack of accessibility caused by a reactor accident that has released radioactive material outside of primary containment (or an accident involving the other SFP)

Note that sabotage events have been excluded from the scope of this study.

Items #(1) (seismically-induced station blackout), #(2) (cask drops), and #(4) (extended 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. The frequency and consequences of a reactor accident is not considered and the effect of a reactor accident on a spent fuel pool scenario is partially considered here, but not rigorously (see Section 2.2 of this study for more information).

Past studies have had generally similar conclusions about the relative contribution to risk from the various initiating events considered. Table 1 summarizes fuel uncovery frequencies from NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools,"" issued April 1989, and NUREG-1738. 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	7×10^{-6}	2×10^{-6} (LLNL) 2×10^{-7} (EPRI)[2]
Cask / heavy load drop	3×10^{-8}	2×10^{-7}
LOOP – severe weather	-	1×10^{-7}
LOOP – other	-	3×10^{-8}
Internal fire	-	2×10^{-8}
Loss of pool cooling	6×10^{-8}	1×10^{-8}
Loss of coolant inventory	1×10^{-8}	3×10^{-9}
Inadvertent aircraft impacts	6×10^{-9}	3×10^{-9}
Missiles – general	1×10^{-8}	-
Missiles - tornado	-	$< 1 \times 10^{-9}$
Pneumatic seal failures	3×10^{-8}	-

[1] These numbers have not been multiplied by the stated conditional probability of having a Zirconium fire of 0.25.
[2] NUREG-1738 presented results for the two different seismic hazard models in wide use at the time (the Electric Power Research Institute (EPRI) and Lawrence Livermore National Labs (LLNL) models).

For these reasons, 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 Chapter 3). This range of ground motions represents a good compromise between more likely events that would not be expected to lead to any consequences and less likely events that would lead to greater consequences (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 configuration from an isolated pool to a pool that is hydraulically connected to the reactor vessel (and back again)—these configurations will be referred to as pool-reactor configurations to distinguish from the different spent fuel loading configurations;
- may have spent fuel temporarily offloaded from the reactor;
- will have spent fuel permanently offloaded 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 neutron absorber monitoring)
- may have older spent fuel offloaded into storage casks and transferred to an ISFSI;
- will experience changes in the peak assembly fission product decay power (of interest for draindown events and spray mitigation) because of the above considerations 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 considerations as well as radioactive decay.

To rigorously represent these changing conditions, the study breaks 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 approach allows for more accurate representation of the annualized frequencies of offsite consequences. The specific selection of these phases is described further in Section 5.2 of this report.

1.7 Overview of Past Studies

A number of past studies have been performed to look at various aspects of spent fuel and SFP safety, security, and risk. The major regulatory activities are shown in Figure 1. 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.

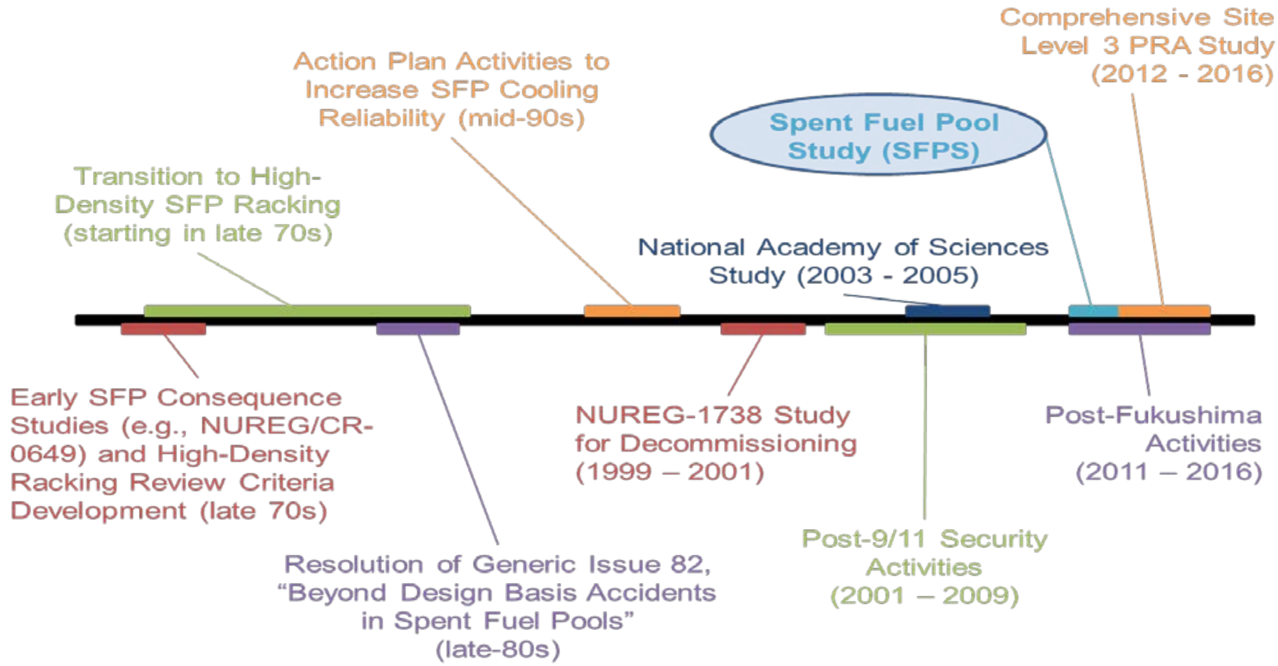


Figure 1 Graphical overview of significant SFP-related activities

In March 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 an SFP and to predict the conditions under which clad failure would occur. The report concluded that the likelihood of clad failure caused by rupture or melting following a complete drainage is extremely dependent on the storage configuration and the spent fuel decay period. Furthermore, 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 (GI)-82, "Beyond Design Basis Accidents in Spent Fuel Pools," culminated in the publishing of several related reports: NUREG/CR-4982, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," issued July 1987 (NRC, 1987), NUREG/CR-5281, "Value/Impact Analysis of Accident Preventive and Mitigative Options for Spent Fuel Pools," issued March 1989 (NRC, 1989a), and NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools,'" (NRC, 1989b). In particular, NUREG/CR-5281 investigated options including limited low-density re-racking of spent fuel, installation of water sprays above the SFP, and installation of redundant cooling, makeup systems, or both. The results of these studies indicated that the measures were, in general, not likely to be cost effective because of the low likelihood of an SFP 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 ensure the safe handling of heavy loads in the vicinity of SFPs, thus reducing the likelihood of the structural failure of the pool and rapid loss of water inventory resulting from a cask drop event.

The latter report (NUREG-1353), which draws from the preceding reports, concludes that if the decay heat level is high enough to heat the fuel rod cladding to about 900 degrees Celsius (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 based on differences in assumed rack geometry. 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 SFP risk is derived from beyond-design-basis earthquakes, this risk is no greater than the risk from core damage accidents caused by seismic events beyond the safe-shutdown earthquake (SSE). Therefore, reducing SFP risk resulting from events beyond the SSE would still leave at least a comparable risk from 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 and issued an Idaho National Laboratories (INL) study entitled, "Loss of Spent Fuel Pool Cooling PRA: Model and Results," (INL, 1996). This study considered a dual-unit plant and the following initiators:

- loss of SFP cooling
- LOOP
- loss of SFP water inventory (did not include heavy load drops)
- loss of primary (reactor) coolant
- seismic events

The results of this study indicated that, for the plant studied, the annual probability of SFP boiling is 5×10^{-5} and the annual probability of internal plant flooding associated with SFP accidents is $1x10^{-3}$. Qualitative arguments are provided to show that the likelihood of core damage from SFP boiling accidents is low for most U.S. commercial nuclear power plants. The INL study also showed that, depending on the design characteristics of a given plant, the likelihood of either (1) core damage from SFP-associated flooding or (2) spent fuel damage from pool dryout may not be negligible. Section 6.3.4 further discusses this issue.

The next year, three additional reports were issued: (1) NUREG-1275, Volume 12, "Operating Experience Feedback Report: Assessment of Spent Fuel Cooling," (NRC, 1997a), (2) "Follow-up Activities on the Spent Fuel Pool Action Plan," (NRC, 1997b), and (3) NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants," (NRC, 1997c). The first of these reports 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 one event per 100 reactor years. Loss of SFP cooling with a temperature increase greater than 20 degrees Fahrenheit (F) occurred at a rate of approximately three 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 offloads 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 SFP cooling event early in the outage.

In the second of these reports (known as 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 SFP design issues were sufficiently low that further analyses were not warranted. In one instance in which 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.

The third report (NUREG/CR-6451) presents a regulatory assessment for generic BWR and PWR plants that have permanently ceased operation. In addition to an assessment of regulatory requirements in the context of decommissioning, this study looked at the potential offsite consequences for four phases of decommissioning (hot fuel in the SFP, cold fuel in the SFP, all fuel in dry cask storage, and no spent fuel onsite). The following conclusions are based on an assumption that for the second configuration (cold fuel in the SFP), a zirconium fire would not occur, and that consequences are driven by an accident where a single fuel assembly is dropped during movement within the SFP (akin to the design-basis fuel handling accident). The report concluded that, "Since the estimated consequences of the Configuration 1 representative accident sequence approximate those of a core damage accident, it is recommended that all offsite and onsite emergency planning requirements remain in place during this period, with the exception of the Emergency Response Data System requirements of Part 50, Appendix E. Subject to plant specific confirmation, the offsite emergency preparedness (EP) requirements are expected to be eliminated for Configuration 2, on the basis of a generic boundary dose calculation. Part 50 offsite EP requirements can also be eliminated for Configurations 3 and 4 because the spent fuel has been transferred to an ISFSI (subject to Part 72 requirements) or transported offsite."

Several years later, the NRC re-visited these issues by conducting an SFP 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 (NRC, 2001). The results of the study indicated that the risk at SFPs 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 decision-making on SFPs. However, there are some important conservatisms associated with this study that need to be considered if it is applied outside of its intended context (e.g., exemption requests from NRC requirements for offsite emergency preparedness for decommissioning reactors). These conservatisms include: (1) the use of assumed and often bounding configurations, (2) simplified treatment of the thermal-hydraulic response, (3) simplifying assumptions regarding the pool failure leakage rate for large seismic event and cask drop (i.e., instantaneous draindown), and (4) emergency preparedness response representative of a decommissioned site.

On the heels of the aforementioned study, the agency also released NUREG/CR-6441 in March 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, 2002a). 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 airflow, and the building ventilation rate. Note that the SFP 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 (owing to its mechanistic treatment of severe accident phenomena), with supporting analysis using the COBRA-SFS, FLOW3D and Fluent codes, along with confirmatory experiments at Sandia National Laboratories (SNL).

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," dated April 21, 2003 (hereafter referred to as the 2003 Alvarez paper) (Alvarez et al., 2003). In response, the NRC issued a review of the paper (also in 2003) which concluded that the assessment performed of possible SFP 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 relied 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 5 years old be placed in dry casks through an expedited 10-year program costing many billions of dollars, was not justified.

Continued discussions on the issue of SFP safety and security led to a 2004-2005 National Academies study, documented in "Safety and Security of Commercial Spent Nuclear Fuel Storage," issued in 2006 (NAS, 2006). This study was Congressionally mandated (e.g., see [Congress, 2005]). The National Academies committee was briefed on numerous occasions by the 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., site-specific assessments of licensee response to develop strategies to maintain or restore SFP cooling capabilities).

The NRC's initial response to the study was documented in a letter from the NRC Chairman (Nils Diaz) to Senator Peter Domenici, dated March 14, 2005 (NRC, 2005a). In that response, NRC expressed its appreciation for the insights of the National Academies committee, noting that many of the conclusions mirrored the NRC's conclusions from prior work, which guided NRC initiatives. However, the NRC disagreed with some of the conclusions from the National Academies study, including the finding that the NRC might determine that the earlier movement of spent fuel from pools to dry cask storage would be prudent, depending on the outcome of plant-specific vulnerability analysis. "The Commission views the results of security assessments completed to date as clearly showing that storage of spent fuel in both SFP and in dry storage casks provides reasonable assurance that public health and safety, the environment, and the common defense and security will be adequately protected. The NRC will continue to evaluate the results of the ongoing plant-specific assessments and, based upon new information, would evaluate whether any change to its spent fuel storage policy is warranted." The NRC's position on each finding or recommendation that it disagreed with is contained in the report to Congress that accompanied the March 2005 letter.

In parallel to the National Academies study, the NRC continued performing the aforementioned security assessments, which were completed in 2006-2008. While the results of these studies are not publicly available because of their nature (i.e., containing sensitive information that could be useful to an adversary), the conclusions of the studies were integrated into the NRC's regulatory licensing and oversight processes (e.g., 10 CFR 50.54(hh)(2) as a result of the Power Reactor Security Rulemaking). 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 March 2007 (NRC, 2007). The report's analysis indicates 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 multipurpose 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 1.8×10^{-12} during the first year of service, and 3.2×10^{-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 leak sizes.

Two other documents of regulatory interest were issued in 2008 and 2009. The first was the denial of two PRMs, as documented in SECY-08-0036, "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)," dated March 7, 2008, 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 SFPs.

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 (NRC, 2009). This document reevaluated SFP environmental considerations related to SFPs 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 SFPs (as quantified in NUREG-1738) can be comparable to those from reactor accidents at full power (as estimated in NUREG-1150 (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 SFP accidents; and 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 Daiichi Accident" (NRC, 2011a). This report makes two sets of conclusions and recommendations related to spent fuel pool safety. The first occurs in the section of the report on prolonged loss of ac power. In this section, the task force stated the following:

> The Commission's [station blackout] SBO requirements provide assurance that each nuclear power plant can maintain adequate core cooling and maintain containment integrity for its approved coping period (typically 4 or 8 hours) following an SBO. Also, if available, the equipment used for compliance with 10 CFR 50.54(hh)(2) would provide additional ability to cool either the core or the spent fuel pool and mitigate releases from primary and secondary containment during a prolonged SBO. The implementing guidance for SBO focuses on high winds and heavy snowfalls in assessing potential external causes of loss of offsite power, but does not consider the likelihood of loss of offsite power from other causes such as earthquakes and flooding. Also, the SBO rule does not require the ability to maintain reactor coolant system integrity (i.e., PWR reactor coolant pump seal integrity) or to cool spent fuel….
>
> The Task Force concludes that revising 10 CFR 50.63 to expand the coping capability to include cooling the spent fuel, preventing a loss-of-coolant accident, and preventing containment failure would be a significant benefit.

The task force went on to recommend orders requiring reasonable protection of the equipment provided pursuant to 10 CFR 50.54(hh)(2) and the acquisition of additional sets of equipment as needed to address multiunit events. The task force also recommended a rulemaking

> to revise 10 CFR 50.63 to require each operating and new reactor licensee to (1) establish a minimum coping time of 8 hours for a loss of all ac power, (2) establish the equipment, procedures, and training necessary to implement an "extended loss of all ac" coping time of 72 hours for core and spent fuel pool cooling and for reactor coolant system and primary containment integrity as needed, and (3) preplan and prestage offsite resources to support uninterrupted core and spent fuel pool cooling, and reactor coolant system and containment integrity as needed, including the ability to deliver the equipment to the site in the time period allowed for extended coping, under conditions involving significant degradation of offsite transportation infrastructure associated with significant natural disasters.

The second set of conclusions and recommendations is included in the section of the report on SFP Safety, where the task force concluded the following:

> 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: (1) SFP instrumentation, (2) safety-related ac power for SFP makeup, (3) technical specification revision regarding onsite ac power for SFP makeup and instrumentation, and (4) a seismically-qualified spray capability. The task force also recommended rulemaking or licensing actions (or both) to require the above actions.

The U.S. nuclear industry has also undertaken various studies related to spent fuel storage and transportation. Examples include the following:

- Electric Power Research Institute (EPRI) TR-1003011, “Dry Cask Storage Probabilistic Risk Assessment Scoping Study,” issued in 2002
- EPRI TR-1009691, “Probabilistic Risk Assessment (PRA) of Bolted Storage Casks: Updated Quantification and Analysis Report,” issued in 2004
- EPRI TR-1021049, “Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling,” issued in 2010
- EPRI TR-1025206, “Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage After Five Years of Cooling, Revision 1,” issued in 2012

The last two reports are of particular interest for the present effort. EPRI TR-1021049 assesses the cost and risk impacts (from a worker dose perspective) associated with transfer of spent nuclear fuel from SFPs to dry storage after 5 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 as a result of the additional handling of spent fuel. With respect to SFP accidents, the report estimates that an additional 711 dry storage packages would have to be handled, as compared to the case without expedited fuel movement, thus increasing the risks associated with cask movement (based on a need to reduce the number of assemblies in some casks when loading more recently-discharged fuel to maintain overall heat load limits). A report prepared by NAC International entitled, “NAC White Paper on Establishing a Balanced Perspective on Wet and Dry Storage of Used Fuel at U.S. Reactors,” dated July 7, 2011, makes similar arguments with respect to the impacts of expediting fuel movement (NAC, 2011).

The updated EPRI study, EPRI TR-1025206 (EPRI, 2012), revised the 2010 study to evaluate the dose and cost impacts of accelerating transfer of spent fuel considering two scenarios – one where the campaign takes 10 years and one where it takes 15 years. The report also adds estimates of the reduction in Cs-134 and Cs-137 inventories in the SFP due to accelerated transfer of spent fuel. The updated report estimates the worker doses to be much higher than the 2010 study projected (3 to 4 times higher), while the costs are roughly equivalent. The reduction in Cs-134 and Cs-137 inventories reported range from 43% to 53%. None of the industry studies attempt to calculate offsite consequences associated with postulated SFP accidents, which is a significant difference between those studies and the study documented in this report.

Regarding the amount of fuel older than five years, and its associated decay heat, the table below compares industry averages reported in the NAC study with those from the study presented in this report.

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%	58-90%
5-9	22%	27%	22%	6-22%
10-14	16%	18%	9%	2-8%
15-19	15%	19%	6%	1-7%
20-24	10%	17%	3%	1-4%
25-29	6%	1%	1%	0-1%
30-34	4%	-	<1%	-
Remainder	4%	-	<1%	-

The NAC white paper and the latter (2012) 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 SFP heat load (and thereby the cooling requirements and mitigative time available for beyond-design-basis SFP 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 range of values in this study represent snapshots during the operation cycle). That said, the values from the reference plant (across the representative operating cycle) only strengthen the argument that the SFP heat load is driven by the fuel less than five years old.

1.8 Potential Follow-On Work and Related Activities

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, the following:

- NRC Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (NRC, 2012g)
- NRC Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (NRC, 2012h)
- 10 CFR 50.54 (f) letters to licensees to review seismic hazards (NTTF Recommendation 2.1)
- 10 CFR 50.54 (f) letters to licensees to review onsite shift minimum staffing levels for emergency response and performance of mitigating strategies in accordance with Order EA-12-049 (NTTF Recommendation 9.3)

- SECY 12-0095, Recommendation AR 5 "Expedited Transfer of Spent Fuel from Spent Fuel Pools to Dry Storage"

- an ongoing rulemaking related to security requirements for ISFSIs

- reevaluation of the role of defense-in-depth in regulatory decision-making

- reconsideration of the use of land contamination and economic consequences in the context of regulatory decision-making

- assessment of the effects of seismic events and accident conditions on neutron absorber materials used in SFPs

- performance of a site Level 3 PRA for Vogtle Units #1 and #2 (operating PWRs), including consideration of both wet and dry storage per SECY-11-0089.

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.

1.9 Layout of Remainder of This Report

The remaining sections of this report provide the following information:

- 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
- human reliability analysis
- sensitivity studies to investigate selected assumptions
- comparison of results with past wet and dry storage consequence and risk studies
- summary of backfitting screen analysis

Finally, Appendix A provides details on the emergency response models, Appendix B provides a gap analysis related to the larger question of assessing the impacts of expedited fuel movement, Appendix C provides study-related correspondence with the US NRC's Advisory Committee on Reactor Safeguards (ACRS) and Appendix D provides a regulatory analysis for expedited transfer at the reference plant.

2. MAJOR ASSUMPTIONS

2.1 Study 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 combined with the fact that the nuclear reactors that felt the higher tsunami waves and stronger ground motions from the March 11, 2011 Tohoku earthquake, which includes those at the Fukushima Daiichi nuclear power plant, were all BWR reactors. Its selection does not denote a belief that this type of design is more vulnerable.
	The **beyond design basis earthquake** is assumed to occur. This is an unlikely event.	The earthquake studied has an estimated frequency of occurrence of one time in 60,000 years. The likelihood of the event is included in the reporting of frequency-weighted consequences.
	The **reliability of mitigation** is not included in the likelihood estimates provided in Chapter 5 through Chapter 7.	The accident progression and consequence analyses were originally conducted without the benefit of a human reliability analysis. The results were then used to frame a human reliability analysis, which is provided in Chapter 8.
	Multi-unit / concurrent reactor accidents are not, in general, considered.	Specifically, the reactor (and its decay heat) is 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 human reliability analysis presented in Chapter 8 does consider multi-unit effects, and Sections 9.3 and 2.2 further discuss this assumption.
	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 1.5 for more information on this assumption.
	Multi-unit aspects are only considered for certain aspects of the study.	See Section 2.2 for more information on this assumption.

Topical Area	Major Assumption	Comment
	Inadvertent criticality events are not considered.	See Section 2.3 for more information on this assumption.
	Other considerations associated with expedited fuel movement.	See APPENDIX B: for a qualitative consideration of fuel movement and APPENDIX D: for quantitative considerations.
Seismic Hazard Characterization	**Vertical PGA** equal to the horizontal PGA and vertical spectral accelerations equal to the horizontal spectral accelerations	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 which indicates that for the seismic bins of interest (high PGA, low likelihood hazard) the contributors to the hazard would be earthquakes with magnitudes less than 6 at about 20 km from the site.
	Seismic hazard models - this study used the existing USGS 2008 model instead of the model in the ongoing program.	A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (NRC, 2012b), it was not available at the start of this scoping study. In addition, the GMPE update is still in progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is appropriate to use for this study because it was the most recent and readily available hazard model for the selected site at the start of the study.

Topical Area	Major Assumption	Comment
Structural and Related Initial Damage State Characterization	**In-structure response spectra (ISRS)** for the study are obtained by scaling ISRS developed for the seismic PRA for PBAPS for the NUREG-1150 study.	Given the differences in the ground motions for the NUREG-1150 PRA and for this study, the use of this scaling is likely to be at or somewhat past the limit of acceptability. The scaling was nevertheless used because it is consistent with the practice of expedited studies of risk or margin. In addition, the assumption (and the uncertainties that it introduces) were deemed to be consistent with the uncertainties in other approximations used in the structural and seismic assessments for the study.
	A **static nonlinear pushover analysis** is used to estimate the overall response of the SFP, concrete strains and cracking, and related liner strains. This analysis used equivalent seismic forces, including hydrodynamic forces, based on elastic ISRS.	As the structure cracks and behaves in a nonlinear manner, it becomes sensitive to frequencies less than its elastic frequencies and dissipates energy, especially if the structure can respond in a non-brittle failure mode. For the ground motions considered in this study, lower frequencies of vibration tends to correspond to lower loads on the structure because spectral accelerations also decrease as the frequencies of vibration decrease. The approach used is, therefore, thought to be conservative for the frequency content of the ground motion expected at the site. Some of this conservatism was accounted for, in part, by: (i) considering a higher damping ratio for the SFP/Reactor-building system (10-percent) than that used for the design basis loads, (ii) calculating equivalent loads using a reduced concrete stiffness, and (iii) by considering a small range of reduction for the calculated liner strains that also accounts for possible ground motion incoherency. A preferred approach to account for this conservatism and also ground motion incoherency effects would involve the sampling of representative ground motion time-histories, both vertical and horizontal, and their use in time-history analyses of the coupled reactor building and SFP structures. This approach, ideally also modeling the small embedment of the reactor building using soil-structure interaction analysis, would more rigorously account for the anticipated conservatisms referred to above but was deemed to be outside the scope of the study.
	Aftershocks are not likely to induce subsequent additional damage to the SFP	The main event would crack the SFP in this study, however it is expected that the SFP's structure would remain stable after the earthquake and resistant to additional loading cycles at this level.

Topical Area	Major Assumption	Comment
	No significant **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 as a direct result of the seismic event itself.
	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 (i.e., do not depend on air pressure, ac power, or dc power) that are unlikely to fail under the earthquake.
	Failure of nearby dams is not explicitly addressed.	The Conowingo Dam is located approximately 9 miles downstream of the site. Failure of that dam could not flood the site. It could lead to additional complications for accident management strategies relying on the river as a water source. The Holtwood Dam is located approximately 6 miles upstream of the site. Failure of this dam, or partial failure in combination with the probable maximum flood, is considered in the plant's Updated Final Safety Analysis Report (UFSAR), in Section 2.4.3.5. Based on the UFSAR, should a complete failure in the upstream dam take place, the rise in the Conowingo Pond level at the site is not expected to exceed grade level, since the pond is about 1 mi wide at the site and the water level would be relieved by the downstream dam.
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.	This assumption is not expected to have a significant effect on the results. See Section 5.2 for more information.

Topical Area	Major Assumption	Comment
	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 (along with the larger volume of water) until the point that water level drops below the fuel transfer canal (and the reactor well and SFP become hydraulically disconnected). That being said, radiological release 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. 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.	This fuel would be placed in 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. See Section 5.2 for additional information.
	Use of a **1x4 pattern**, rather than the 1x8 pattern currently in use at PBAPS.	The 1x8 pattern currently in use at PBAPS is believed to be atypical and is not required by regulation. The timing of obtaining the actual pool configuration, along with modeling conveniences associated with how the MELCOR SFP model is currently designed, also played a role in the decision to use the 1x4 pattern. In cases where the use of a 1x8 pattern might affect study conclusions, this is identified, and Section 9.2 investigates this assumption.
	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 pattern (1x4). Additionally, to get the full benefit of low-density racking, BWR fuel would likely need to have the channel boxes removed.
	Effects on results if a **contiguous storage pattern** were used during the outage.	See Section 9.3 for more information on this assumption.

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. Note that Section 5.4 does include information about dose rates on the refuel floor associated with uncovering a single assembly in the lifted position.
	50.54(hh)(2) **mitigation capacities** (i.e., 500 gpm makeup delivered or 200 gpm spray delivered) are based on the generic NRC-endorsed capacities in NEI-06-12, Revision 2.	For PBAPS, the capacities of the available equipment are somewhat 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
	Mode of mitigation deployment (i.e., use of makeup versus spray)	For OCP 1 and OCP 2 with the "moderate" leakage condition, makeup is deployed. Other, equally reasonable assumptions about mitigation deployment could result in the deployment of sprays instead (which have a potential advantage in terms of mitigation for these conditions). This difference in mode of deployment shows the potential benefit of the additional instrumentation required by NRC Order EA-12-051. A sensitivity study related to this assumption is presented in Section 9.3, for a uniform pattern.
	Use of ac power fragility as a surrogate for **loss of normal SFP cooling** and makeup availability	This study used the ac power fragility from NUREG-1150 of 0.84 as a surrogate for the conditional probability of normal SFP cooling and makeup not being available. This simplifying assumption was made in light of the fact that the study is not a PRA (but rather a consequence analysis with probabilistic considerations) and that this value already approximates the upper bound of 1. In reality, the availability of normal SFP cooling and makeup would be a combination of the AC fragility, the fragility of the actual equipment and its support equipment, and operator actions to recover the equipment, which could result in a conditional probability higher than the value used here.

Topical Area	Major Assumption	Comment
Accident Progression Analysis	The study uses best-estimate **ruthenium release rates** calculated by the MELCOR code. 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 spanning a very large range of uncertainty rather than mechanistic and integrated modeling. Section 6.1.5 of this report provides additional information.
	Radionuclide releases occur only if the fuel has become uncovered by 48 hours and the radiological release has commenced before 72 hours. Otherwise, the study assumes the scenario results in no offsite consequences.	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, the project staff judged 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 of the report for additional discussion on mitigation assumptions in this study and Section 9.8 for time truncation sensitivity.
	The study does not consider **debris** entering the pool as a result of any modeled **hydrogen combustion event**.	Such debris could be generated and could fall into the pool. However, the occurrence of a hydrogen combustion event in this study denotes that the fuel in the SFP 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 combustion event from a concurrent reactor accident has the potential to generate debris which could impair SFP natural circulation air or steam cooling (should the fuel in the SFP become uncovered) for conditions in which the fuel might otherwise be cooled by means of these passive cooling modes. However, this latter situation is inherently tied to the study's lack of a comprehensive treatment of multiunit aspects.
	Aerosol resuspension inside the reactor building, such as from hydrogen deflagration, will not be significant.	Hydrogen burns in the refueling 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.

Topical Area	Major Assumption	Comment
	The study does not consider the effects of **molten core-concrete interaction (MCCI).**	The MELCOR code models heat transfer from the debris to the pool floor, as well as the fission product release from hot debris. In some cases, the debris temperature remains above typical concrete ablation temperatures (~1500 K). MCCI may occur in selected scenarios in which the fuel relocated to the bottom of the pool following the failure of the rack baseplate and its temperature exceeded the concrete ablation temperature. These cases involve large-scale debris relocation and large releases of volatile fission products. Even without MCCI, the fuel in debris form continues to release fission products resulting in very large releases of volatiles. Section 9.5 of this report presents a sensitivity calculation.
	The effective time dependent **decontamination factor** (DF) of the reactor building can be used to reasonably estimate a cumulative release.	The use of an effective DF is based on a new methodology (see Section 6.1.5 of the report) for SFPs 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, and not just the chemical group, to allow the offsite consequence code to process the source term.
	Criteria for **release of radionuclides in the fuel cladding gap**	MELCOR does not have a fuel cladding deformation and strain model. It uses a value of 900°C for widespread cladding failure. NUREG-1738 cites a temperature range of 700–850°C for rod ballooning and burst; however, the security assessment work mentioned in Section 1.7 showed that rod ballooning has a low impact on the timing to breakaway oxidation and the impact on the peak cladding temperature response was relatively insignificant. In addition, NUREG-1738 assumes 900°C as the temperature at which the onset of significant fission product release is expected. In general, there may be some fuel cladding failures at lower temperatures but MELCOR is mostly concerned with larger thermal release from the fuel. In this sense, the gap release temperature of 900°C is taken as a surrogate for start of rapid fuel heat up associated with breakaway oxidation and initiation of zirconium fire and its propagation.
Offsite Consequence Analysis	Calculated results are from **atmospheric-type releases** only	Atmospheric releases are the primary scope of the project.

Topical Area	Major Assumption	Comment
	A straight-line **Gaussian plume** segment dispersion model is used for the atmospheric transport	The current model is a straight-line transport of plume segments; therefore, it does not capture the effects from changes in wind direction after the plume segment has been released. Despite this, the atmospheric transport model in MACCS2 has compared favorably to Lagrangian particle tracking models [NRC 2004]. This is because the use of ensemble averages of many meteorological conditions, such as the consequences reported in this study, has been shown to make reliable weather-averaged results.
	Distance truncation (from point of release)	Health effect risk estimates (e.g. latent cancer fatality risk and early fatality risk) are with respect to distance. The reported latent cancer fatality risk includes all distances that have doses above the modeled dose limit for habitability, as determined by the Pennsylvania Code Title 25 § 219.51. Total health effect estimates are not a function of distance, and have no distance truncation.
	The effect of **low dose radiation** on latent cancer fatalities is uncertain, and therefore a range of dose truncations are reported.	See Section 7.1.3 for more information on this assumption.
	The **public** will behave in an orderly fashion during a severe accident, and can be represented by cohorts.	See Section 7.1.4 more information on this assumption.
	The seismic event has a limited effect on **emergency response**.	The study assumed that the seismic event would not significantly affect emergency response. This is based on an assessment in NUREG-1935 of the same site and seismic event that assumed the damage to local infrastructure is limited to 12 bridges, partly due to the few large structures in the area. Also, the extended loss of ac power is assumed to be limited to the EPZ (~10 miles) due to the assumption that the strength of the seismic event is from the proximity of the seismic event to the site, rather than being a wider impact from a larger magnitude. See section 7.1.4 for more details.

Topical Area	Major Assumption	Comment
	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 guidance is currently being drafted. In addition, guidance could recommend the development of localized cleanup goals after an accident, to account for sociopolitical, technical, and economic considerations. Given that 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 to decide what land is to be decontaminated. This is consistent with previous studies. See Section 7.1.5 for more information on this assumption.
	A single value for **habitability** is used for all affected areas.	See Section 7.1.5 for more information on decisions regarding land interdiction and associated relocation of the public.

2.2 Multi-Unit Considerations

Observations Regarding a Concurrent Reactor Event:

There are four broad interplays that can be defined between the SFP and the reactor:

(1) an initiating event that directly affects both the reactor and the SFP

(2) a reactor accident that prevents accessibility to the SFP for a prolonged period of time (e.g., due to high radiation fields), leading to a SFP accident

(3) a reactor accident that includes ex-containment energetic events (e.g., a hydrogen combustion event) or other ex-containment interplays (e.g., steaming through the drywell head that affects refuel floor combustible gas mixtures) and creates a hazard to the SFP (e.g., by causing debris to fall in to the pool) or otherwise changes the SFP event progression[5]

(4) an SFP accident that prevents accessibility to key reactor systems and components for a prolonged period of time or 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

[5] For instance, a hydrogen combustion event caused by a reactor accident that affects the refuel floor superstructure can lead to additional oxidation (for an otherwise oxygen-limited situation), which in turn may result in higher releases from the SFP. Note that this can also include positive effects, in the sense that steam leaking through the drywell head can serve to steam inert the refueling floor.

pool or excessive condensation from continuous boiling of SFP water), leading to a reactor accident

For each of these interplays, large seismic events and severe weather SBO events are logically the most relevant initiators, as they are the type of initiators that are most likely to initiate an accident at the reactor and SFP, while simultaneously hampering further 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. The study also explores these effects on mitigation (Section 8), and addresses some aspects of the uncertainty associated with this treatment (Section 9). However, explicitly modeling multiunit effects was not a focus of this study, because of the existing limitations with the available computational tools. An ongoing project described in SECY-11-0089 will attempt to more rigorously address these effects in the framework of a multiunit Level 3 PRA for Vogtle Electric Generating Plant Units 1 and 2.

<u>Observations Regarding a Multiunit Event:</u>

Along with the possibility of a concurrent SFP and reactor accident, there is the possibility for a concurrent accident at the SFP of one unit with an accident at the SFP or reactor of the other unit. Again, a large seismic event or a severe weather SBO are the events that are most likely to lead to a multiunit event. In general, if accidents at both SFPs 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 unlikely because the two (or more) SFPs are very unlikely to have the same total pool heat load or peak assembly heat load. (Recall that for multiunit sites, the reactors did not usually start operation at the same time and outages are intentionally staggered.) Even if this symmetry did exist, the offsite consequences would not follow a linear scaling because of a number of nonlinearities associated with that portion of the analysis. Again, capturing these effects was not a focus of this study, and future work (the SECY-11-0089 Level 3 PRA) will attempt to more rigorously treat these effects.

2.3 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. Further, if an ICE were severe enough to produce significant heat, the fuel will be harder to cool and short-lived radionuclides will be produced. 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 reference plant (and other U.S. SFPs), 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 by means of either (1) direct movement of the fuel, (2) direct degradation of the poison material, or (3) indirect effect on either the fuel or poison material because of high temperatures associated with an induced accident, there are several “advantageous” considerations, including the following:

- The reactivity of fresh BWR fuel is suppressed by the high content of burnable absorbers.

- The majority of the fuel in the SFP has low net reactivity since it has gone through more than one operating cycle in the reactor.

- 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).

- 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.

- 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.

- Low-enriched uranium fuel assemblies (which are used in all U.S. reactors) are generally 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.

There are also a few counter considerations:

- 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 subcriticality for high reactivity assemblies).

- The effects of large seismic events on already degraded SFP rack poison material are not easy to quantify.

- The rack panels and poison material have a lower melting temperature than the cladding and fuel.

- Termination of a SFP ICE during an event that required deployment of mitigation equipment could be difficult.

- The possibility of a criticality event cannot be summarily dismissed.

Finally, the offsite 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 reflood. In consideration of all of the above, common accident management practices in the United States call for the use of any available water in responding to fuel uncovery in either the reactor or SFP. This study shows the 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.

3. SEISMIC HAZARD CHARACTERIZATION

3.1 Basis for Probabilistic Estimates

The primary sources of information for seismic hazard estimates at nuclear power plant sites include (1) the NRC/ LLNL (Bernreuter et al., 1989; Sobel, 1994) model (hereafter referred to as the LLNL model); (2) the EPRI model (Toro et al., 1989); and (3) the USGS model developed in the mid-2000s (Peterson et al., 2008) (hereafter referred to as the USGS 2008 model. The implementation of the individual plant evaluation for external events (IPEEE) program utilized both the LLNL and EPRI models (NRC, 2002b). The National Seismic Hazard Mapping Project utilized the USGS 2008 model. The NRC also utilized the USGS 2008 model for the seismic hazards estimates used in screening level assessments for Generic Issue 199 (GI-199) (NRC, 2012a).

The seismic hazard assessment in this study is the US Geological Survey (USGS, 2008) hazard model. A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (NRC, 2012b), it was not available at the start of this scoping study. In addition, the GMPE update is still in progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is appropriate to use for this study because it was the most recent and readily available hazard model for the selected site at the start of the study.

Figure 2 (PGA) and Figure 3 (1-, 5-, and 10 Hz spectral acceleration) graphically show comparisons of hazard estimates for the reference site (a rock site) with the three information sources listed above. These comparisons are provided to compare the model used in this study to well-known and extensively documented information sources (LLNL model and EPRI model) that were used in past SFP risk studies. The comparisons support the following observations:

- For the PGA, the USGS 2008 model predicts higher annual probability of occurrence for high-level, low-probability events, specifically for events with PGAs greater than about 0.35g.

- For moderate PGAs, from about 0.1g to 0.35g, the LLNL model is higher than the USGS 2008 model. For events above about 0.35g, which are lower probability events, the USGS 2008 model is higher than the LLNL model until both hazard estimates differ by factors of about 2.5 at 0.75g and about 3 at 1.0g.

- The EPRI model hazard estimates are lower than those from the USGS 2008 model for all PGAs. Specifically, hazard estimates based on the USGS 2008 model are about 2 times greater at 0.2g with the difference increasing to about 10 times at 1.0g.

- Thus, in terms of PGA, the seismic hazard estimates used for this study are about 2.5 times greater than LLNL model estimates and about 6 times greater than EPRI model estimates at 0.75g.

- Curves for the USGS 2008 model and the LLNL model are comparable for each representation, with the USGS 2008 model sometimes being higher (higher annual probability of occurrence) and the LLNL model 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 this is the fact that the 5-Hz LLNL model curve, which is higher than the 10-Hz curve.

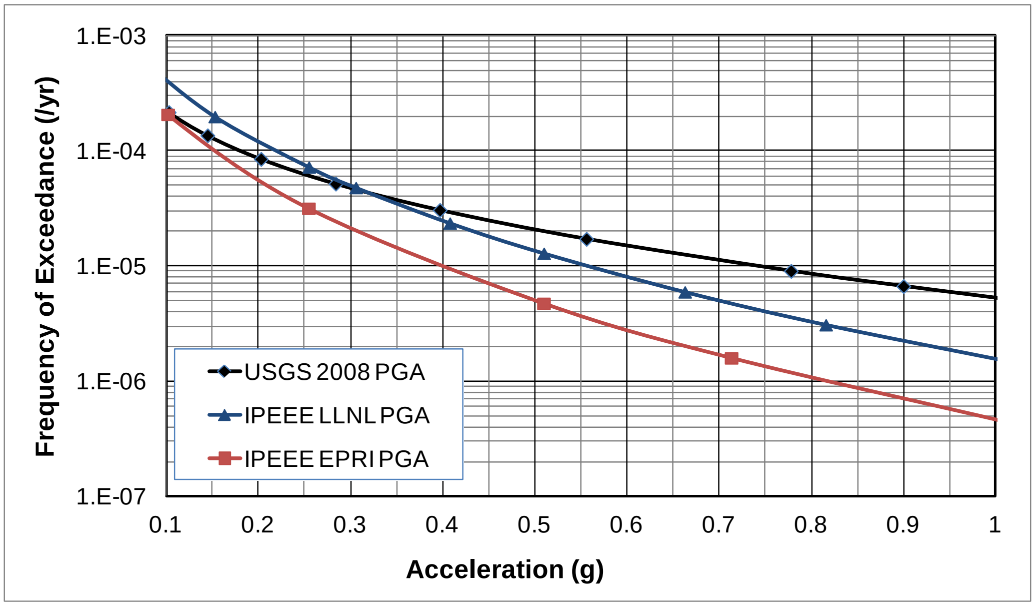


Figure 2 Comparison of PGA exceedance frequencies at the reference plant

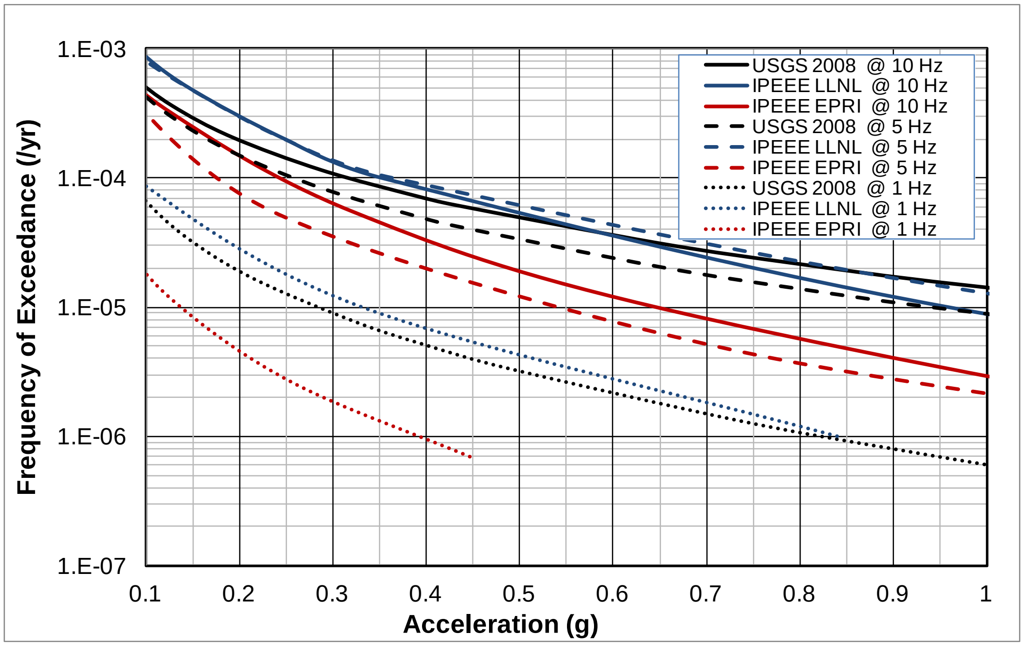


Figure 3 Comparison of spectral exceedance frequencies at the reference plant (rock hazard curves)

A comparison of the annual frequency of exceeding a given PGA for all Mark I sites (Figure 4) shows that the reference plant 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 5), the reference site is in the upper half of the group.

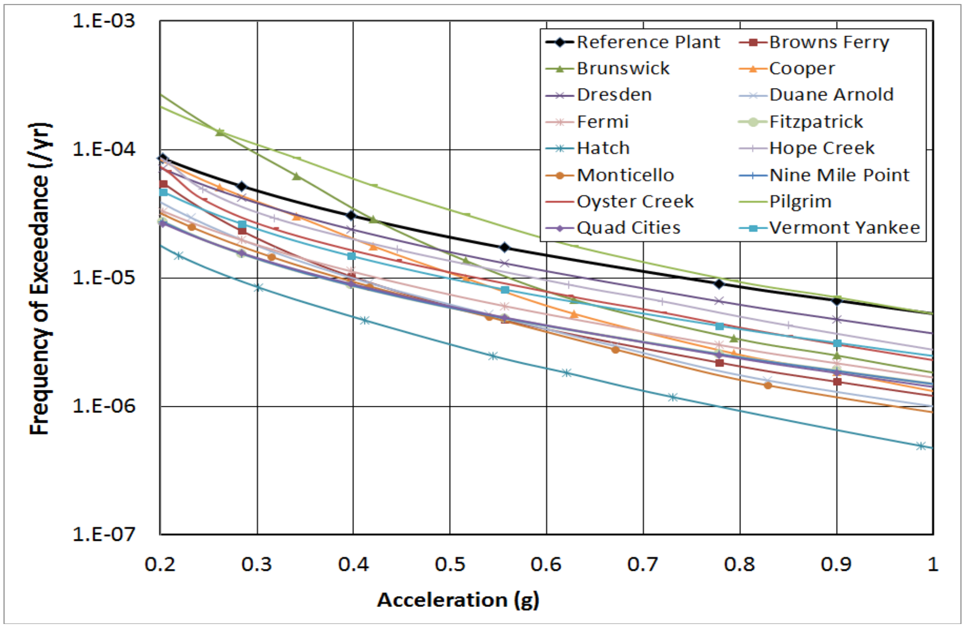


Figure 4 Comparison of annual PGA exceedance frequencies for U.S. Mark I reactors (USGS 2008 model) (rock hazard curves)

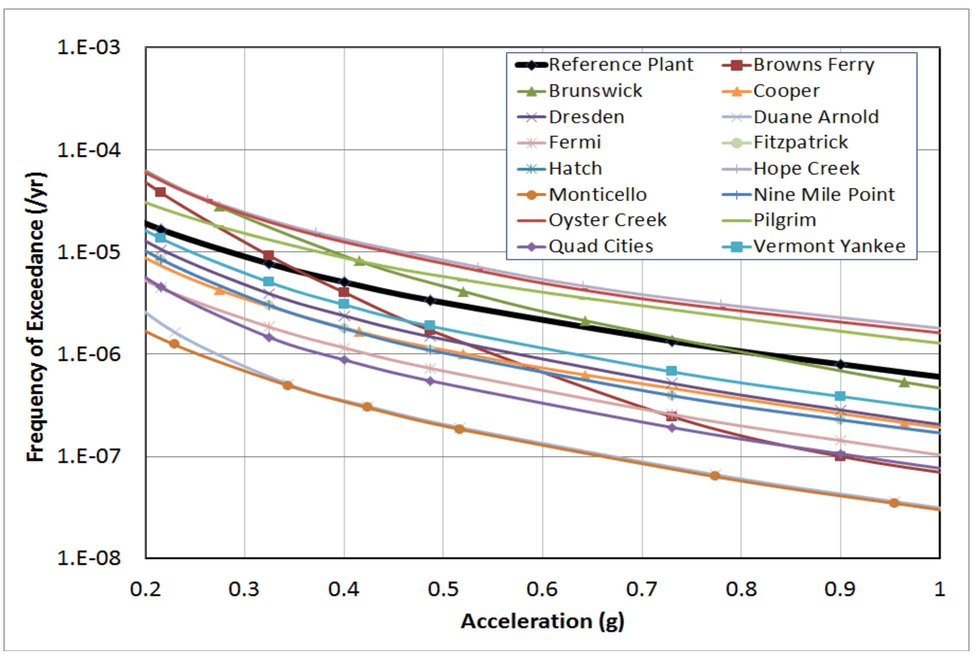


Figure 5 Comparison of annual exceedance frequencies for 1 Hz spectral accelerations for U.S. Mark I reactors (USGS 2008 model) (rock hazard curves)

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). Revision 1.01 of the NRC handbook entitled, "Risk Assessment of Operational Events, Volume 2—External Events," issued January 2008 (NRC, 2008b), recommends the use of at least three bins unless plant-specific considerations require more bins. The SFPS used four bins.

Table 4 shows the resulting bins, along with the tabulated frequencies for various spectral and peak accelerations. Note that for bin 4, the representative bin PGA has been set to 1.2g by convention, whereas for the other bins, it is the geometric mean of the interval endpoints. Figure 6 shows these results graphically.

Table 4 Seismic Bins and Initiating Event Frequencies

Bin #	Bin Range (g)	Bin PGA (g)	Approximate Initiating Event Frequency (USGS 2008 model) (/yr)
1	0.05 - 0.3	0.12	5.2×10^{-4}
2	0.3 - 0.5	0.4	2.7×10^{-5}
3	0.5 - 1.0	0.7	1.7×10^{-5}
4	> 1.0	1.2[1]	4.9×10^{-6}

[1] Assumed based on PRA modeling convention

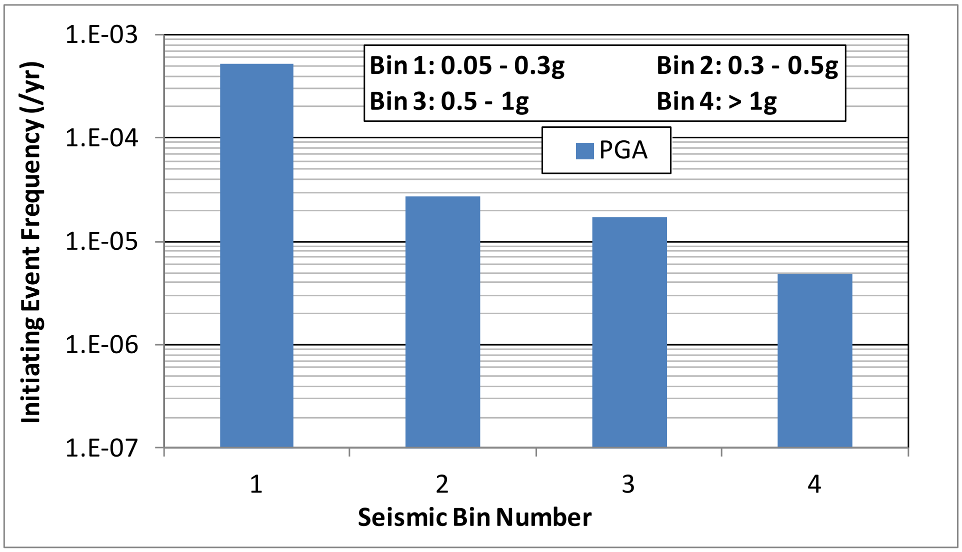


Figure 6 Comparison of seismic initiating event frequencies

Based on this information, and on a review of the results of past studies which indicate that damage to the SFP and other relevant structures, systems, and components (SSCs) is not credible for events in bins 1 and 2, this study focused on bin 3. The project staff concluded 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 (NRC, 2001)) indicates that events in bin 3, with initiating annual frequencies on the order of 1×10^{-5} to 2×10^{-5}, could challenge the integrity of the SFP (i.e., of causing a leak) at the reference plant. Thus, this is the initiating earthquake chosen for this study. It is an event that is no more severe than events considered in past reactor and SFP PRAs and consequence studies.

This study therefore considers a challenging, but very low probability earthquake as the initiating event, selected based on the considerations indicated above. This decision translates into a seismic event with a PGA several times greater than that associated with the design-basis earthquake, currently called the safe-shutdown earthquake or SSE. The PGA for the reference plant SSE is 0.12g. (This is about a magnitude 5.3 earthquake at about 25 kilometers (km).) While the probability of occurrence of this earthquake was not used in its selection, 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 (NRC, 2001; Prassinos et al., 1989) and engineering judgment, was that the ground motions associated with the SSE (bin 1) would not be large enough to damage the SFP at the reference plant.

The information above coupled with the review of previous studies (NRC, 2001) suggests that the frequency of a seismic event that could challenge the integrity of the SFP at the reference plant is on the order of 1.7×10^{-5} per year (i.e., approximately one event in 60,000 years) or less. Table 5 contrasts this frequency against other sources of information. The Mineral, VA, earthquake of August 23, 2011, which occurred near the North Anna nuclear power plant, can serve as a point of reference. In that case, the NRC staff concluded, using data from USGS instruments, that the PGA at the North Anna site was about 0.26g (NRC, 2011b). Using the USGS 2008 information for North Anna, the hazard frequency for an event with this PGA is about 1.2×10^{-4} per year (one event every 8,300 years). This frequency places the Mineral, VA, event in bin 1.

Table 5 Comparison of Seismic Frequencies from Various Sources

Source	**Estimated initiating event frequency of a large seismic event**[1]	**Notes**
USGS 2008	$1.7\text{x}10^{-5}$/yr (one event in 60,000 years)	Frequency of seismic bin 3 of 4 (0.5 to 1g)
The reference plant's SPAR-EE Model (v3.21, Rev. 1)	$1.3\text{x}10^{-5}$/yr (one event in 77,000 years)	Frequency of seismic bin 3 of 3 (> 0.5g)
NUREG-1738[1]	$1.1\text{x}10^{-5}$/yr (one event in 90,000 years)	Frequency of seismic hazard between 0.51g to 1.02g

[1] Initiating event frequencies reported are those based on the LLNL models (Sobel, 1994).

In addition to the PGA, ground motions at a site are also characterized by their frequency content expressed in terms of response spectra. Section 3.3 describes the procedure used to develop the horizontal and vertical acceleration response spectra for the input ground motion for this study.

Other response spectra of interest for this study are (1) the plant's SSE response spectra and (2) the free-field response spectra used in the seismic PRA for the NUREG-1150 study. 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. Volume 1, Part 3, of NUREG/CR-4550, "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events," issued December 1990 (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. As shown in Section 3.3, the spectral shape for this study differs from the SSE response spectrum, as well as from the median response spectra considered in the NUREG-1150 seismic PRA. Frequency content for the SSE and the NUREG-1150 PRA spectra generally resemble each other.

3.3 Characterization of the Ground Motion Response Spectra

Spectral shapes developed for the safety/risk assessment results for the GI-199, which utilized the USGS (2008) model, were used to develop the free-field acceleration response spectra for this study. The free-field acceleration response spectrum developed for the GI-199 for this site has a zero-period spectral acceleration (PGA) of about 0.34g. The acceleration response spectra for the free-field ground motion for the initiating seismic event considered for this study (bin 3 in Table 4 and a PGA of 0.7 g) were derived from the GI-199 spectra shape as follows:

- Horizontal shaking: horizontal response spectrum is the GI-199 spectral shape scaled to the bin 3 PGA (zero-period acceleration) of about 0.7g (specifically 0.71g). While it is recognized that the frequency content of ground motions may change somewhat with increasing PGA levels, scaling of the spectral shape for the 0.34g PGA to the bin 3 PGA is considered a reasonable approximation for low probability hazard for this rock site and for the purposes of this study. Figure 7 compares the horizontal input acceleration response spectra for this study to the horizontal response spectrum for the plant's Safe Shutdown Earthquake (SSE) (PGA of 0.12g) for 5-percent damping.

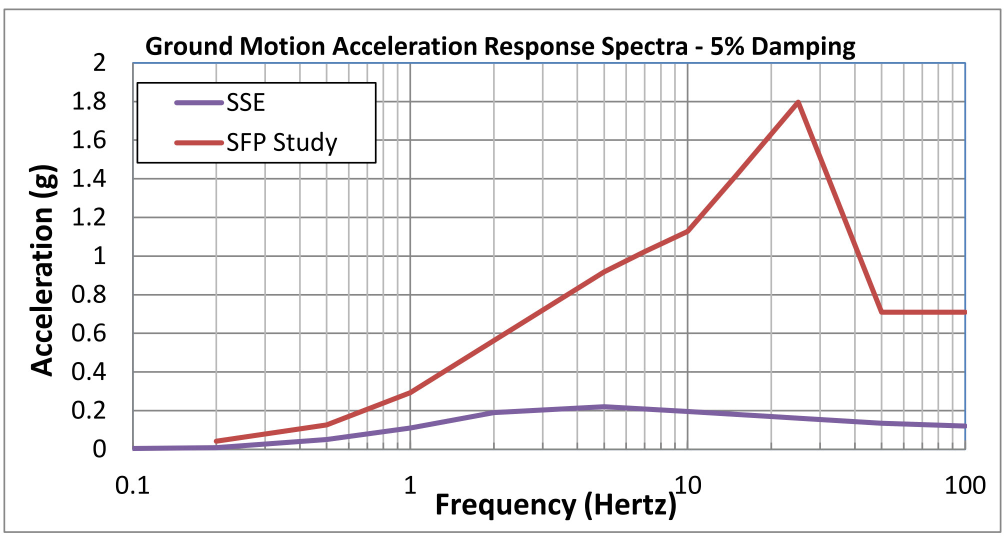


Figure 7 Input acceleration response spectrum and SSE (Horizontal Ground Motion)

- Vertical shaking: vertical spectral accelerations and the vertical PGA (0.7 g) are assumed to be the same as the horizontal spectral accelerations and PGA. 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 which indicates that for the seismic bins of interest (high PGA, low likelihood events) the contributors to the hazard would be earthquakes with magnitudes less than 6 at about 20 km from the site.

Other response spectra of interest for this study are the free-field response spectra used in the seismic PRA for the NUREG-1150 study (Lambright et al., 1990). These spectra are of interest because in-structure response spectra calculated for that ground motion were scaled, in approximation, to estimate in-structure response spectra for the free-field ground motion considered for this study. Figure 8 compares the frequency content of the horizontal response spectra (5-percent damping) for the SSE, the median response spectrum used in the NUREG-1150 study, and the spectral shape used in Spent Fuel Pool Study. For this comparison, all spectra are scaled to a PGA of 1.0g. When the three response spectra under consideration are scaled to the same PGA, the information in Figure 8 supports the following observations:

- For frequencies between about 10 Hz and 45 Hz, the spectral shape used in this study 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, which is generally the frequency range most damaging for nuclear power plant structures, the spectral shape used in this study has lower spectral accelerations than the ground shaking considered for the SSE and the NUREG-1150 study.

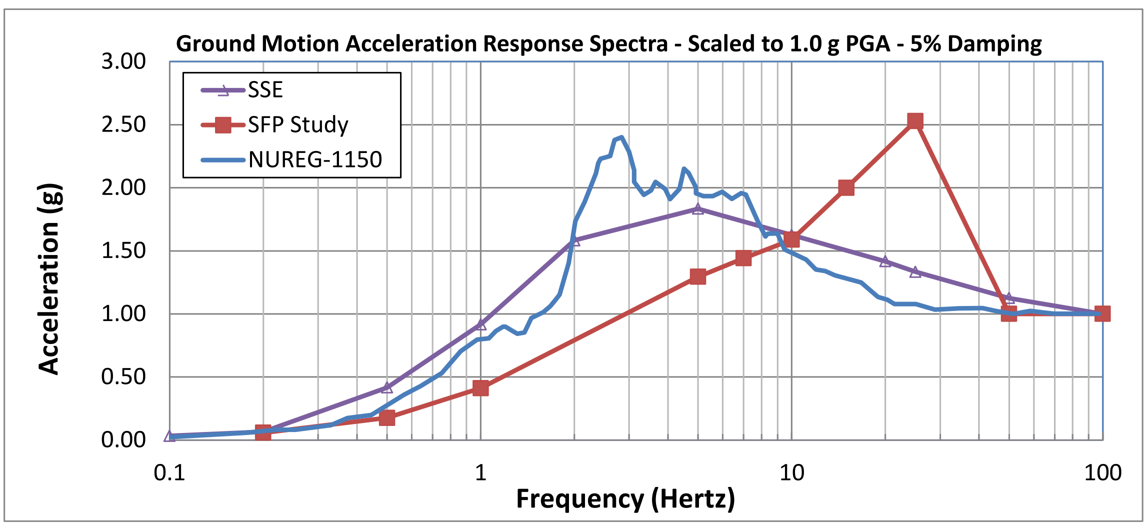


Figure 8 Response spectrum for 5-percent damping scaled to 1.0 g PGA: SSE, NUREG/CR-4550 (NUREG-1150 PRA), and this study (GI-199)

As noted above, the input horizontal acceleration response spectrum for the event considered in this study is the spectral shape derived for the GI-199 study using the USGS 2008 model (PGA of about 0.34g) scaled to a PGA of about 0.7g. Figure 9 shows the horizontal response spectra (5-percent damping) for the event considered in this study, for the SSE (0.12g PGA), and for the response spectrum used in the NUREG-1150 PRA. The NUREG-1150 response spectra shown in the figure is scaled from a PGA of 0.3g to the PGA of the event for this study. 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 in Table 4. They are also richer at the high frequencies (greater than 10 Hz), which generally tend to be less challenging to nuclear power plant structures.

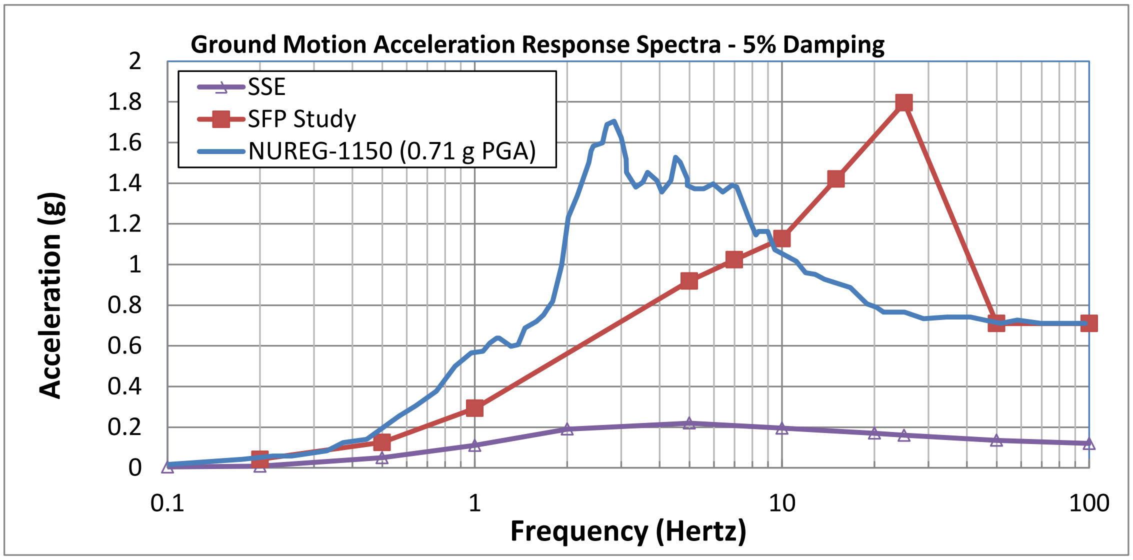


Figure 9 Horizontal response spectrum (5-percent damping): SSE, SFP Study and NUREG-1150 PRA (for 0.71g PGA)

4. STRUCTURAL ANALYSIS AND RELATED INITIAL DAMAGE CHARACTERIZATION

This section documents the structural analysis performed to estimate the initial damage states for the accident progression analysis. It provides:

- the objective,
- the approach including assumptions,
- the structural modeling and analyses, and
- the 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 that would constitute the initial conditions for the accident progression analysis. Structural and related damage states have been divided into the following two major categories:

(1) structural damage to the spent fuel pool structure with potential locations of leakage from concrete cracking and related liner tearing

(2) other damage states including:

- amount of water, if any, displaced by sloshing of the water out of the SFP
- damage to the refuel gate, support systems and penetrations, as well as 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 pool structure, namely concrete distortions, concrete cracking, and metallic liner strains.

Section 4.3 provides a review of the performance of SFPs at four nuclear power plant sites with a total of 20 reactors under two major recent earthquakes in Japan. This review compares relevant aspects of the seismic scenario and estimated damage states for this study with known aspects of the seismic scenario and performance of SFPs affected by those earthquakes. The review summarizes known or presumed reductions in water levels of the SFPs affected by those earthquakes associated with either water leakage from structural damage, if any, or water loss from sloshing. Although these reviews and comparisons use information available at the time of the execution of this study, they assist in the interpretation of the results obtained for the seismic scenario and SFP considered.

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 the approach reported in NUREG/CR-5176, “Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Power Plants,” issued January 1989 (Prassinos et al., 1989) modified to address specific needs of this study. (The general approach is fully described in the following section.) The analyses reported in NUREG/CR-5176 were conducted in conjunction with research activities related to Generic Issue 82 (GI-82) (NRC, 2012c). Appendix 2 to NUREG-1738 (NRC, 2001), a technical study of spent fuel pool accident risk at decommissioning nuclear power plants, also addresses the seismic fragility of spent fuel pools and refers to the results and approach used in NUREG/CR-5176. The seismic evaluations in NUREG/CR-5176 considered ground motions with frequency content that differs from those considered in this study. Specifically, NUREG/CR-5176 considered ground motions with maximum response spectra amplitudes for frequencies below 10 Hz while the ground motions considered for this study have maximum response spectra amplitudes for frequencies greater than 10 Hz. This difference in the characteristics of the ground motions tends to induce conservatism in the approach when applied to this study as indicated below.

<u>Approach</u>

The overall approach used 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 nine steps:

(1) Obtain free-field acceleration response spectra (horizontal and vertical) for the site considered (a rock site and a reactor building with small embedment) as indicated in Section 3.3 and shown in Figure 7.

(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 SFP (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.) The SFP Study used the median-centered ISRS calculated for Peach Bottom 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 et al., 1990).

(3) Estimate ISRS for the ground motions of interest for this study at the elevations of interest (Step 2) by scaling the ISRS from previous studies (Step 2). The scaling accounts for differences in the response spectra for the NUREG-1150 seismic PRA and for this study. Given the differences in the ground motions for the NUREG-1150 PRA and for this study, the use of this scaling is likely to be near the limit of acceptability. Use of this scaling is justified on the basis that the approximations and uncertainties introduced are consistent with the uncertainties in other approximations used in the structural and seismic assessments for the study. It is expected that efforts by the NRC and industry related to Requests for Information in SECY-12-0025 (NRC, 2012f) associated with the Near Term Task Force (NTTF) Recommendation 2.1 (NRC, 2011a) will result in updated staff guidance on ISRS scaling.

(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 (1) peak vertical and horizontal accelerations of the floor and walls of the SFP structure (seismic coefficients), (2) peak vertical and horizontal hydrodynamic impulsive pressures on the floor and walls of the SFP from the water in the pool and (3) 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) Use a simplified three-dimensional (3D) finite element model of the SFP structure to estimate or verify these loads. Specifically, use elastic solid elements and special fluid elements to model the water to estimate natural frequencies and mode shapes for the SFP structure. Use this model to calculate the spatial distribution of peak vertical and horizontal accelerations of the structural components using the ISRS from Step 3 as input.

(b) Calculate hydrodynamic impulsive pressures and peak vertical and horizontal pressures on the basis of simplified methods (Housner, 1963; AEC, 1963; Malhotra et al., 2000). Use the 3D finite element model in Step 4a together with the ISRS from Step 3 as input to estimate peak hydrodynamic pressures. This provides for verification and adjustment of the hydrodynamic pressures calculated using simplified methods.

(c) Use the 3D finite element model in Step 4a together with the ISRS from Step 3 as input 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. Such analysis provides the load deformation behavior of the SFP for the loading pattern and intensity considered. 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, 2006a). 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) Based on guidance for combining effects from three spatial components of an earthquake in Regulatory Guide 1.92 (NRC, 2006a), peak vertical seismic loads were combined with 40-percent of the peak horizontal loads. A combination of peak horizontal loads on both directions with 40-percent of the vertical loads was also considered. Preliminary analyses indicated that the load combination involving peak vertical loads and 40-percent of the horizontal loads would likely 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 median 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 beyond-design-basis 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 events (NEI, 2011; Prassinos et al., 1989).

(6) Review and process the calculated structural distortions (as measured by the displacement of nearby nodes), structural deformations, concrete strains and liner strains for the following purposes:

(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 with 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 such that water leakage is controlled by the size of the tearing in the liner.

(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 through 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 as well as 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. The use of finite element analyses in Step 4 is done 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 and adjust hydrodynamic impulsive loads on the floor and walls of the SFP. The use of finite element analyses in Step 5 is done to track the development of cracking and liner strains and then relate 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 SFP structure. These conservative aspects are as follows:

- As the high-frequency structure of the SFP (fundamental frequency on the order of 15 to 25 Hz) cracks under the applied seismic loads, its natural frequencies decrease and are no longer resonant with the high frequency components of the ground motion (i.e., the frequencies corresponding to the higher spectral accelerations). Since the spectral accelerations decrease as the frequencies of the SFP structure decrease after cracking, the use of seismic loads calculated assuming elastic frequencies can introduce conservatism in the analysis for the seismic event considered. This would be the case if the SFP structure were to remain stable with only minor distortions after cracking as in the case of the SFP studied. This aspect was partially accounted for in this study through a small reduction in the spectral accelerations and by the use of a small reduction of the concrete stiffness in the calculation of the natural frequencies of the SFP structure. Assessment of the conservatism introduced by the approach used, which was outside the scope of this study, would involve the sampling of representative acceleration time-histories, both vertical and horizontal, and their use in time-history analyses of the SFP response to the seismic loads considered.
- Generally ISRS accelerations do not 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, both the structure of the reactor building and of the SFP may crack and dissipate energy thus dampening the response of the building. This effect is taken into account, in part, by reducing ISRS spectral accelerations by the ratio of spectral amplification factors for 10-percent and 5-percent damping (Newmark and Hall, 1978). The reduction of spectral accelerations implied by the use of a higher damping ratio is further justified by the decrease in the fundamental frequency of the structure related to cracking which, for the input free-field ground motion for this study (see Section 3.3 and Figure 9), would decrease the spectral accelerations.
- 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 (1) the scaling of ISRS calculated from a ground motion with response spectra markedly different from the ground motion spectra considered for this study (addressed in item 5(c)), (2) the decoupling of the response of the SFP from the response of the reactor building, and (3) neglecting the small embedment of the reactor building, as also done in previous studies, which may affect the calculation of horizontal ISRS. Follow-on subsections address these and the above approximations. 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 a better assessment of these possible conservatisms. However, these were outside 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 explicitly in the estimation of the initial stresses in the SFP components (Step 5a). The weight of those equipment and appurtenances is expected to be small in comparison to the other dead loads in the pool and accounted for by the approximations in the estimation of those dead loads. 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.

Seismic Loads

Chapter 3 of this report discusses the bases for the free-field ground motion response spectra for the seismic event considered for the SFP Study. As noted in Chapter 3 other free-field response spectra of interest in this study are those documented in NUREG/CR-4550 (Lambright et al., 1990) and used in the probabilistic risk assessment (PRA) for NUREG-1150. That report 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 motion 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 et al. (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 natural 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 dominant, elastic (uncracked) frequencies of vibration of the SFP structure, considering hydrodynamic effects of the water and the mass of the spent fuel, range from about 17 Hz (vertical response of the floor slab) to 28 Hz (horizontal deformations of the walls). 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 Natural Frequencies of Vibration for the Peach Bottom Reactor Building (Lambright et al., 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 et al. (1990) were scaled to estimate floor vertical and horizontal ISRS at the elevation at the bottom of the SFP (Elevation 195 ft) as well as horizontal ISRS at the midheight 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 study. This scaling was done using the reported median-centered ISRS for 5-percent damping, the vertical and horizontal (EW) components for the ISRS (examination of the charts indicates that the horizontal (EW) component tends to have the higher spectral accelerations). The SFP 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 reductions in the high frequency spectral accelerations is provided at the end of this subsection.

Figure 10 shows a comparison of the vertical ISRS for the elevation at the bottom of the SFP, calculated as indicated above to the corresponding ISRS (smoothed) for the NUREG-1150 seismic PRA. Likewise, Figure 11 provides a similar comparison for the horizontal ISRS at the midheight of the SFP structure (average of the ISRS at Elevation 195 ft and Elevation 234 ft).

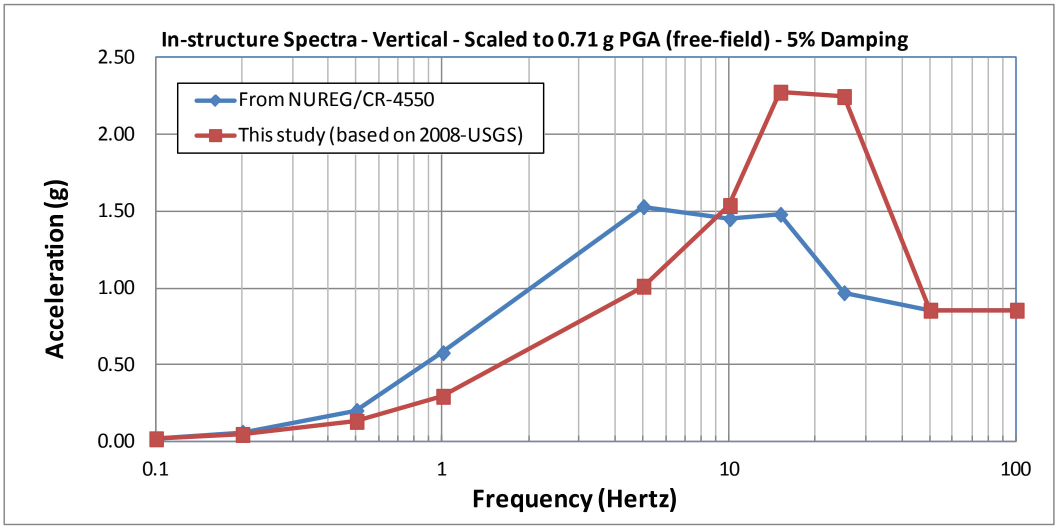


Figure 10 Vertical ISRS for 5-percent damping at Elevation 195 ft (bottom of the SFP)

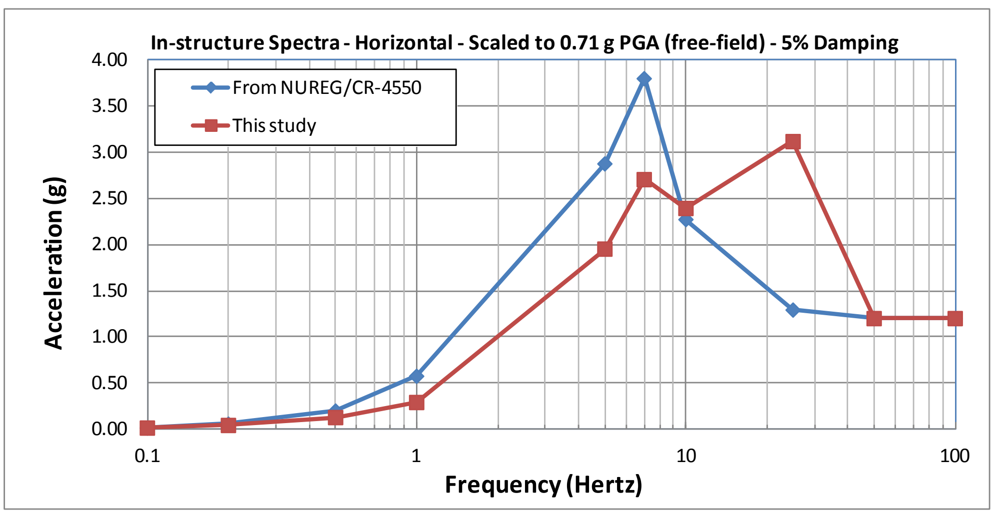


Figure 11 Horizontal ISRS for 5-percent damping midway between Elevation 195 ft and Elevation 234 ft (midheight of the SFP)

The spectra shown in Figure 10 and Figure 11 are for 5-percent damping (reactor building and equipment). Calculation of seismic load coefficients for the SFP floors and walls as well as of hydrodynamic impulsive pressures considered a reduction of these spectral accelerations. Specifically, seismic coefficients and hydrodynamic pressures calculated using the 5-percent damping ISRS were reduced by the ratio of scaling factors for 10-percent and 5-percent damping reported in NUREG/CR-0098 (Newmark and Hall, 1978). As noted above, this is done to account for, in part, the energy dissipation (damping) from cracking of the SFP and minor cracking of the reactor building. This is further justified by the reduction in the natural frequency of the SFP structure from cracking that would lead to reduced spectral accelerations for the input free-field ground motion (see Section 3.3 and Figure 9). An assumption is, for example, 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). This will absorb and dissipate energy and damp the response. ISRS obtained by reducing the 5-percent damping ISRS in this manner, herein called 10-percent damping ISRS, are shown in Figure 12 and Figure 13.

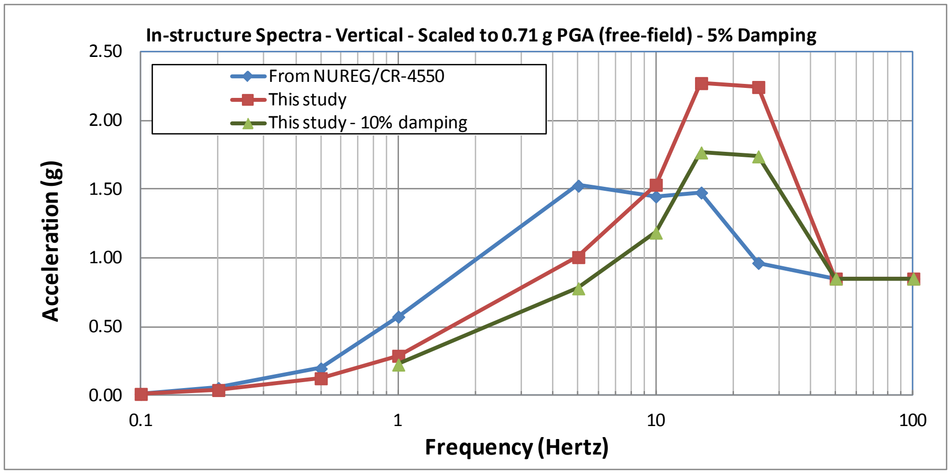


Figure 12 Vertical ISRS for 5-percent and 10-percent damping at Elevation 195 ft (bottom of the SFP)

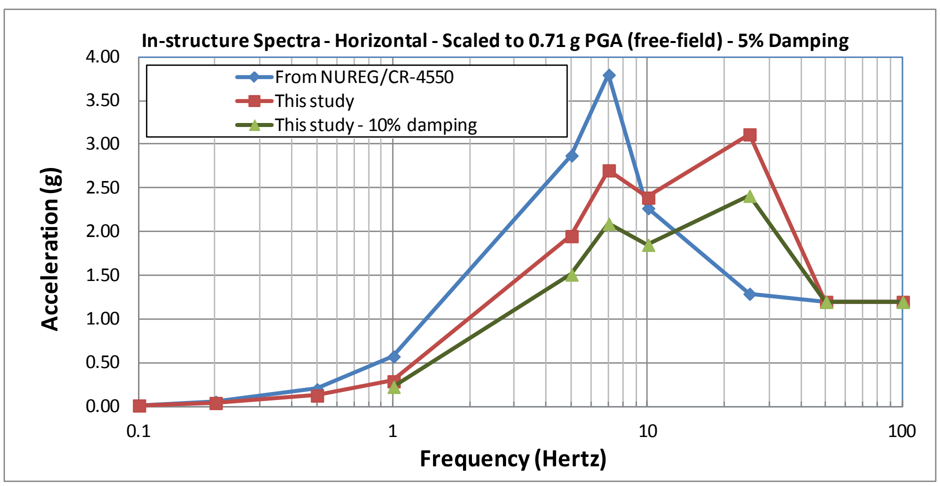


Figure 13 Horizontal ISRS for 5-percent and 10-percent damping midway between Elevation 195 ft and Elevation 234 ft (midheight of the SFP)

The scaling used to obtain the 5-percent damping ISRS does not take into account reductions on spectral accelerations for frequencies greater than 10 Hz that 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 (RC) 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 14). The foundation slab above this rock pedestal is still an 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 arguable. 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 on 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.

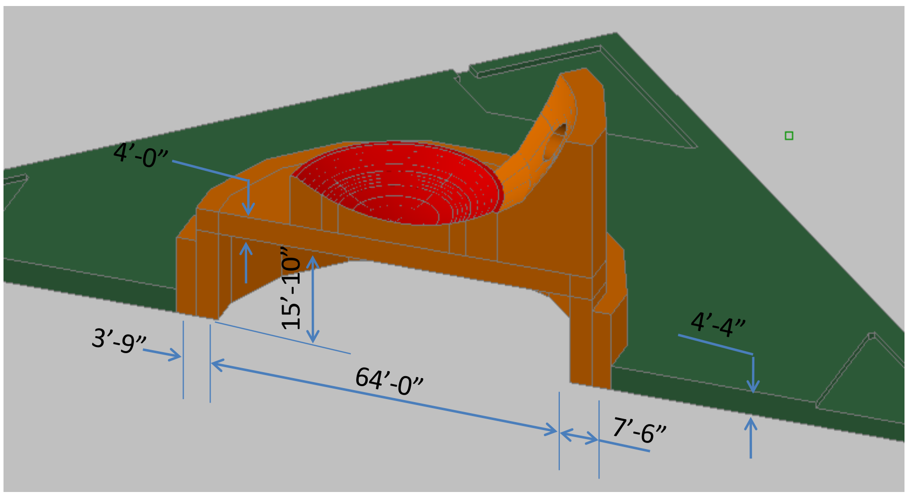


Figure 14 Schematic diagram of the reactor building foundation near the drywell

4.1.2 Description of the Spent Fuel Pool Structure

This section provides a brief description of the SFP structure and its relation to the main reactor building. The description identifies the main structural components and other aspects relevant for this study.

The final safety analysis report (FSAR) for PBAPS 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 at the ends by RC exterior walls on opposite sides of the reactor building. Figure 15 is a 3D representation of the SFP structure and dryer-separator storage pool. Figure 16 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 on the left-hand-side of that figure 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 17) 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 ft deep (above the top of the SFP floor) and about 6 ft 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, spent fuel racks, and the partition wall (south, S, wall). The E and W walls are supported by the thick RC biological shield building on the north (N) side and by the outer wall of the building (on the south side). A cavity exists between the SFP itself and the outer wall of the reactor building.

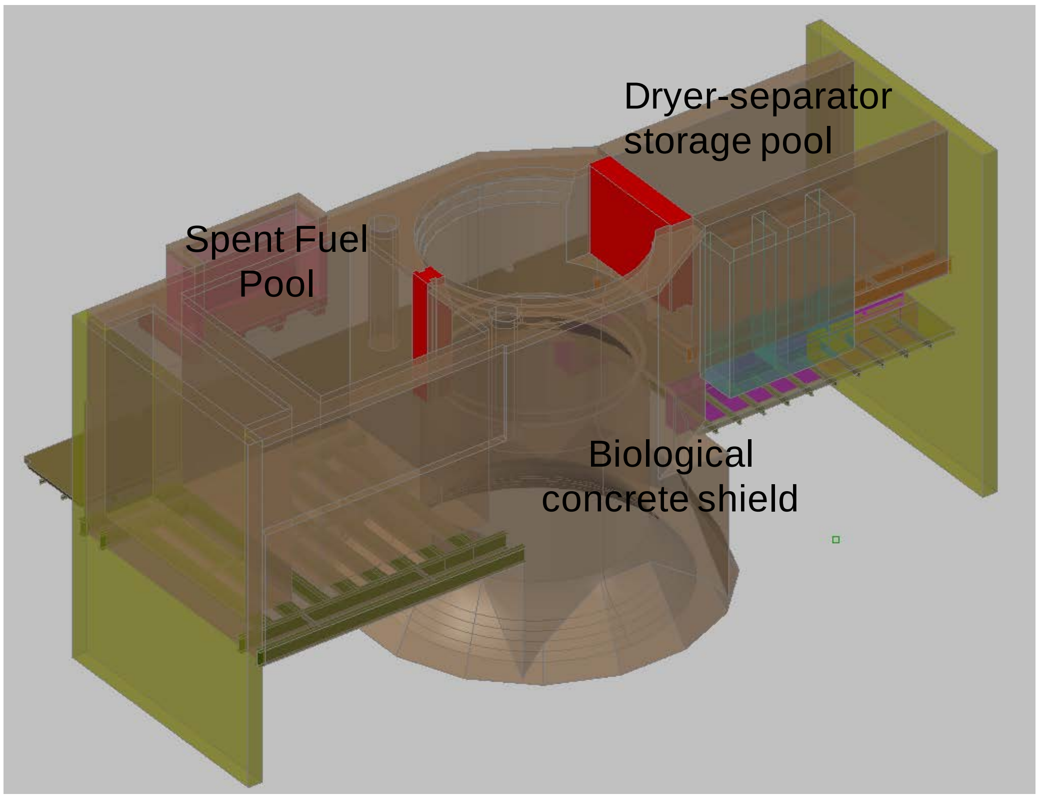


Figure 15 SFP details in cutout of 3D CAD model

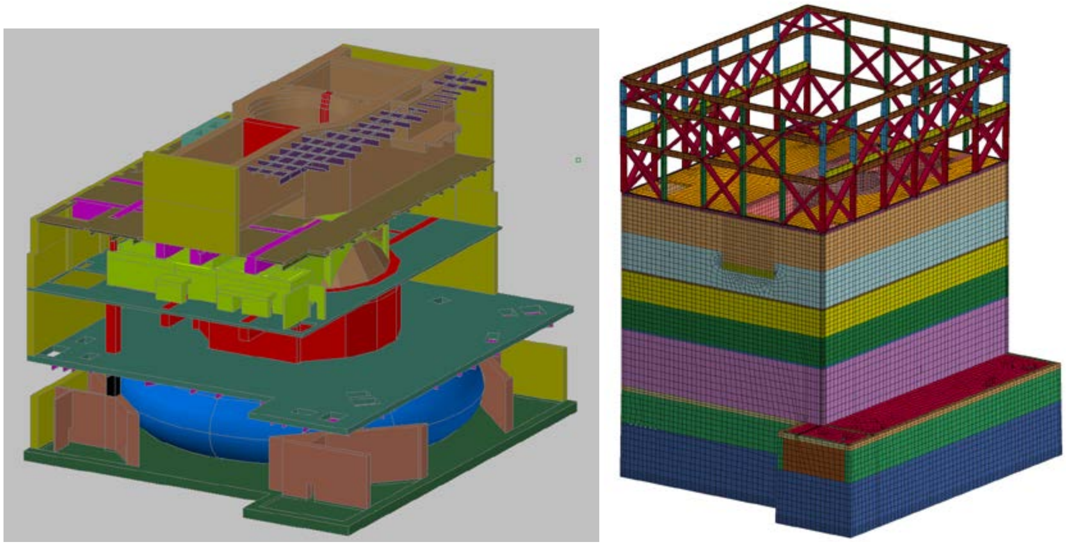


Figure 16 Cutouts of 3D CAD models of the reactor building and SFP

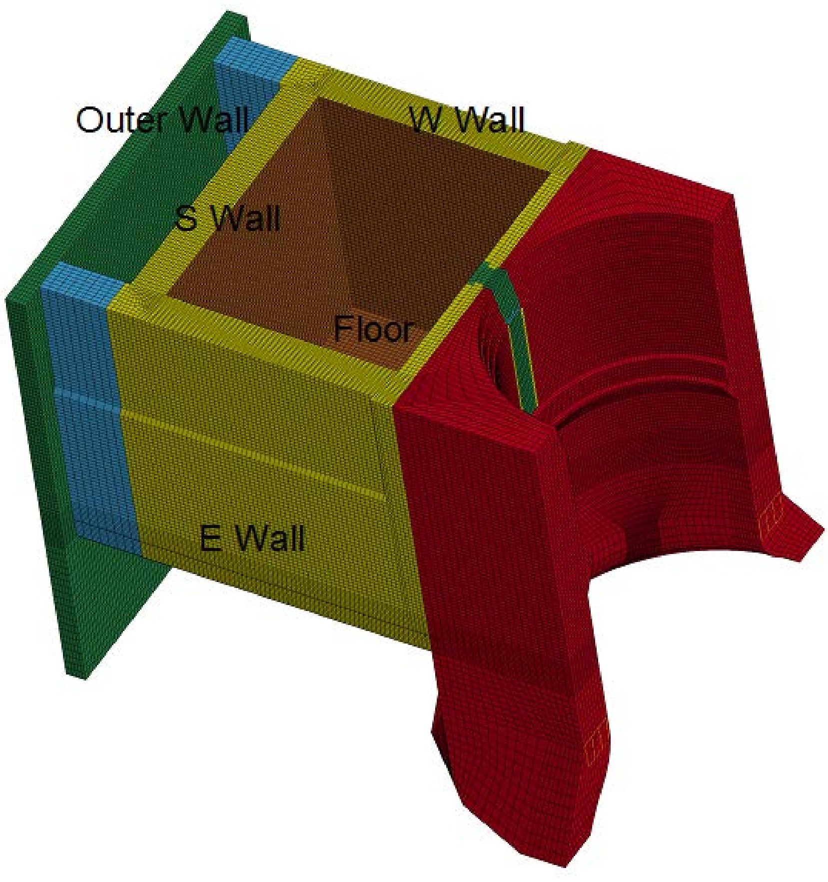
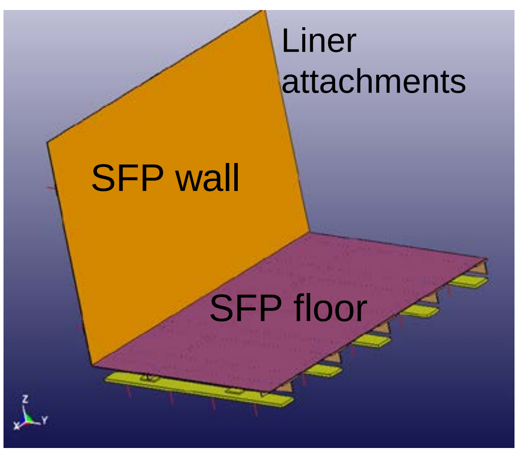


Figure 17 Finite element model of the SFP structure with labels for the floor and walls

Of interest for the study is assessment of damage and cracking to the walls identified in Figure 17 as well as to the floor of the pool from the low probability, seismic event considered in this study. The walls are RC walls with vertical and horizontal layers of #11 (1.41 in. diameter) reinforcing steel bars near each face as well as near the mid surface of the walls.

The SFP floor consists of an RC slab 6 ft 3 in. thick, with embedded heavy steel W-Shapes (I beams) as shown in Figure 15. This floor framing was used during construction and designed to carry the weight of the wet concrete but the beams and decking were left embedded in the concrete floor to the depth of the lower flange of the shapes. The beams that extend from the biological concrete shield to the outer 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. Vertical reinforcement near each face of the wall extends vertically into the floor slab and some of those bars bend and then extend horizontally into the upper half of the pool floor. This is done to provide adequate embedment to the reinforcing bars.

The floor and walls of the SFP are covered with a 1/4-in. thick stainless steel liner which is designed to preclude inadvertent loss of water and that is attached to the concrete using steel anchors, and welds to steel plates and shapes embedded in the concrete floor and side walls. Figure 18 , 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 (E and W walls), is used 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 sump drain.



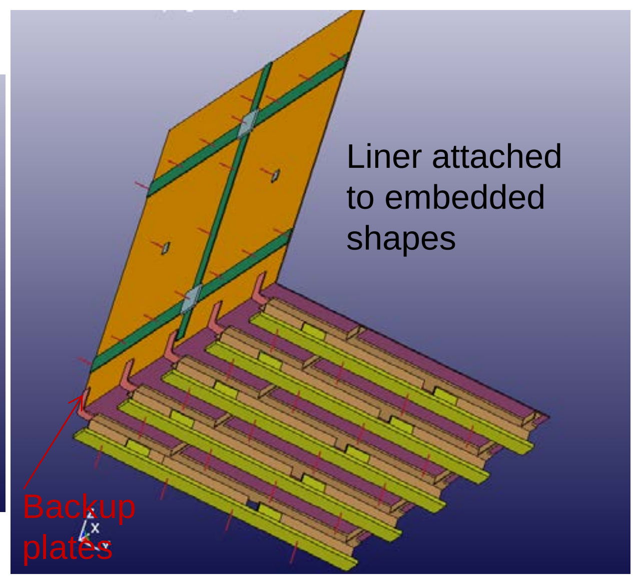


Figure 18 Outline of detailed finite element model of the SFP liner representing attachments to the SFP floor and walls (E and W walls)

According to the FSAR, 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 sentence 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 gate opening (in red in Figure 15) is covered with concrete blocks and closed by two steel gates, in which one steel gate backs the other to provide redundancy in the case of malfunction of a single gate. Each gate consists of steel plates with steel stiffeners. Each gate has polymeric seals around its perimeter that are kept under pressure by the mechanical locking system for the gates. Pressurization of the seals is not a pneumatic system that requires pressurization by electric power systems.

4.1.3 Finite Element Model Description

Step 5 of the approach described in section 4.1.1, the nonlinear pseudodynamic analysis of the SFP 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 LS-DYNA finite element software was used for the analysis (LSTC, 2007). Figure 17 shows the overall detailed finite element model. 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 major reinforcing bars for the floor and walls of the SFP structure as well as 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 LS-DYNA shell elements. In addition, the finite element model also includes the steel liner on the inside surface of the SFP. Figure 19 shows some of the components included in the finite element model.

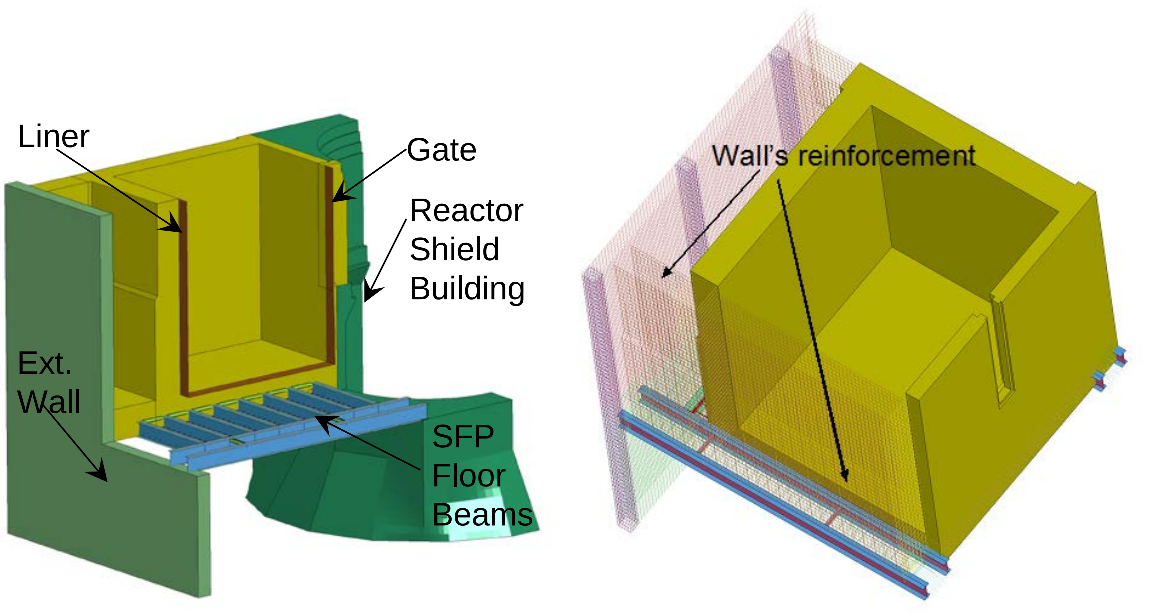


Figure 19 Cutouts of 3D finite element model showing components included in the model

Given the complexity of the structure, rather than using node-to-node modeling for the embedded shell elements modeling the steel beams, the model used the "Constrained

Lagrange in Solid" option available in LS-DYNA to represent the coupling between the embedded elements and the concrete. For the steel liner, two levels of modeling detail 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 bonded to the concrete (node-to-node connections). A more detailed model of sections of the liner (see Figure 18 above) with elements as small as 3.7 millimeters (mm) (0.15 in.) wide was subsequently used as an embedded gage to assess strain concentrations in the liner plates at the intersection of the floor and walls as discussed in the following section.

Boundary conditions for the nonlinear finite element analysis are as follows: (i) vertical and horizontal displacements fixed at the bottom of the exterior wall and of the radiological concrete shield building; (ii) horizontal displacements fixed at the edges of the exterior wall and at the edges of the radiological concrete shield building; and (iii) horizontal displacements in the direction perpendicular to the E and W walls fixed at the top of the E and W walls from the exterior wall to the radiological concrete shield building. Fixing the horizontal displacements at the top of the E and W wall in the direction perpendicular to the walls is justified on the basis of the 1 ft 7 in. thick composite floors with a reinforced concrete deck continuous with the SFP walls on each side of the SFP at the top of the E and W walls (Elevation 234 ft) and that extend to the exterior walls.

The finite elements in the model for the nonlinear analysis are as follows:

- reinforcing bars—LS-DYNA beam elements with the truss option.
- concrete—Constant stress LS-DYNA solid elements (reduced integration)
- shell elements—Belytschko-Tsay shell elements.

Two material models were used as follows:

- Concrete—LS-DYNA material model 159 known as 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—LS-DYNA 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 be the materials that would most influence 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, 2011) and a nominal concrete strength of 4,000 pounds per square inch (psi) (27.5 MPa). For the other materials, the table primarily lists nominal properties. In the case of the liner, nominal material properties were assumed for its yield strength and Young's modulus. These properties and the liner itself are not expected to have a significant effect in the overall response of the SFP structure. However, liner strains and failure strains for the liner are critical in assessing the leakage potential for the SFP. An approach to assess failure of steel liners in reinforced concrete containments is used together with simple probabilistic models to estimate the relative likelihood of the damage states as described in Section 4.1.5.

Table 7 Material Properties for the Nonlinear Finite Element Analyses

Material	Properties	
Concrete	Unconfined compressive strength	6,400 psi (44.6 MPa)
	Aggregate diameter	1.5 in. (38 mm)
	Unit weight (and density)	145 lb/ft^3 (2.33 g/cm^3)
	Young's Modulus (for reference)	4.5x10^6 psi (31,000 MPa)
Rebars	Yield strength (Grade 40)	47,850 psi (330 MPa)
	Yield strength (Grade 60)	69,000 psi (475 MPa)
	Young's modulus	31x10^6 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)	36,000 psi (250 MPa)
	Young's modulus	30x10^6 psi (2.07x10^5 MPa)
	Tangent modulus	15x10^4 psi (1,000 MPa)
	Unit weight (and density)	479 lb/ft^3 (7.7 g/cm^3)
	Failure strain	Treated as variable
Beams	Yield strength	36,000 psi (250 MPa)
	Young's modulus	30x10^6 psi (2.07x10^5 MPa)
	Tangent modulus	25x10^4 psi (1,700 MPa)
	Unit weight (and density)	479 lb/ft^3 (7.7 g/cm^3)
	Failure strain	0.10
Anchor studs	Yield strength	36,000 psi (250 MPa)
	Young's modulus	30x10^6 psi (2.07x10^5 MPa)
	Tangent modulus	25x10^4 psi (1,700 MPa)
	Unit weight (and density)	479 lb/ft^3 (7.7 g/cm^3)
	Failure strain	0.10

This study used a simpler version of the model used for the nonlinear analysis. This 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 (ANSYS, 2011). 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 it 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 elements 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: (1) Young's modulus of 3.15x10^{6}psi (reduced to 70-percent of the Young's modulus of reference to account partially for cracking effects on stiffness) (21,700 MPa), (2) unit weight of 145 lb/ft^3 (2.33 g/cm^3), and (3) a Poisson ratio of 0.15.
- Water: (1) bulk modulus of 3.16x10^5 psi (2,180 MPa), (2) unit weight of 62.4 lb/ft^3 (1 g/cm^3), and (3) a viscosity of 1.64x10^{-7} psi-s (1.13x10^{-9} MPa-s).

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 were done to estimate seismic load coefficients for structural components and to verify the magnitude of the hydrodynamic pressures on the SFP walls.

Summary of Dead and Seismic Loads for the Finite Element Analysis

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 on the 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. Horizontal hydrodynamic loads (not shown in Table 9) considered hydrodynamic pressures from the horizontal ground motions as well as pressures on the wall from the vertical ground motions.

Table 8 Approximate Dead Loads on the SFP Floor in Terms of an Equivalent Vertical Floor Pressure

Load	Approximate equivalent floor pressure in lb/ft^2 (in kPa in the parentheses)
Weight of the floor	900 (43)
Vertical hydrostatic pressure	2,300 (110)
Weight of spent fuel assemblies and racks	1,700 (80)
Total	4,900 (230)

Table 9 Approximate Peak Equivalent Seismic Loads in Terms of an Equivalent Static Vertical Floor Pressure

Load	Approximate equivalent floor pressure in lb/ft^2 (in kPa in the parentheses)
Floor slab acceleration	1,400 (67)
Hydrodynamic impulsive vertical pressure	4,840 (230)
Dynamic forces from spent fuel assemblies and racks	1,750 (85)
Total	7,990 (385)

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 the 5-percent vertical and horizontal ISRS described in Section 4.1.2. It is noted that the (lower) natural frequencies of the SFP, considering a reduction of the concrete Young's modulus to about 70-percent of its original value, the water, and the mass of the spent fuel assemblies and racks, range from about 14 Hz (vertical motion of the floor) to 24 Hz (horizontal motion of the walls). 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 for this study shown in Chapter 3 of this report indicates 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 20 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 are for a free-field PGA of 1.0g and 5-percent damping ISRS. They were multiplied by 0.71 and by the ratio of spectral amplitudes for 10-percent and 5-percent damping 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 4-ftx4-ft 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 LS-DYNA 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. Given the significance of these pressures, deterministic response spectrum analysis with the simplified ANSYS finite element model of the SFP was used in their calculation. Figure 21 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 21 are for a free-field PGA of 1.0g and the 5-percent damping ISRS. They were multiplied by 0.71 for the PGA of interest and by the ratio of spectral amplitudes for 10-percent and 5-percent damping to obtain the values shown in Table 9. Note that water pressures from the vertical accelerations also apply hydrodynamic pressures to the walls, which decrease with height above the floor. The analysis accounted for these pressures.

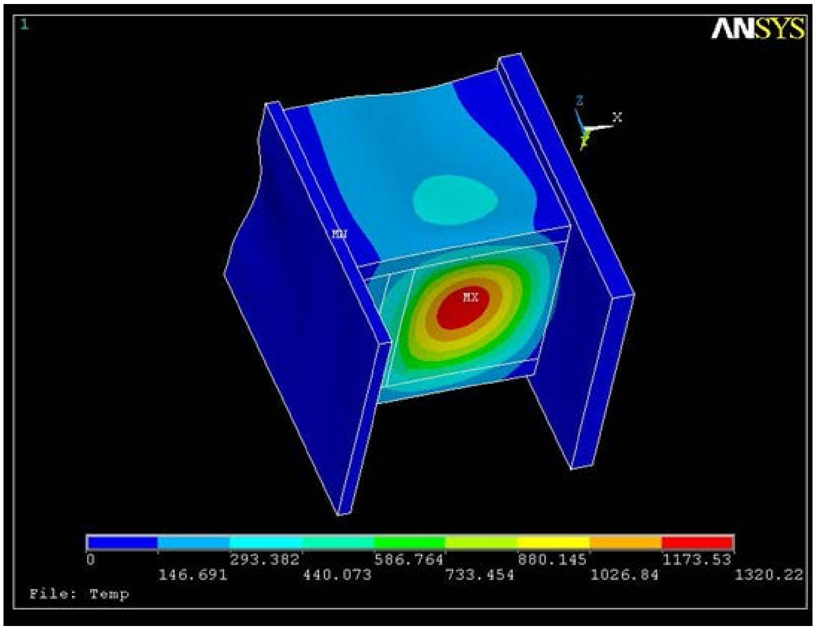


Figure 20 Estimated peak vertical accelerations (in/sec^2) of the SFP floor from response spectrum analysis and vertical ISRS as input (1.0g PGA and 5-percent damping)

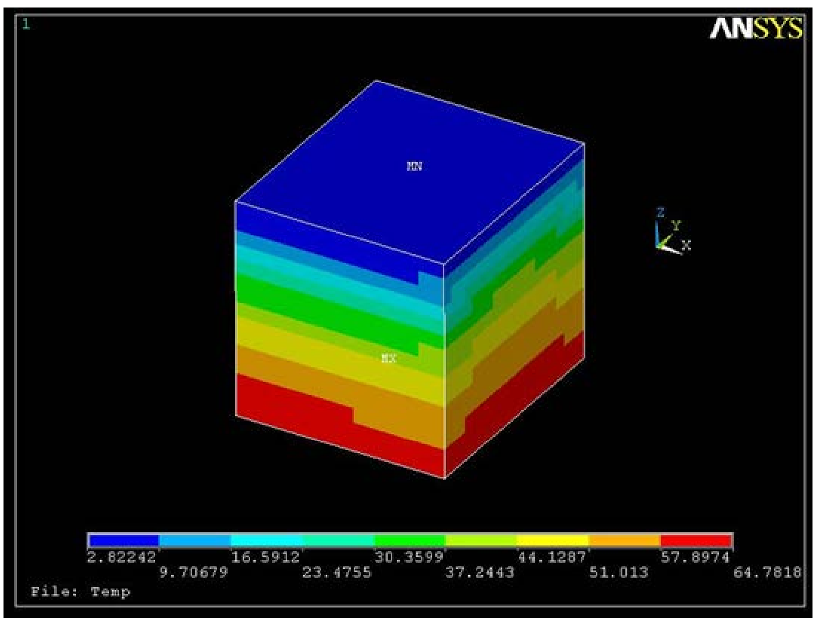


Figure 21 Estimated peak hydrodynamic pressures (psi) on the SFP floor from response spectrum analysis and vertical ISRS as input (1.0g PGA and 5-percent damping)

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 estimated in Section 4.1.3. 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 LS-DYNA software which is an explicit dynamic finite element code. Since this is an equivalent static analysis, the analysis used mass scaling (with only minor 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 they reached their full values. Subsequently, the analysis slowly and proportionally applied all the equivalent seismic static loads until they 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.

Figure 22 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 are small on the order of 0.6 in. (15 mm) or about 0.6/(40x12) = 1/800 of the clear span. Small displacements are a result of the high stiffness of the SFP structure which consists of thick RC slabs and walls (on the order of 6 ft) and comparatively short spans (from about 35 ft in the N-S direction and about 40 ft in the E-W direction).

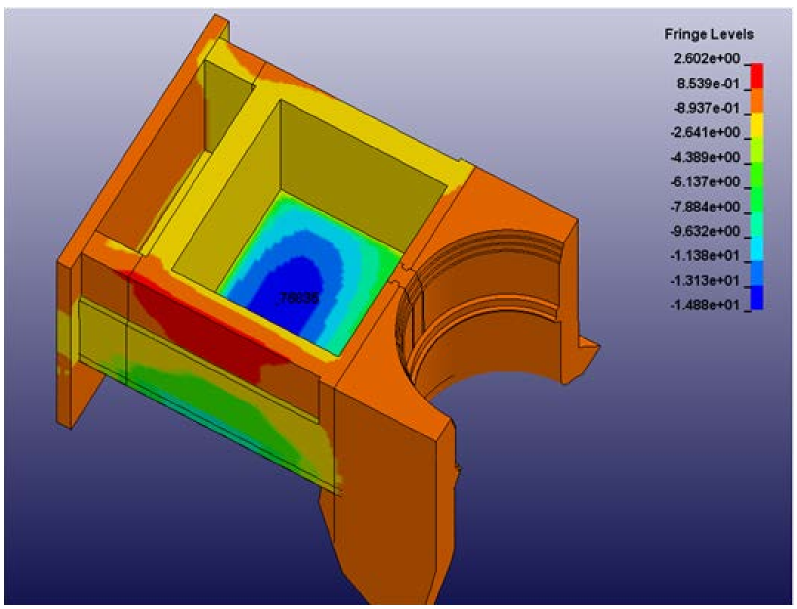


Figure 22 Contours of vertical displacements (mm) of the SFP floor and walls

Figure 23 shows vertical displacement along the outside face of the W wall. Of special interest in Figure 23 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 this is the region of possible strain concentrations in the SFP liner as shown in Figure 24. Finally, Figure 25 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 could likely develop. The crack would start as a flexure crack and develop into a mostly tension-flexure crack though the thickness of the wall accompanied by shear friction at the bottom of the wall.

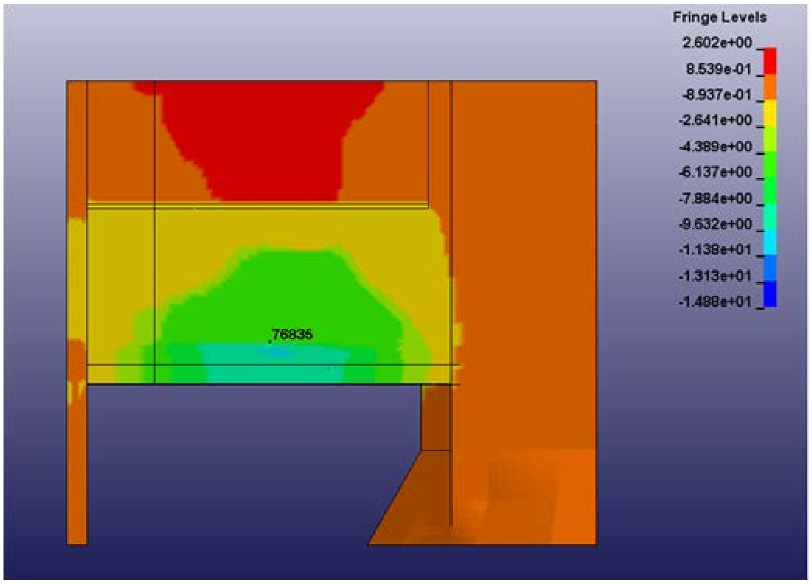


Figure 23 Contours of vertical displacement (mm) of the SFP walls

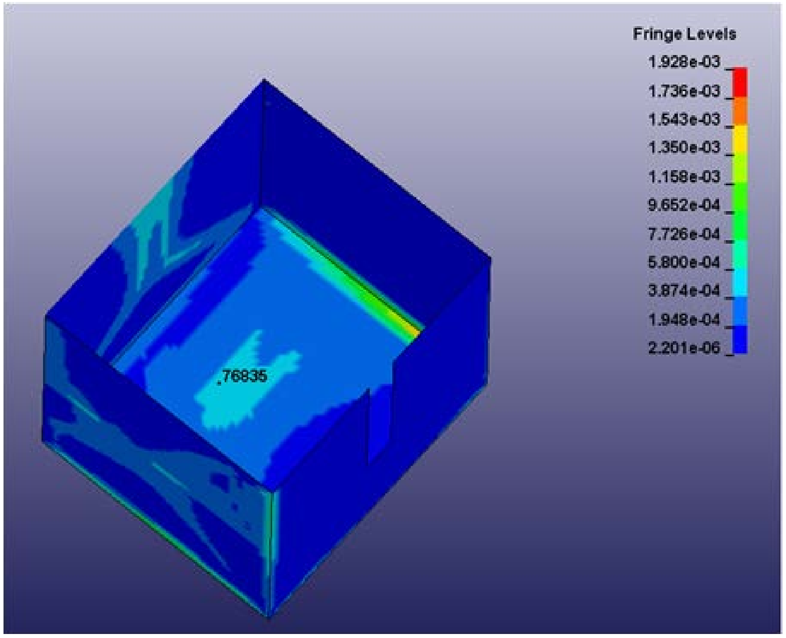


Figure 24 Liner strains (overall response not fully accounting for strain concentrations)

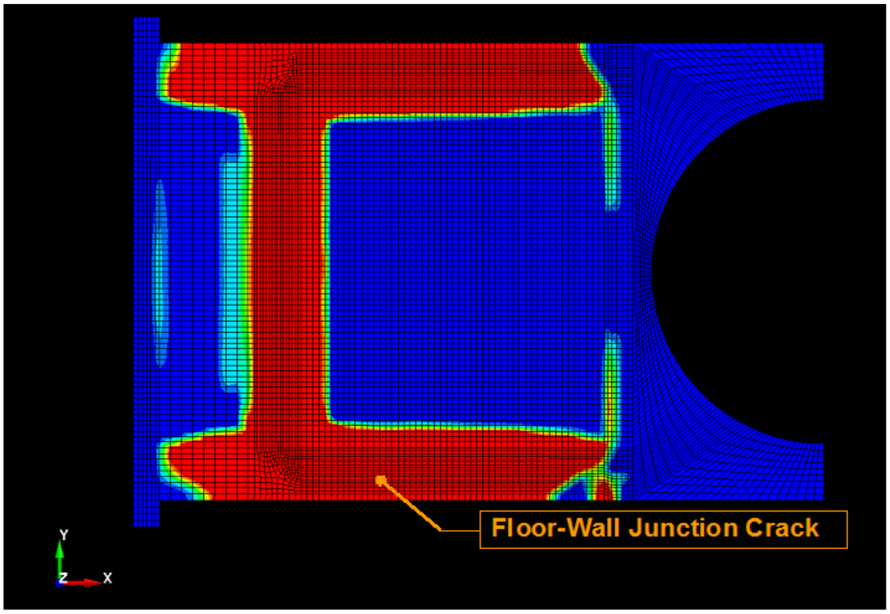


Figure 25 Region of concrete cracking initiation at the floor-wall junction

The higher liner strains in Figure 24 are, as expected, at the intersection of the SFP wall with the SFP floor, which is the region of strain concentrations. Although this is a region of strain concentrations, the liner strains shown are small, of the order of $5x10^{-4}$ to $1.9x10^{-3}$. For comparison, the nominal liner yield strain is $1.2x10^{-3}$. 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 details of 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 18. 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 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 LS-DYNA and appropriate contact definitions) to assess strain concentrations in the liner. Note that the actual liner and the liner in the model are attached to the concrete only at a few discrete locations. Elsewhere, the liner is only in contact with the concrete. 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 18) 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 26 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 does not affect the overall response of the SFP in a significant manner. However, it permits an estimation of the strain concentrations in the liner.

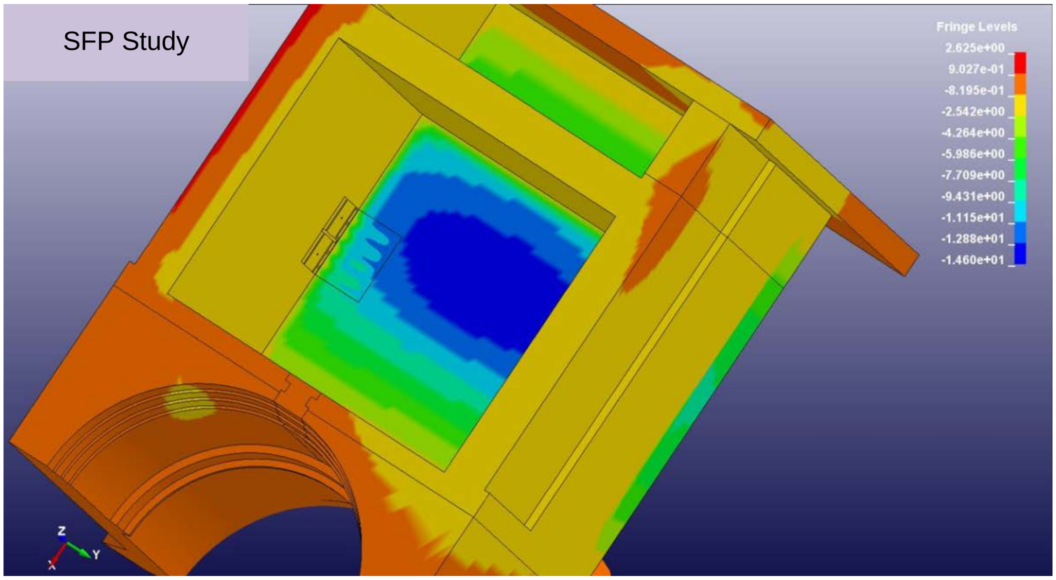


Figure 26 SFP displacements (mm) with detailed liner insert

Figure 27 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, i.e., 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 27 is about 3.7 percent (0.037). The following section uses these strains as well as estimates of the width and extent of the concrete cracking (see Figure 25), to assess liner tearing likelihoods for the scenario considered.

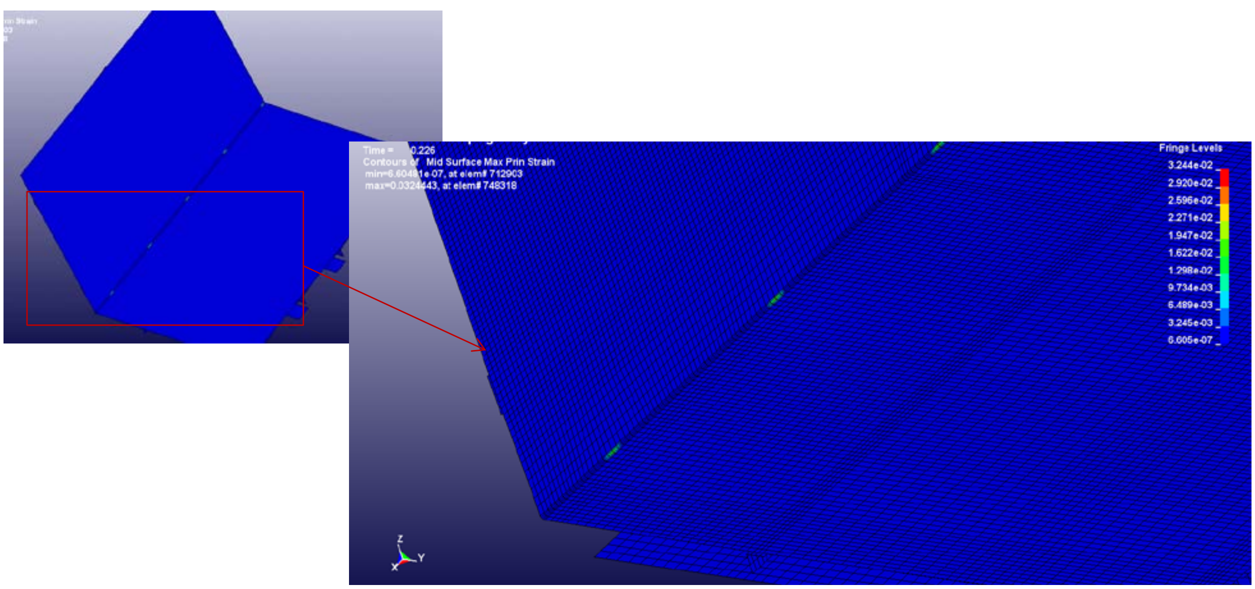


Figure 27 Strain concentrations in the SFP liner

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 4.1.4 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. The section starts with the approach used to estimate the likelihood for each damage state for the initiating seismic event considered. Then, the section provides the estimation of the leakage rates for the damage states with leakage.

<u>Damage States and Relative Likelihoods</u>

Step 6 of the approach (Section 4.1.1) defined three initial damage states as follows:

a. <u>No leakage:</u> 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 previous subsections) but without tearing of the liner.

b. <u>Moderate leakage rate:</u> 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.

c. <u>Small leakage rate:</u> 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.

This study uses an approach and strain criteria, including uncertainties, for tearing of steel liners in reinforced concrete containments (Cherry, 2001 and 1996) together with uncertainties in the calculated liner strains to estimate the relative likelihoods for the three initial damage states listed. Uncertainties in the calculated liner strains account for (1) uncertainties in the ISRS spectral accelerations (of the order of 25 percent), (2) uncertainties in liner strains from uncertainties in concrete properties (namely concrete strength) and (3) an additional reduction in spectral accelerations to account for both ground motion incoherency and nonlinear effects.

The analysis used information in Cherry (1996) to estimate upper and lower bounds for the limiting failure strain, which were then used with a triangular probability density function to estimate their mean and coefficient of variation (Ang, 1984). This is expected to be a conservative assumption in that the SFP liner is of stainless steel which is likely to have larger limiting failure strains. This approach, adjusts the failure strain from coupon tests using reduction factors that account for the multi-dimensional state of stress (triaxiality effects), uncertainties in material properties, and the level of detail in the analysis used to estimate strain concentrations. Bounds in the liner limiting failure strain use a failure strain from coupon tests of 21-percent (0.21) together with a triaxiality factor of 1.75 (typical of a cylindrical state of stress). Estimation of these bounds considers a high level of detail in the model for the calculation of strain concentrations which used elements as small as 0.15 in (3.7 mm) wide. Accordingly, this study considered a range of reduction factors for the analysis detail that range from 0.4 to 0.9. Reduction factors for material properties were those reported in Cherry (1996). On these bases, the bounds on the failure strain for the purposes of estimating its mean and coefficient of variation came to be 0.045 and 0.14, and the resulting mean and coefficient of variation came out to be 0.09 and 0.20, respectively.

Maximum effective tensile strains in the liner were calculated assuming reduced material properties and were found to be sensitive to the concrete strength. Effective strains calculated for the median concrete strength and a reduced concrete strength were used to assess the derivative of this strain to the concrete strength. This derivative was then multiplied by the standard deviation of the concrete strength which was calculated using an estimated coefficient of variation for the concrete strength of 0.15 (Lambright et al., 1990) to estimate the standard deviation and coefficient of variation of the liner strain associated with uncertainties in the concrete strength. This coefficient of variation was estimated to be about 0.65.

Maximum effective concrete strains calculated for the 0.7 g PGA and for reduced seismic loads (about 70-percent of the initial loads) were used to estimate the sensitivity of the effective strain to the estimated spectral amplitudes. Nonlinear analyses were done to estimate maximum effective strains for spectral accelerations equal to about 80-percent of the original to account for effects of ground motion incoherency and further reductions from nonlinear effects. Additional uncertainty measures for the calculated strain were then estimated using the calculated strains for the base case and the case with reduced spectral amplitudes in conjunction with an asymmetric triangular distribution for the calculated strains. The assumed triangular distribution used the strain for the reduced value as the least likely value and that for the base case as the most likely value. This procedure resulted in an adjustment of the median strain (reduction factor equal to 0.93) and an additional coefficient of variation (0.09) for the liner strain. An additional coefficient of variation for the ultimate strain of about 0.25 was used to account for uncertainties in the estimate of floor response spectra ordinates (Lambright et al., 1989).

Uncertainties calculated in this manner were then used to estimate medians and coefficients of variation for the limiting failure strain (capacity) and for the induced strain (demand). Using these quantities and assuming lognormal distributions, the probability of liner tearing conditional on the occurrence of the seismic event was estimated to be less than 10-percent (Ang, 2006; Ang, 1984). This estimate indicates that the state with no leakage (no tearing of the liner) is the most likely with a relative likelihood in excess of 90-percent. The relative likelihood of the two states with leakage from the bottom of the SFP is estimated at less than 10-percent. Assigning relative likelihoods to the two damage states with leakage is subject to considerable uncertainties at this time. Accordingly, the assumption is made that both states are equally unlikely.

<u>Concrete Cracking and Moderate Leakage Rate</u>

Postprocessing 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. The 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 subtracting 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 of about 3.6 mm (about 0.14 in.) and an average crack length of about 33,000 mm (about 108 ft), with a non-smooth and non-uniform surface. An average crack width is used because the overall change in the crack width is not expected to be large along the perimeter of the floor.

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 envelope 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 testing are (1) an equation to estimate the leakage flow rate through concrete cracks that involves a friction factor and (2) quantification of that friction factor based on the experimental data. 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 was calculated as shown in Figure 28 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. The average flow rate for this condition to a height of about 16 ft above the SFP floor is about 1,500 gallons per minute.

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 that the liner tearing spreads to an extent such that the leakage rate through the liner is greater than the leakage rate through the concrete cracks. In this case, concrete cracking 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.

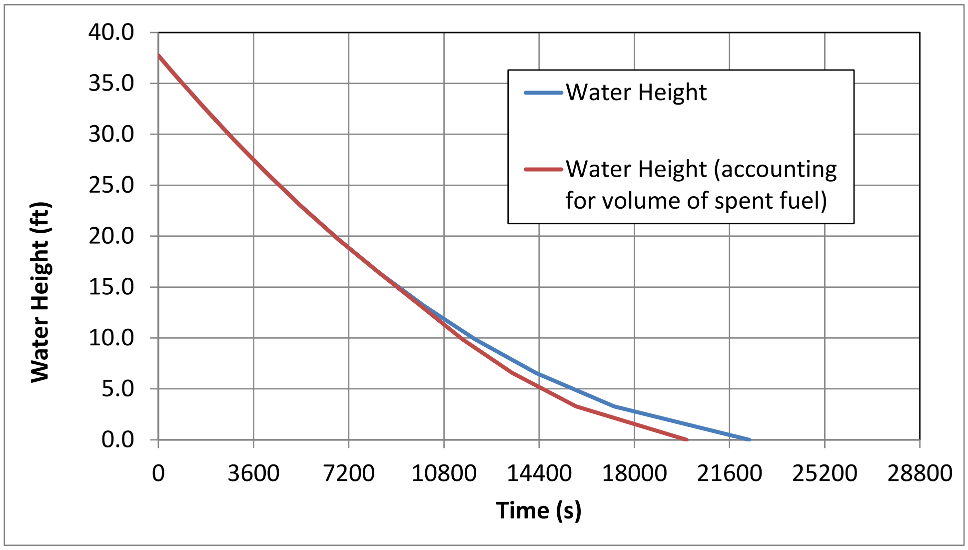


Figure 28 Moderate leakage flow rate (through concrete cracks)

<u>Liner Strains and Small Leakage Rates</u>

Maximum effective membrane liner strains from strain concentrations at the floor-walls junction are on the order of 0.037 (3.7 percent). These strains are localized at the backup plates, which are spaced 24 in (609.6 mm) 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 are smaller. Moreover, liner tearing or through wall (or floor) concrete cracking are not expected near the biological concrete shielding. Accordingly, tearing of the liner, if it were to occur, would be only along the base of the E and W walls.

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 0.15 in (3.7 mm) 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.

The resulting crack width for a liner tear localized at the location of the backup bar is then estimated at 0.15x0.10 = 0.015 in (0.37 mm). 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 24 in (609.6 mm), a total of 40 backup bars (20 on each wall) are used to estimate the summed length of all localized cracks as 40x4 = 160 in (4,064 mm). The estimated width for each crack, if it were to occur, is then 0.015 in (0.37 mm) and the depth of the crack is the depth of the liner which is equal to 0.25 in (6.35 mm).

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%. 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.

Estimation of a friction factor was made using data in Paul et al. (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 temperatures and the driving pressures are much higher than those applicable to the SFP. Review of other flow models reported in Paul et al. (1994) indicates that for relatively smooth cracks friction will be low. Assuming relatively smooth cracks, the equation for the flow through concrete cracks was applied for flow through steel tears together with a small friction factor (0.11) in order to estimate the leakage flow rates. For this friction factor, the estimated leakage flow through the steel cracks (small leakage flow) is as shown in Figure 29. Considerable uncertainty continues to exist in the estimation of leakage flow rates for these localized liner tears. Given the assumption that the crack surface is relatively smooth, it is estimated that the flow rates in Figure 29 would be greater than the actual flow rates. The average flow rate for this condition to a height of about 16 ft (488 mm) above the SFP floor is about 200 gallons per minute (757 liters per minute).

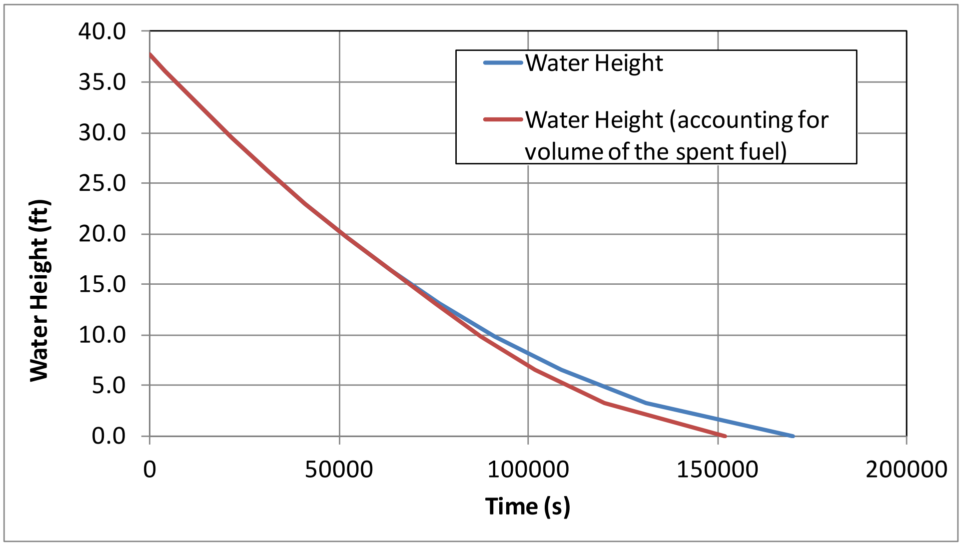


Figure 29 Small leakage flow rate (through localized steel tears)

4.2 Other Damage States

Assessment of other damage stages is primarily based on (1) finite element deterministic response spectra analysis to estimate maximum vertical displacements of the water surface (sloshing), (2) seismic fragilities used in conjunction with the NUREG-1150 seismic PRA study (Lambright et al., 1990), (3) the examination of design details for certain appurtenances such as the refueling gate, and (4) 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. Finite element analysis using the ANSYS finite element model described above, show that the natural frequencies of the sloshing modes in the two horizontal directions parallel to the walls of the SFP are about 0.27 Hz and 0.29 Hz, corresponding to periods of vibration on the order of about 3.8 to 3.5 seconds. These results resemble those obtained using analytical methods (e.g., AEC, 1963; Malhotra et al., 2000).

The free-field ground motion specified for the study does not have high spectral velocities and accelerations at the sloshing frequencies. Consequently, sloshing amplitudes are expected to be small. Deterministic response spectrum analyses with the simplified ANSYS finite element model of the SFP using the horizontal ISRS at midheight of the SFP (for the frequencies of interest to sloshing) as input and considering the low damping of the sloshing mode, show that the sloshing amplitude will not exceed about 20 in. Given that the water at the pool is about 1 ft below the top of the SFP, sloshing is not expected to cause more than 1 ft of water loss. Accordingly, an initial 1.5 ft 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 or dc 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, the study assumes that the refueling gate will not fail for the seismic event considered and will continue to maintain its intended function during the accident progression.

SFP penetrations: According to the FSAR, 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. Results of the nonlinear finite element analysis also indicate that overall distortions of the pool walls are small (on the order of a few millimeters). These distortions are not expected to lead to seismically induced damage of the penetrations that would lead 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 study assumes 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 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. In addition, the high-frequency components (greater than 10 Hz) of the motion would not be expected to induce large sliding or rocking motions.

Damage to the Reactor Building and Other Relevant SSCs

According to the fragility analysis for the NUREG-1150 seismic PRA (Lambright et al., 1990), the median fragility for the reactor building is about 1.6g. 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 for horizontal modes of vibration of the reactor building 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 with the same PGA considered in earlier evaluations of the median fragility). On these bases, seismically-induced failure or severe damage to 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 ends as a consequence of the ground shaking, that end of the crane bridge would not fall inside the SFP. Depending on the end of the crane bridge losing support, the crane could fall only a few feet from the SFP, but not inside the SFP.

A LOOP 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 study (Lambright et al., 1990) indicates a high probability of loss of onsite ac power (about 0.84). This estimate is based on either direct failure of the onsite emergency diesel generators (assumed to be sensitive to spectral accelerations around the 20 Hz frequency) or failure of either the emergency service water or the emergency cooling water systems that provide cooling water for the diesel generators. The probability of losing dc power based on the fragility of the inverters alone is estimated to be close to but less than 50-percent for the seismic event considered in this study.

4.3 Review of Spent Fuel Pool Performance under Recent Major Earthquakes in Japan

Five Japanese nuclear power plant sites with a combined total of 20 reactors and 20 SFPs were subjected to severe ground motions from two major earthquakes in the past 5 years (NERH, 2011a; NERH, 2011b; Kawamura, 2008; Sato, 2010):

- March 11, 2011, Tohoku earthquake (with moment magnitude M_w= 9.0)
  - Fukushima Daiichi (5 BWR Mark I and 1 BWR Mark II SFPs)
  - Onagawa (3 BWR SFPs)
  - Fukushima Daiini (4 BWR SFPs)
  - Tokai (1 BWR SFP)
- July 16, 2007, Niigataken Chuetsu-Oki earthquake (M_w= 6.6)
  - Kashiwazaki-Kariwa (7 BWR SFPs)

This review addresses reductions in water levels for the SFPs affected by those events that might have resulted from either water leakage from structural damage or water loss from sloshing, if any.

No leakage of water near the bottom of the SFPs has been reported for any of the 20 SFPs in those five nuclear power plants for these two major earthquakes. For the Kashiwazaki-Kariwa site, the only report of water loss (leakage or sloshing) for the seven SFPs at the site was a loss of about 320 gallons (about 1.2 cubic meters) from sloshing of the water in the SFP of Unit 6 (Kawamura, 2008).

Loss of water other than from sloshing was not reported for the SFPs of the power plants affected by the March 11, 2011 Tohoku earthquake (NERH, 2011b). According to the NERH (2011b) report, minor leaks of radioactive material (all contained inside buildings) at the Onagawa plant were attributed to sloshing of SFP water, and SFP sloshing overflow lead to a 8 in (20 cm) decrease of the water level in the SFP at Tokai. Actual decreases in SFP water levels from sloshing at the Fukushima Daiichi units are not known, but decreases in water level from sloshing have been assumed in evaluations of SFP performance (NERH, 2011b). Specifically, a water level reduction of about 1.6 ft (0.5 m) was assumed for Unit 2 as a result of sloshing induced by the ground motion while reductions of about 5 ft (1.5 m) were assumed for Units 1, 3 and 4 from sloshing associated with ground motions and explosions.

This review also 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 this review and comparison use information available at the time of the execution of this study, they assist in the interpretation of the results obtained for the seismic scenario and SFP 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 Study. A possible exception to this would be 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). Differences in the seismic design-basis loads and uncertainties regarding the construction details (e.g., out of plane shear reinforcement if any) for the various SFPs listed above add to the overall level of uncertainty in the comparisons. However, this section provides a comparison of the structure of the SFP considered in this study and the structure for the SFP of Fukushima Daiichi Unit 4, for which some structural information was available at the time of the writing of this report.

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, are not directly comparable to the free-field PGA considered in the study. However, the free-field ground motion for this study is also taken to be the foundation ground motion because the reactor building is considered to be a fixed-base structure. Additional sources of uncertainty are the type of reactor (several of the plants have Mark II containments instead of Mark I containments), site conditions (soil versus rock sites), reactor building foundation (slab thickness and uniformity) and reactor building embedment. Generally, the foundation slabs for the reactors listed above are thicker and more uniform than that for the reactor considered in the study. Also, the site for the study is a rock site and stiffer than the sites for Fukushima Daiichi and Kashiwazaki-Kariwa.

An additional source of uncertainty for the comparison is that ISRS reported for some plants may be affected by localized structural details such as the vertical response of a floor slab. Such ISRS would not be representative of the seismic loads on the SFP in the same sense as the ISRS used in this study. Precise determination of the location of the accelerometers used for the observed ISRS was not done for these review and comparison. 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 0.7g. 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 study with the exception of that for Kashiwazaki-Kariwa Unit 1.
- Vertical PGAs at the foundation slabs of all reactors are for the most part less than horizontal PGAs with the exception of Fukushima Daiini Unit 1 and Kashiwazaki-Kariwa Units 6 and 7.

- Vertical PGAs at the foundation slabs of all reactors are less than those considered in the study.
- The difference between the recorded PGAs and the PGA for the study is greater for the vertical accelerations than for the horizontal accelerations.
    - The study assumes that the vertical PGA is approximately equal to the horizontal PGA (see Section 3.3).

Table 10 Fukushima Daiichi, Measured and Design (DBGM S_S) PGAs at Foundation Slab (Tohoku, 2011 Earthquake)

Unit	Containment	Measured (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark I	460	447	258	487	489	412
2	Mark I	348	550	302	441	438	420
3	Mark I	322	507	231	449	441	429
4	Mark I	281	319	200	447	445	422
5	Mark I	311	548	256	452	452	427
6	Mark II	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 (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	BWR	540	587	439	532	529	451
2	BWR	607	461	389	594	572	490
3	BWR	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 (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark II	254	230	305	434	434	512
2	Mark II	243	196	232	428	429	504
3	Mark II	277	216	208	428	430	504
4	Mark II	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 (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark II	214	215	189	393	400	456

Table 14 Kashiwazaki-Kariwa, Measured and Design PGAs at Foundation Slab (Chuetsu-Oki, 2007 Earthquake)

Unit	Reactor	Measured (cm/s^2)			Design Values (cm/s^2)		
		Horizontal		Vertical	Horizontal		Vertical
		NS	EW		NS	EW	
1	Mark II	311	680	408	274	273	235
2	Mark II	304	606	282	167	167	235
3	Mark II	308	384	311	192	193	235
4	Mark II	310	492	337	193	194	235
5	Mark II	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 study is a rock site and dominant seismic event for this scenario would be an earthquake in the CEUS at a distance of about 15 km or less. Accordingly, the ground motion response spectra for the seismic scenario considered has maximum spectral accelerations for frequencies greater than about 10 Hz and at frequencies near the lower fundamental frequencies of the spent fuel pool structures.

Figure 30 includes vertical response spectra for 5-percent damping at the foundation slab of Unit 1 (the case with a horizontal PGA of about 0.7g) and Unit 4 of Kashiwazaki-Kariwa together with the corresponding response spectrum for the vertical ground motion considered for the study. This comparison indicates that the ground motion for this study has higher vertical spectral accelerations near the lower fundamental frequencies of vibration of the SFP structure. Spectral accelerations for the ground motion used in the study remain higher than those for Unit 4 down to a frequency of about 5 Hz and those for Unit 1 down to frequencies of about 4 Hz. The results shown are typical of those for the other reactors at Kashiwazaki-Kariwa (with the possible exception of Unit 6 which has significantly higher spectral accelerations between 6 and 2.5 Hz). The reactors at this plant are Mark II reactors, have reinforced concrete base slabs several times thicker than the reactor considered in this study, and are deeply embedded in the ground.

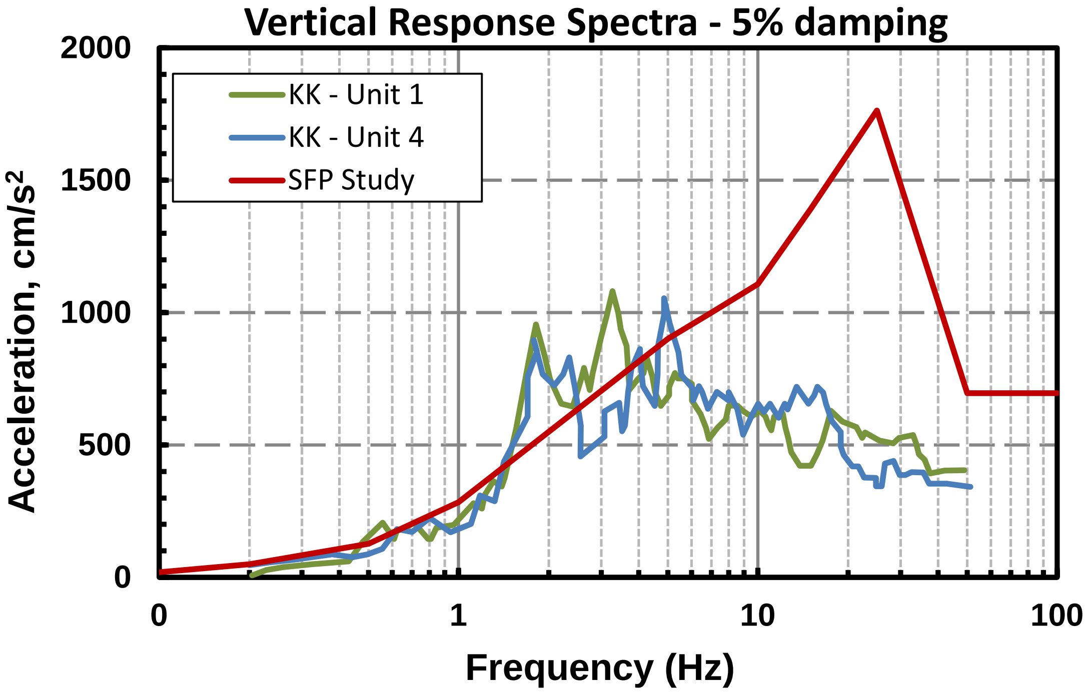


Figure 30 Vertical response spectra: Kashiwazaki-Kariwa Units 1 and 4 (foundation level) and SFP study (free-field)

With the exception of Unit 4 at Fukushima Daiichi, vertical response spectra for the reactors affected by the March 11, 2011 Tohoku earthquake were not available at the time of the study, so the comparison of foundation response spectra are, for the most part, made using horizontal spectra. Figure 31 shows horizontal response spectra for 5-percent damping at the foundation slab of Unit 1 and Unit 4 of Fukushima-Daiichi and the corresponding response spectrum for the horizontal ground motion used in the study. Comparison of those spectra indicates that the ground motion for this study (rock site) has higher horizontal spectral accelerations at the lower natural frequencies of the SFP structure (about 0.05 seconds). Spectral accelerations for the ground motion used in the study remain higher than those for Unit 1 for frequencies down to about 4 Hz and those for Unit 4 for frequencies down to about 3 Hz. The results shown are, in general, typical of those for the other reactors at Fukushima Daiichi.

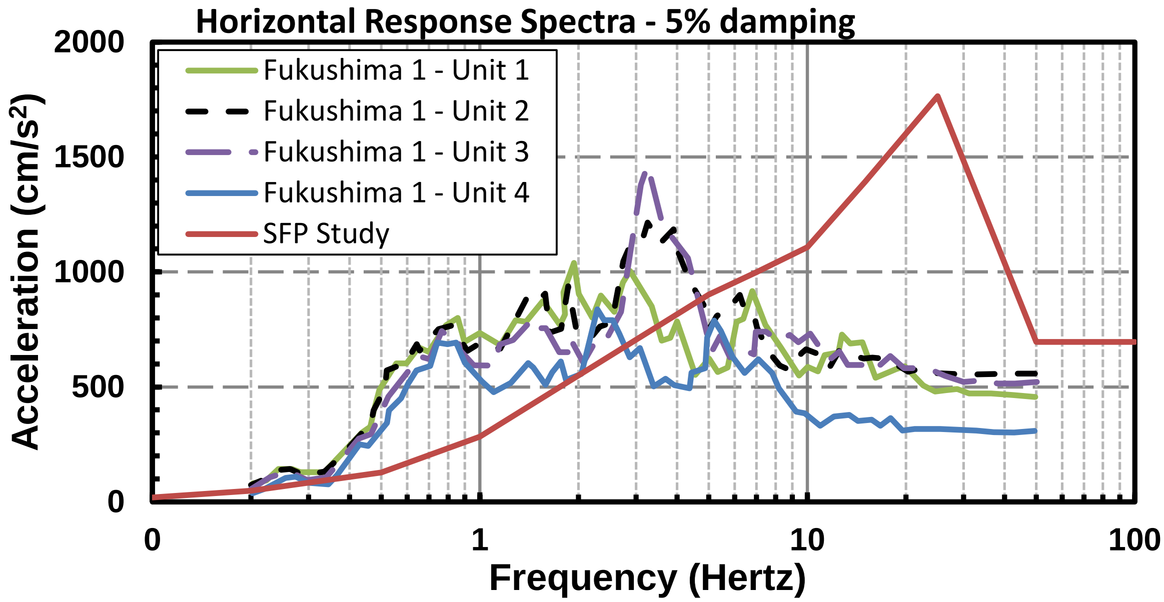


Figure 31 Horizontal response spectra: Fukushima Daiichi Units 1 and 4 (foundation) and SFP study (free-field)

Figure 32 shows vertical response spectra for 5-percent damping at the foundation slab of Unit 4 of Fukushima-Daiichi and the corresponding response spectrum for the vertical ground motion used in the study. Comparison of those spectra indicates that the ground motion for this study (rock site) has higher horizontal spectral accelerations at the lower natural frequencies of the SFP structure (about 0.05 seconds). Spectral accelerations for the ground motion used in the study remain higher than those for Unit 4 for frequencies down to about 3.5 Hz.

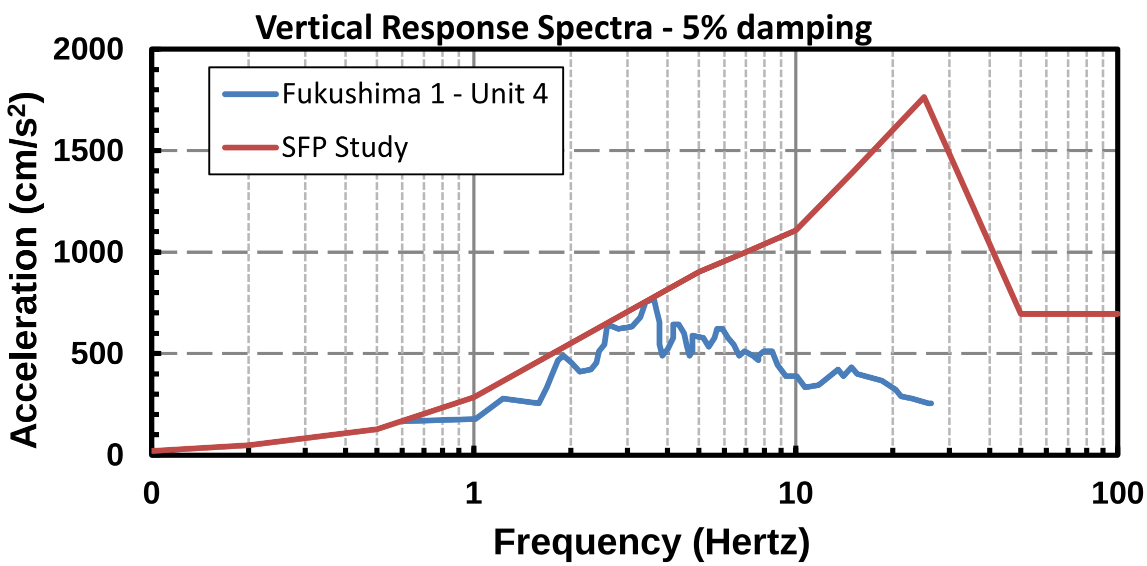


Figure 32 Vertical response spectra: Fukushima Daiichi Unit 4 (foundation) and SFP study (free-field)

Figure 33 shows vertical ISRS at an elevation at about midheight of the SFP for Unit 1 and Unit 4 of Kashiwazaki-Kariwa together with the vertical ISRS for the study. ISRS for the study are shown for 5-percent and 10-percent damping. In both cases, the ISRS for the study is higher than the observed ISRS for frequencies close to the lower natural frequencies of the SFP considered in the study. The 5-percent ISRS for this study remains above that for Unit 4 down to frequencies of about 12 Hz and approximately equal to it for frequencies down to about 7 Hz. The 5-percent ISRS for this study remains higher than that for Unit 1 for frequencies down to about 4 Hz.

The 10-percent damping ISRS for the reactor building approaches that of Unit 4 at frequencies equal to about 17 Hz. The 10-percent ISRS for this study is higher than that for Unit 1 for frequencies down to about 6 Hz and is close to it at about 12 Hz. The ISRS for Unit 1 is typical of those for the other units with the exception of Unit 3, which approaches the 5-percent damping ISRS for the study at about 11 Hz.

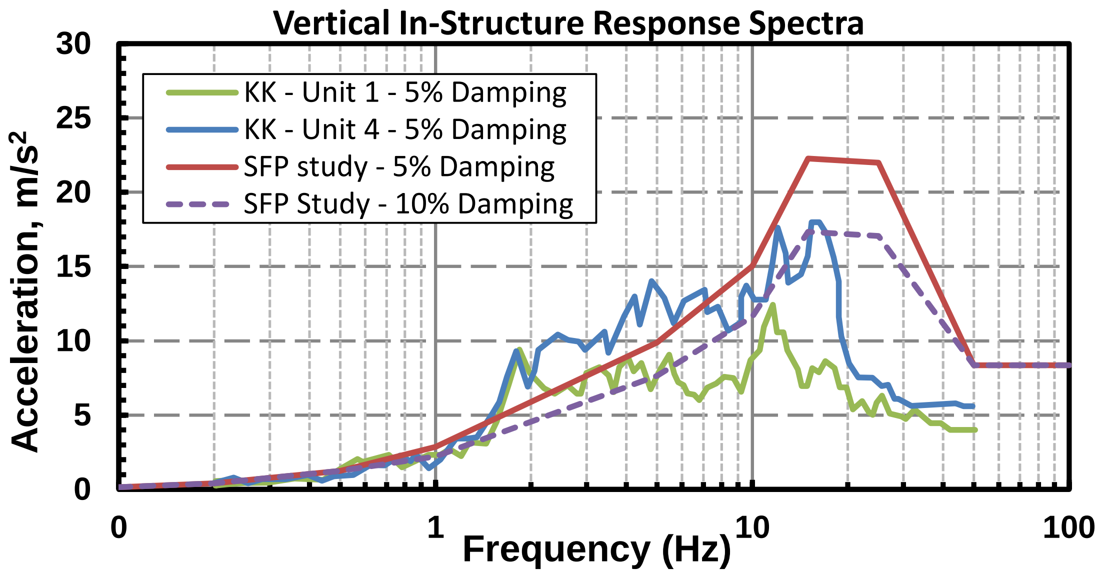


Figure 33 Vertical ISRS for Kashiwazaki-Kariwa Units 1 and 4 and for the SFP Study

The comparisons, especially the comparison of the vertical response spectra at the foundation of Unit 4 of Fukushima Daiichi and at the base of the SFP for the study, indicate that the vibratory loads for this study, especially the vertical loads, are likely to be more challenging to the SFP than those from the actual events.

<u>Structural and Construction Details</u>

The seismic design loads for the various reactors considered in this comparison differ, for the most part, from the design basis loads for the SFP considered in this study. A possible exception to this would be Unit 1 at Fukushima Daiichi, which initially considered comparable seismic design-basis loads. However, seismic design basis loads for Unit 1 were subsequently revised upwards (those are the design loads reported in this comparison).

The depth of the 20 SFPs affected by the recent earthquakes in Japan is similar to that for the SFP considered in the study. The horizontal dimensions for the SFPs in Fukushima Daiichi (EW and NS dimensions with reference to Figure 17) are also similar with the exception to the

SFP for Unit 1, which has a significantly smaller NS span that tends to make the SFP for Unit 1 less vulnerable to seismic loads. The thickness of the floor slabs for the SFPs at Fukushima Daiichi are likely similar to those for the SFP considered in the study. The SFP for which more structural details were known at the time of the writing of this report is the SFP for Unit 4 at Fukushima Daiichi (hereafter referred to as Unit 4). The following provides a comparison of structural details for Unit 4 with those of the SFP considered in this study.

- For Unit 4, the thickness of the SFP floor is about 1.5 m (about 5 ft) which is less than the thickness of the floor of the SFP considered in this study which is about 1.83 m (about 6 ft).
- Available information indicates that the reinforcement of the wall of the SFP in Unit 4 is not significantly different from the reinforcement in the wall of the SFP considered in this study.
- Although no reference is made to out-of-plane shear reinforcement for Unit 4 of Fukushima Daiichi, it is not known with certainty at the time of the writing of this report if out-of-plane shear reinforcement was provided at the edges of the floor slab of Unit 4 or at the intersection of this floor slab with the vertical walls.
- For the SFP of Unit 4 no reference is made to a grid of steel beams analogous to that embedded in the floor and bottom of the walls of the SFP considered in this study (used to support the weight of wet concrete during construction).
- Cross section drawings of the reactor building for Unit 4 indicate the possibility of a load bearing wall under the South wall (with reference to Figure 17) of the SFP of Unit 4, which does not exist for the SFP considered in this study. This difference, if confirmed, would result in a longer span for the entire structure of the SFP considered in this study.

Although there are differences between the structures of the Unit 4 SFP and the SFP considered in this study, these differences do not seem to be sufficiently significant to assert without further analysis that the Unit 4 SFP would be stronger for the same seismic demands than the SFP considered in this study. Differences in the vertical and horizontal response spectra at the foundation of Unit 4 and at the base of reactor building considered in this study (a fixed base structure) (see Figure 32 and Figure 31) indicate that the seismic forces for Unit 4 would have been significantly less than those considered in this study. The difference between these seismic demands would have been the main factor affecting the relative performance of the Unit 4 SFP (under the March 11, 2011 earthquake) and the performance of the SFP considered in this study under the hypothetical beyond design basis earthquake.

Major observations from these comparisons are:

- For the challenging events that affected 20 reactors and SFPs, leakage from the bottom of the SFPs of the 20 BWR reactors was not reported. This is consistent with the highest relative likelihood estimate for this study being that for the state with no leakage.
- Possible differences in the design and construction of the reactor buildings and SFPs, which considered higher design-basis seismic loads, and the SFP considered in this study, introduces uncertainties in these observations.
- The ground motion used in this study may be more challenging for the spent fuel pool structure than those experienced at the Fukushima Daiichi nuclear power plant from the earthquake that occurred on March 11, 2011, off the coast of Japan, which did not cause spent fuel pool leaks at the bottom of the walls.

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 SFP. 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 five operating cycles at the reference 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.)[1]	—
Cell pitch	15.95 cm (6.28 in.)	—
Open vs. closed cell	Closed cell[2]	—
# of storage locations	3,819	
Fuel loading		
Min. assem. during outage[3]	3,819 – 764 – 284 = 2,771	284 × 2 = 568
Max assem. during outage	3,819 – 764 = 3,055[4]	284 × 3 = 852
# of assem. after outage	3,819 – 764 = 3,055	284 × 3 = 852
Newer fuel (<5 years)	GE14/GNF2[5]	—
Older fuel (>5 years)	Actual, based on 2003 info.	N/A
Pattern for “hot” fuel	prearranged in 1x4[6]	1x4 “with empties”
Coherent downcomer area[7]	Yes	—
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)[9]	—

1 Later in the conduct of the study the authors became aware that the distance between the pool floor liner and the bottom of the rack baseplate is actually (on average) closer to 26 centimeters (cm) (10.25 in.), depending on adjustments made to the leveling pad during installation. For the cases studied in this report, in which the leakage location is at the junction of the floor and side wall, side calculations have shown that the results are insensitive to this difference (i.e., even at 18 cm sufficient cross-sectional flow area exists to accommodate natural circulation flow). Nonetheless, any future analyses for this site (particularly if they involve leak locations on the pool side wall), should consider correcting this error.

2 This terminology refers to a rack design in which the sides of the rack cells have panels that inhibit or prevent cross-flow, while being relatively open at the top and bottom for axial flow.

3 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 percent of the core) based primarily on the information in Exelon (2011), with a slight change from 270 to 284 assemblies for MELCOR modeling convenience.

4 Sixty of these rack locations may be reserved for storing guide tubes. The study does not address this situation, but it 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 percent of the overall SFP inventory (and less than 2 percent of the radionuclide inventory available for release).

5 See Exelon (2011) for more information.

6 See Section 9.3 of this report for a discussion of how the use of contiguous (uniform) patterns would affect the results.

7 This term is used to describe whether an open area exists within the pool (such as an unracked region, a cask laydown area, or large gaps between the edge of the racks and the pool walls) that would facilitate downflow during conditions resulting in natural circulation air flow under the rack baseplate.

8 Note that the decay heat from the fuel left in the reactor is considered when the pool and reactor well are hydraulically connected.

9 After results were calculated based on a 700 day operating cycle, the authors realized that the correct operating cycle length should be 725 days (including the 25 day outage) rather than 700 days (which didn't include the outage). This error is expected to have a small impact on the overall results.

The above table depicts a 1x4 storage pattern for the recently discharged fuel, based on the approach PBAPS has taken to meet the requirements associated with license condition 2.C.(11) and 10 CFR 50.54(hh)(2). The plant studied actually currently utilizes a 1x8 pattern. Because this pattern is believed to be atypical (relative to the fleet), it is not modeled as the base case in this study. Section 9.2 of this report provides additional analysis that shows the benefit of the 1x8 pattern. Section 9.3 discusses of how the use of contiguous (uniform) patterns would affect the results. Figure 34 illustrates the different patterns.

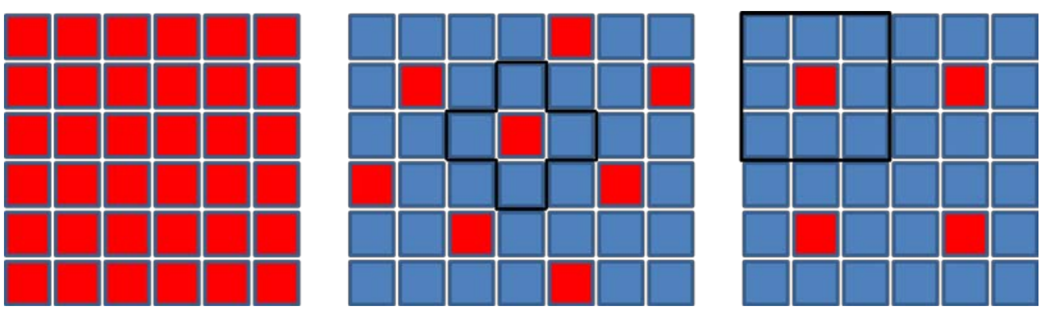


Figure 34 Illustration of SFP patterns

From left to right: Uniform/contiguous; 1x4 (used as base case);
1x8 (actually used by plant as of May 2012)
Red = a recently discharged assembly; Blue = an older, lower decay heat assembly, black outline shows repeating pattern

5.2 Operating Cycle Phase Specification

As described in Section 1.5, constant changes to the conditions in the SFP affect the consequences of a postulated accident (e.g., changes in the decay heat, changes in the inventory of fuel in the pool). Thus it is necessary to discretize this continuous behavior into a manageable set of discrete quasi-steady snapshots. Further, it must be recognized that the number of quasi-steady snapshots (or 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, defining the OCPs becomes a minimization/optimization problem (i.e., the analysis 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).

Based on these 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 a set of five OCPs, as outlined in Table 16.

Table 16 OCP Definition for the Modeled Operating Cycle

OCP #	OCP Description	Time (d)	% of opera -ting cycle	Pool-reactor configuration	Modeled 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	1x4	Existing[1] + (27% of offloaded assemblies) @ 4 days[2]	Highest powered offloaded assembly @ 4 days[2]
2	Reactor T&M / inspection and refueling	8–25	2.4	Refueling	1x4	Existing[1] + (offloaded assemblies) @ 13 days[2]	Highest powered offloaded assembly @ 13 days[2]
3	Highest decay power portion of nonoutage period	25–60	5	Unconnected	1x4	Existing[1] + (offloaded assemblies) @ 37 days[2]	Highest powered offloaded assembly @ 37 days[2]
4	Next highest decay power portion of nonoutage period	60–240	25.7	Unconnected	1x4	Existing[1] + (offloaded assemblies) @ 107 days[2]	Highest powered offloaded assembly @ 107 days[2]
5	Remainder of operating cycle	240–700; 0–2	66	Unconnected	1x4	Existing[1] + (offloaded assemblies) @ 383 days[2]	Highest powered offloaded assembly @ 383 days[2]

1 The term “existing” refers to the fuel residing in the SFP at t = 0 (before offload).
2 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

The following key assumptions in the above OCP definition warrant highlighting:

- The study does not explicitly treat the offloading of older fuel into casks (as part of the normal fuel management practices as opposed to an expedited fuel movement program). Rather, a stylized assumption is made that the 284 assemblies that would be loaded into dry casks during the operating cycle are instantaneously removed from the pool just before the outage.

- The study does not treat new fuel. This fuel would be placed into the SFP just before 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 available to participate in a propagating zirconium fire, but would have a negligible effect on the source term.[6]

- The actual time at which the snapshots are evaluated is based on the mean decay heat during the OCP, as opposed to the mean 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 OCP1, 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

One of the objectives of this study is to provide insights into the effectiveness and benefits of mitigation measures currently employed at nuclear power plants. In addition to the redundant and diverse physical systems designed to prevent severe accidents, NRC requires plant owners to have preplanned emergency measures in the unlikely event an accident occurs. When they are successfully implemented, NRC expects these emergency measures will mitigate accident consequences by preventing, delaying, or reducing a potential release of radioactive material from the SFP. These measures include a site-specific emergency plan, emergency operating procedures, severe accident management guidelines, and 10 CFR 50.54(hh)(2) mitigation measures put in place to respond to the loss of large areas of the plant due to fires or explosions. NRC requires its licensees to train and practice emergency measures to ensure that they have proper equipment, procedures, and training. NRC inspectors periodically observe these activities to help ensure that NRC regulations are met at each plant. The study assumes that the licensee's emergency response organization would implement these measures in accordance with approved emergency plans, procedures, and guidelines.

Regarding onsite mitigative actions, the assumptions chosen by the project team to define the scenarios analyzed using MELCOR and MACCS2 are described here. Two cases are modeled for each scenario, a mitigated case and an unmitigated case. In the mitigated case, the model includes what would happen if the operators are fully successful in carrying out onsite mitigating actions. However, NRC analyzes extreme events to gain insights on the safety margin provided by NRC's regulatory framework. The uncertainties associated with the response to a beyond-design-basis seismic event, and the resultant effects on the SFP, make consideration of unmitigated scenarios prudent from an informed decision-making standpoint. Thus, each scenario is also analyzed assuming that the operators are not successful in implementing onsite mitigating actions.

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[6] The radioactive material that is of concern during an accident is the fission products generated while the fuel is in the reactor. The uranium dioxide (UO_2) present in fresh fuel would not contribute noticeably to the source term, and in particular, not in a SFP accident in which the temperatures during a postulated accident are lower than those during a reactor accident.

In the unmitigated case, 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 10 CFR 50.54(hh)(2)[7] equipment. The cases which assume lack of successful mitigation are presented to (1) acknowledge uncertainties in the effectiveness of these efforts during a beyond-design basis event and (2) demonstrate the effectiveness of successful mitigation. Section 5.3.2 of this report discusses further the rationale for developing results for this situation.

In the mitigated case, (1) mitigative actions associated with the regulatory requirements of 10 CFR 50.54(hh)(2) are successfully deployed, (2) additional onsite capabilities are used to extend the use of this equipment, and (3) arrival of offsite resources allows this equipment to be utilized for an extended period of time (e.g., days) until onsite capabilities can be recovered.

This study's original scope did not include an attempt to quantify the likelihood of successful execution of different mitigative actions that might take place (e.g., makeup using a portable pump, recovery of ac power). Subsequent to completion of the MELCOR (Chapter 6) and MACCS2 (Chapter 7) analyses described in the following chapters, the project staff performed a human reliability analysis for the purpose of providing context regarding human response. The HRA results provide informative data to gain insights on the likelihood of mitigation being successfully implemented as well as possible regulatory enhancements for consideration. Chapter 8 describes the HRA. Since the HRA was performed after the bulk of the analysis was completed, some of the assumptions differ from those described in this Section.

In addition to onsite mitigation, offsite support is considered in the paragraphs below.

The reference plant is supported by an offsite emergency operating facility (EOF). The emergency response organization at the EOF has access to fleetwide emergency response personnel and equipment, including the 10 CFR 50.54(hh)(2) mitigation measures and equipment from the sister plants. Every licensee participates in full onsite and offsite exercises every 2 years where response to severe accidents and coordination with offsite response organizations is demonstrated and inspected by the NRC and the Federal Emergency Management Agency. In addition, the Institute for Nuclear Power Operations and the Nuclear Energy Institute would activate their emergency response centers to assist the site as needed.

Concurrent with the industry response, the U.S. National Response Framework (NRF) would establish a coordinated response of national assets. As described in the Nuclear/Radiological Incident Annex to the NRF, the NRC is typically the Coordinating Agency for incidents occurring at NRC-licensed facilities. As Coordinating Agency, the NRC has technical leadership for the Federal Government's response to the incident. The NRF conducts periodic exercises and provides access to the full resources of the Federal Government. The NRC has an extensive, well-trained and exercised, emergency response capability and has onsite resident inspectors. The NRC would activate the incident response team at the NRC regional office and Headquarters. The focus of the NRC response is to ensure that public health and safety are protected and to assist the licensee with the response.

[7] This section of the regulations deals with the development and implementation of guidance and strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with loss of large areas of the plant resulting from explosions or fire.

However, for the large beyond-design-basis seismic event under consideration in this study, it is possible that significant damage to local infrastructure could occur, requiring emergency resources to also be needed in other areas. Additionally, radiation and other hazards (discussed in Section 5.3.2 of this report) could hinder access to the SFP and key equipment, making prevention or truncation of an ongoing SFP release challenging.

Considering the uncertainties associated with this event as described above, project staff chose a 72-hour time truncation (assumed that the event would be terminated by some means by 72 hours after initiation). The use of a time truncation is a point of uncertainty that can significantly affect the results and is further analyzed in Section 9.8 of this report. Note that like other aspects of this study, the incorporation of ongoing changes in regulatory commitments related to offsite response emanating from the Japan Lessons Learned initiatives is beyond this study's scope.

Regarding offsite support for these situations, the accident progression analysis assumes the following for the purposes of this study:

- Within 24 hours, offsite support arrives.

- Within 48 hours, actions are planned and equipment is staged.

- At 48 hours, if the fuel is not uncovered and the pool can be refilled with an injection rate of 500 gpm (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 because of the additional complexities of (1) accessing the area of the pool when the fuel is uncovered and stopping an ex-containment release in progress and (2) performing a large leak repair.

These assumptions are similar to the assumptions used in NUREG-1935, "State-of-the-Art Reactor Consequence Analyses Project" (NRC, 2012i).

Table 17 summarizes each situation.

Table 17 Summary of Mitigation Assumptions

Item	Situations with successful deployment of onsite mitigation	Situations without successful deployment of onsite mitigation
Installed accident mitigation equipment	Damaged by the event; recovery/repair not credited	
10 CFR 50.54(hh)(2) equipment	Successfully deployed 2 hours after diagnosis	Not credited
Other onsite resources	Successfully deployed to extend operation of 10 CFR 50.54(hh)(2) equipment	Not credited
Offsite resources	Successfully deployed for terminating the accident at 48 or 72 hours (see Section 9.8)	
Emergency preparedness	Effective (see APPENDIX A: of this report for more details)	
Mitigation equipment being considered under NRC Order EA-12-049, dated March 12, 2012	Not considered; may be substantively similar to 10 CFR 50.54(hh)(2) capabilities within the context of this study	

5.3.1 Approach Details and Assumptions

Scenarios that credit successful deployment of the 10 CFR 50.54(hh)(2) measures must include assumptions about how that deployment is executed. In general, this study utilizes some of the limits associated with these capabilities that are contained in Nuclear Energy Institute (NEI) 06-12, Revision 2, "B.5.b Phase 2 & 3 Submittal Guidance," issued December 2006 (which the NRC has endorsed[8]). 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. This study does not invoke that provision because the site in question strives to deploy the equipment within 2 hours regardless of the fuel pattern and the existence of cases without successful deployment of mitigation envelops this effect.

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 delivered per minute for makeup). For PBAPS, the capacities of the available equipment are somewhat higher. The use of the 500 and 200 gpm values in this study 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.[9] As a result, no additional "penalty" is given

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8 The NRC originally endorsed this document for operating reactors by letter dated December 22, 2006 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). B.5.b refers to Section B.5.b of Order EA-02-026, dated February 25, 2002, and later made generically applicable in 10 CFR 50.54(hh)(2).

9 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

for inefficiencies associated with spray coverage (i.e., the spray flow rate is applied uniformly across the pool cross-sectional area without further reduction). In either spray or makeup mode, the licensee would utilize a portable diesel-driven pump to pump water from either the fire ring header, the intake canal, or the emergency cooling tower basin to the refueling floor via hoses that would be run up a reactor building stairwell.

The following set of criteria was established to model the time to diagnosis of the need to deploy 10 CFR 50.54(hh)(2) mitigative strategies:

- no ac power
- SFP level decrease by 1.5 m (5 ft), keeping in mind that 0.5 m (1.5 ft) is lost because of sloshing
- 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 SFP, though these specific criteria do not 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 SFP cooling or loss of SFP inventory do refer plant personnel to the guidelines for use of the 10 CFR 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. Note that the details of onsite response are covered more thoroughly in Chapter 8.

The above criteria could be conservative or nonconservative depending on the priorities of operators, and different criteria would clearly be more applicable to other scenarios, particularly those that did not include loss of offsite and onsite power at time zero. The assumption that pool elevation must drop 5 ft can lead to long diagnosis time periods for slowly progressing events, thus leading to a potentially conservative timeline for mitigative action. However, these same slowly developing scenarios are the ones that are least important for 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. Chapter 8 discusses the issue of diagnosis in greater detail.

Regarding the implementation mode, for cases in which the water level in the pool is greater than 0.9 m (3 ft) above the top of the racks (a surrogate for high radiation levels on the refueling floor near the edge of the SFP (see Section 5.4 of this report)) 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

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entering the pool. The regulatory implementation of 10 CFR 50.54(hh)(2) accounts for this inefficiency effect.

in Figure 2-1 of NEI 06-12, Revision 2 (NEI, 2006), regarding whether SFP leakage is excessive. In some respects it is a more complicated approach than might be used, but is arguably a more straightforward 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 OCP1/2, in which, because of 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 ft 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. Section 9.3 of this report investigates the effect of this assumption for a uniform pattern.

Whichever mode is initiated (spray versus makeup), it is assumed to be used for the duration of the event (i.e., no later switching to a different mode). For OCP 1/2 with the "moderate" leakage condition, makeup is deployed. Other, equally reasonable assumptions about mitigation deployment could result in the deployment of sprays instead (which have a potential advantage in terms of mitigation for these conditions). Section 9.3 presents a sensitivity study related to this assumption, for a uniform pattern.

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 a calculation without successful deployment of mitigation):

- Start of calculation/earthquake occurs.

- When SFP level has decreased by 1.5 m (5 ft), and 30 (diagnosis delay) plus 120 (initial deployment delay) additional minutes have transpired, then the following applies:

    - If the water level is greater than 0.9 m (3 ft) above the top of the fuel, then 500 gpm of makeup into the top of the pool commences.

    - If the water level is less than 0.9 m (3 ft) above the top of the fuel (thus indicating excessive leakage) then 200 gpm of spray at the top of the pool commences.

The above assumptions are characterized as optimistic relative to the unmitigated (pessimistic) case. However, it is important to note that aspects of these assumptions assume failures where they may not occur. For instance, the above set of assumptions only credits 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 SFP, several of which have much higher capacities than the mode selected. Table 10.3.1 of the FSAR captures these alternatives, which 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 in this study), these modes require either multiple manual alignments in the vicinity of the SFP and reactor, the availability of ac power for valve manipulations, or the use of equipment that might be involved in reactor recovery (most notably a residual heat removal pump), as well as ac power for pump operation. Finally, as mentioned previously, 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, uncertainties associated with the response to a beyond-design-basis seismic event, and the resultant effects on the SFP, make consideration of unmitigated scenarios prudent from an informed decision-making standpoint. Some specific issues relevant to the situation considered in this report include the following:

- The regulatory requirements for 10 CFR 50.54(hh)(2) equipment currently focus on the use of this equipment for responding to a loss of a 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). Note that some plants (including the reference plant) have already acquired some additional equipment in anticipation of this requirement, with full compliance scheduled for 2016.

- The large seismic event could damage onsite (and offsite) infrastructure designed to facilitate accident response, as well as cause 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. Section 5.4 of this report describes the shielding analyses that inform this aspect of the accident analysis. Chapter 8 further discusses accessibility issues in the context of human response. Note that, as part of the implementation of 10 CFR 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).[10]

- A concurrent reactor event (resulting from the loss of ac power or other damage), or an ongoing accident at the other unit's SFP, could hamper mitigative actions by reducing accessibility because of radiation fields, impeding accessibility because of other hazards such as hydrogen accumulation, or diverting resources (both personnel and equipment). Chapter 8 discusses this issue further.

- An assembly being moved within the SFP (or from the reactor to the SFP or vice versa) at the time of the event, could lead to an earlier radiological hazard for responders, if this assembly were to become uncovered earlier in the event progression, because of its higher position in the SFP. Section 5.4 of this report provides refuel floor dose rate estimates for this situation.

- Accessibility could be reduced if an inadvertent criticality event in the SFP were to occur. See Section 2.3 of this report for more information about inadvertent criticality events.

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[10] The industry's FLEX proposal, developed in response to NRC Order EA-12-049, includes a specification for a means to connect makeup to the installed SFP cooling system to overcome the potential for lack of access to the SFP deck area. This is primarily to address the potential needs for makeup in a saturated condition caused by boil off for an uncooled pool.

For these reasons, this study presents results for cases in which accident mitigation efforts are unsuccessful for some period of time.

5.4 Refueling Floor Dose Rate Analysis Using SCALE

This study included analyses to predict the radiological conditions on the refuel floor for a range of conditions associated with loss of water in the SFP. Note that the analyses described in this section only account for the radiological conditions stemming from neutron and gamma "shine" from exposed radioactive material and do not account for 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 SFP were to commence, radiation fields in the vicinity of the pool would be extremely high.

The analyses described, which Oak Ridge National Laboratory (ORNL) performed, looked at a range of conditions. This range included both the high-density and low-density loading conditions studied in this report, as well as the situation in which a single assembly is in the lifted position at the time of the event. The times following discharge that were considered are the same as those associated with the different OCPs. This portion of the analyses is plant specific for the reference plant, and utilized 2011 vintage information for representing the fuel design and characteristics in the SFP. Calculations were performed using the ORIGEN and MAVRIC modules of the SCALE code suite. MAVRIC in turn used BONAMI, CENTRM, DENOVO, and Monaco routines, along with the FW-CADIS methodology. The analysis used the flux-to-dose conversion factors in American National Standards Institute/American Nuclear Society (ANSI/ANS) 6.1.1-1977. The 200 neutron group and 47 gamma group cross sections based on the ENDF/B-VII.0 cross-section library distributed with SCALE 6.1 were used.

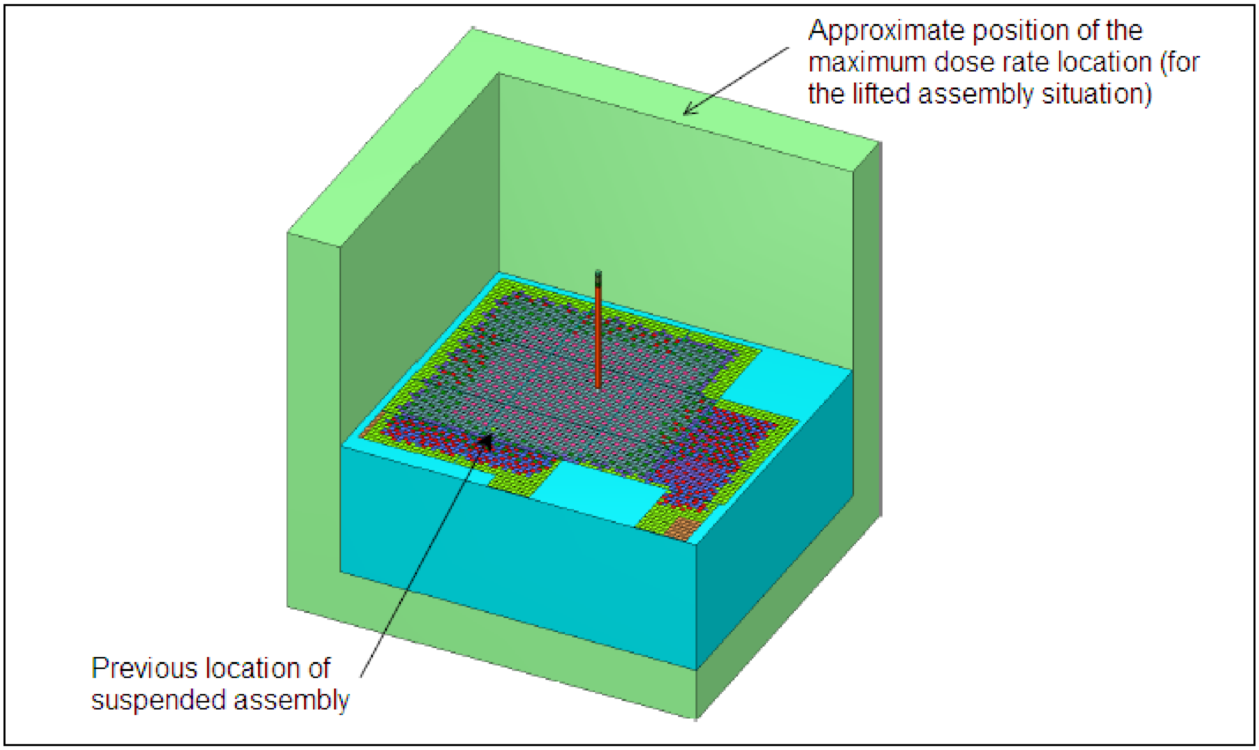

Figure 35 Cutaway depiction of a lifted assembly with water level at the top of the racks

Results of the analyses for the high-density loading situation can be summarized as follows:

- For water depths of 3 m (10 ft) above the top of the racks, projected dose rates are very, very low (less than 0.1 millirem (mrem) per hour). This is consistent with Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," which uses this water depth as a conservative measure of adequate shielding.

- Dose rates for the maximally exposed location on the refueling floor once the water level drops to the top of fuel hardware are very high (on the order of 450 to 600 rem per hour, depending on the OCP).[11]

- At a water depth of 0.6 m (2ft) above the top of the fuel, the projected dose at the maximally exposed location on the refueling floor surpasses 25 rem in one hour. 25 rem is the 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 U.S. Environmental Protection Agency (EPA) 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," issued May 1992)

- Dose rates elsewhere on the refueling floor are significantly lower than those at the maximally exposed location (e.g., see Figure 36).

[11] This range shows that, for situations in which the water level is at the top of the fuel hardware, the dose rates are somewhat sensitive to the time during the operating cycle (a 33-percent decrease in this case). For instances in which water is covering the fuel hardware, this sensitivity decreases. For example, the analogous range of values for a water level 100 cm (3.3 ft) above the fuel hardware is 1.6 to 1.7 rem per hour.

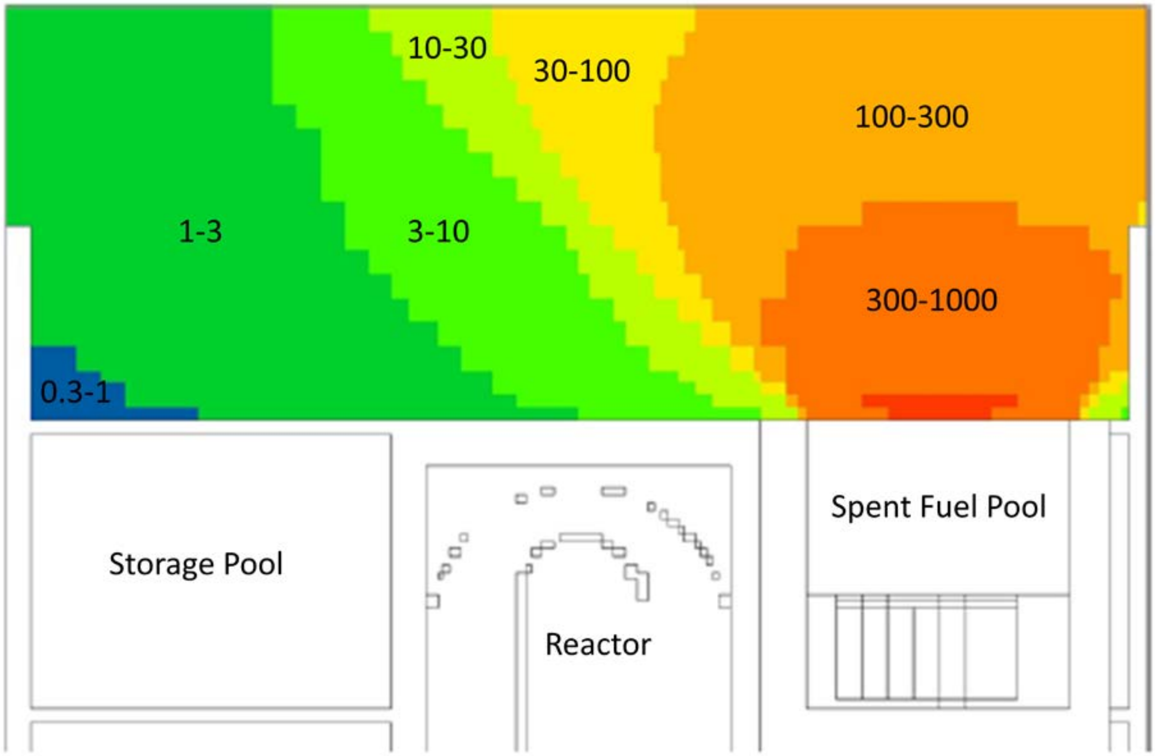

Figure 36 Approximate dose rate of elevation contours, water at top of fuel hardware, around the time of defueling (rem per hour).

Relative to the high-density loading situation, the other situations can be compared as follows:

- For low-density loading situations, dose rates for the maximally exposed location on the refueling floor once the water level drops to the top of fuel hardware are lower than the high-density loading case, but still very high (on the order of 300 to 470 rem per hour, depending on the OCP).

- For a recently discharged assembly in the lifted position, dose rates for the maximally-exposed location are on the order of 3 rem per hour when the water level is at the top of the lifted assembly and over 1,000 rem per hour when the water level is at the top of the racks (i.e., when the lifted assembly is completely exposed). These are dose rate contributions solely from the lifted assembly (i.e., they are in addition to the dose rate contributions from the assemblies in the racks).

- For an older assembly (discharged more than a decade previously) in the lifted position, dose rates for the maximally exposed location are on the order of 0.2 rem per hour when the water level is at the top of the lifted assembly and 7 rem per hour when the water level is at the top of the racks. Again, these are dose rate contributions solely from the lifted assembly.

The high dose rates associated with the single lifted assembly (particularly those for the recently discharged assembly) are sensitive to the assumed position of that assembly. This case assumes that the assembly is located somewhere near the middle of the pool (see

Figure 35), which results in direct line-of-sight from the edge of the SFP. Placement near a wall would reduce the dose rate for locations near the edge of the pool that do not have a direct line of sight to the assembly.

5.5 Discussion of Repair and Recovery

This study makes no attempt 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. 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 an SBO cross-tie line to the Conowingo Dam. The study assumes that the damage sustained by the onsite and offsite electrical distribution systems from the earthquake is enough to significantly delay these recoveries until after the 48- or 72-hour truncation times.

That being said, and as covered previously in this section, the scenarios with successful deployment of mitigation do assume that onsite and offsite resources are able to extend operation of the 10 CFR 50.54(hh)(2) equipment indefinitely, which could represent a situation in which ac power is recovered at an intermediate point and ac-dependent means of SFP makeup are brought back online.

5.6 Scenario Development

5.6.1 Identification of Key Events

The scenario development included the following major assumptions based on the structural analysis documented in Section 4 of this report or other considerations:

- All offsite and onsite ac power is lost as a direct result of the seismic event (see Section 4.2 of this report).
- Direct current power may be lost. Because of the difference from the reactor situation (in which dc power to control turbine-driven systems is important in an SBO), the availability or unavailability of dc power has a much narrower effect. For the specific set of assumptions used in the MELCOR and MACCS2 analyses, there is no effect as analyzed. Chapter 8 further discusses this issue with respect to the HRA.
- The 10 CFR 50.54(hh)(2) equipment (when credited) is available for the duration of the event, following delays associated with diagnosis and deployment (see Section 5.3.1 of this report).
- Initial water loss from “sloshing” will be 0.5 m (1.5 ft) (see Section 4.2 of this report).
- Tearing of the SFP liner is not the most probable outcome, but is possible (see Section 4.1 of this report).
- There is no failure of penetrations, including the refueling transfer canal gate (see Section 4.2 of this report).

- The overhead structures (building debris, crane) do not pose a threat to the SFP in terms of failure resulting from the initiating event (see Section 4.2 of this report).

- Inadvertent criticality, including seismic effects on the integrated poison rack material, is not treated (see Section 2.3 of this report).

5.6.2 Scenario Calculation Matrices

The following table shows how the combinations described thus far translate to the scenarios considered for each OCP.

Table 18 Scenario Breakdown per OCP

Case #	Scenario Characteristics		Radioactive Release Commences before 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		

5.6.3 Summary of Event Split Fractions

As described previously, the analysis considered the available seismic hazard information to obtain an initiating event frequency of approximately one event in 60,000 years for the reference plant.

Table 19 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	5.2×10^{-4}	Damage not expected
2	0.3 to 0.5	0.4	1 in 40,000	2.7×10^{-5}	Damage not expected
3	0.5 to 1.0	0.7	1 in 60,000	1.7×10^{-5}	Damage possible
4	> 1.0	> 1.0	1 in 200,000	4.9×10^{-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 20 Refresher on ac Fragility

Item	Relative Likelihood	Comments
Loss of normal SFP cooling	0.84	This study used the ac power fragility from NUREG-1150 of 0.84 as a surrogate for the conditional probability of normal SFP cooling and makeup not being available. This simplifying assumption was made in light of the fact that the study is not a PRA (but rather a consequence analysis with probabilistic considerations) and that this value already approximates the upper bound of 1. In reality, the availability of normal SFP cooling and makeup would be a combination of the AC fragility, the fragility of the actual equipment and its support equipment, and operator actions to recover the equipment, which could result in a conditional probability higher than the value used here.

As described previously, the structural assessment led to the SFP leakage estimates stated below.

Table 21 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 tens of hours
“Moderate” leakage	0.05	Tearing of SFP liner; damaged concrete limits outflow; drains pool within 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 percent for OCP1 to 66 percent for OCP5 (recall that the OCPs were intentionally “front loaded” because the most change in SFP conditions occurs during the outage).

Table 22 Refresher on the OCP Fractional Contributions

OCP #	Time window (Time of evaluation) (in days)	Fraction of operating cycle	Pool-reactor configuration	Spent fuel configuration for high-density loading
1	2–8 (5)	0.01	Refueling	Dispersed (except for Section 9.3)
2	8–25 (13)	0.02		
3	25–60 (37)	0.05	Unconnected	Dispersed
4	60–240 (107)	0.26		
5	240–700 and 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 do not), so as to assign scenario-specific release frequencies, scenario-specific individual risk of an LCF, or the like. 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 Spent Fuel Pools 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. MELCOR has been developed through the 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 heatup 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 comprises 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. 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. Plant systems and their response to off-normal or accident conditions include the following:

- 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, as well as 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

MELCOR modeling 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 more reasonable execution times. The input structure for MELCOR 2.1 differs completely from that of 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 SFP accidents and (2) it is able to model the accident progression, radionuclide release, and in-building transport/retention. MELCOR 1.8.6 was used in the present study, and the SFP models in both versions of the code (1.8.6 and 2.1) are functionally the same.

As part of NRC's post-9/11 security assessments, the agency developed and applied SFP modeling using detailed thermal-hydraulic and severe accident progression models integrated into the MELCOR code to assess the realistic heatup of spent fuel under various pool draining conditions. SNL performed the analyses for a reference BWR, with additional supporting analyses for separate effects and fluid flow modeling 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 an SFP) and formation and convection of stratified molten pools.

MELCOR 1.8.5 Version RP added two modeling enhancements applicable to BWR SFP modeling (also included in MELCOR 1.8.6 and 2.1): (1) a new rack component, which permits better modeling of an SFP rack and (2) a new oxidation kinetics model. The new BWR SFP 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. These new SFP features can be used to perform two types of SFP calculations: (1) a 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 ruthenium release. A partial loss-of-coolant inventory or boiloff accident could involve no or late uncovery of the bottom of the racks. Boiloff of the coolant leads to steam generation and steam oxidation of the cladding and hydrogen generation that could lead to hydrogen combustion.

Breakaway Oxidation Model

Argonne National Laboratory (ANL) has performed oxidation kinetics testing on zirconium-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 associated with 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 as follows:

$$\frac{dw^2}{dt} = K(T) \qquad (1)$$

where, *w* is the reacted metal mass per unit surface area. 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 Zircaloy-4 oxidation correlations are provided below.

The steam preoxidized, wide-temperature, prebreakaway Zircaloy-4 oxidation correlation (Natesan and Soppet, 2004) is as follows:

$$\mathrm{K(T)} = 26.7 \exp\left(-17{,}490/\mathrm{T}\right) \; [\mathrm{kg}^2/\mathrm{m}^4.\mathrm{s}] \qquad (2)$$

The steam preoxidized, wide-temperature, postbreakaway Zircaloy-4 oxidation correlation (Natesan and Soppet, 2004) is as follows:

$$K(T) = 2.97E4 \exp(-19{,}680/T) \ [kg^2/m^4.s] \tag{3}$$

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:

$$LF = \int_0^t dt' \frac{t'}{\tau(T)} \tag{4}$$

$$\tau(T) = 10^{P_{LOX}} \tag{5}$$

$$P_{LOX} = -12.528 \log_{10} T + 42.038 \tag{6}$$

where P_{LOX} is the MELCOR fit of the timing for the transition from prebreakaway to postbreakaway oxidation reaction kinetics for Zirlo and Zircaloy-4 in the ANL experiments.

The air oxidation model was benchmarked against experimental data from the SNL SFP facility as part of the security assessment work. 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 postbreakaway oxidation kinetics.

Hydraulic Resistance Model

The MELCOR modeling approach for flowpaths 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 flowpath 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:

$$\Delta P = \frac{1}{2} \rho v^2 \left(f \frac{L}{D} + K\right) \tag{7}$$

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:

$$f = S_{LAM}/Re \tag{8}$$

where S_{LAM} is a user-specified MELCOR input parameter, *Re* is the Reynolds number ($\rho v D/\mu$), and μ is the fluid dynamic viscosity.

Hydraulic resistance measurements were performed on a Global Nuclear Fuel 9x9 BWR assembly at SNL (Durbin, 2005) to obtain the required frictional and form loss coefficients, including the effects of grid spacer and partial rods. The present study used these measurements given a lack of data for a 10x10 BWR assembly.

6.1.2 Heat Transfer Modeling within Spent Fuel Pool and to Surrounding Walls

The MELCOR core models calculate the thermal response of the core.[12] 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 SFP 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 preexisting 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 intracell radiation or conduction, but the user is responsible for defining a single input parameter that captures the geometry of the heat transfer path. Figure 37 depicts the heat transfer paths within a ring and across a ring boundary. For radiation between different core rings, the user adjusts the view factors and the surface areas.

The core models radiative heat transfer from the outermost ring components (if present) to the core boundary specified as a heat structure. The SFP 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.

[12] MELCOR core models were originally designed for the reactor core. Because of the code flexibility, the same modeling approach can be used for the spent fuel pool (with the addition of the rack as a separate component). Therefore, as far as code models are concerned (e.g., heat transfer between groups of assemblies and with the fluid, and radionuclide release, transport and deposition), there is no difference between reactor assemblies and spent fuel assemblies. It is up to the user to define the propoer information in the input deck.

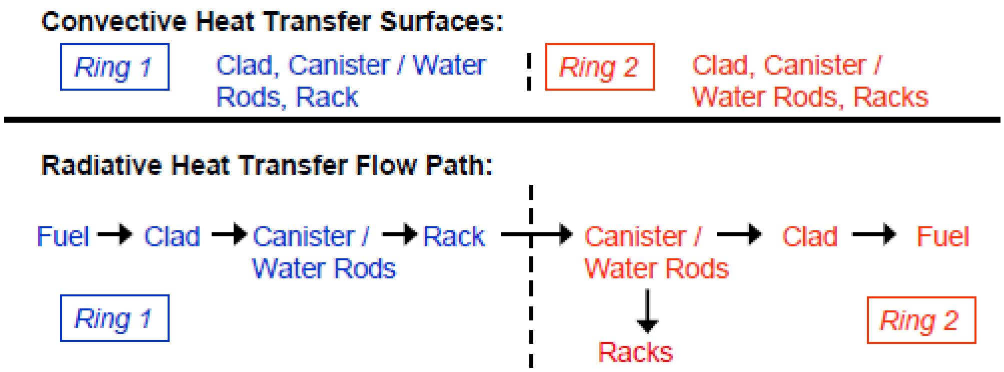

Figure 37 MELCOR modeling of heat transfer paths

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 atmosphere and includes droplet heat and mass transfer and fission product removal capabilities. All calculations used a droplet size of 1,250 microns. The spray was positioned at the top of the SFP (elevation of the refueling bay), thus allowing the droplets to be directed into the assemblies and open spaces based on their respective cross-sectional areas.

The interphase momentum model, which replicates the Wallis flooding curve, controls the penetration of the spray water into the assembly. 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 38). 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 percent (i.e., $\alpha > 0.998$). Because of 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.

MELCOR thermal-hydraulic model interprets the liquid film as a small pool at the bottom of each control volume (see Figure 38). Because of 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 which permits a better local representation of the fluid conditions and the location of the spray dryout. Parametric calculations are performed to show the impact of this modeling parameter (i.e., flow regime model active or inactive as discussed in 6.3.1).

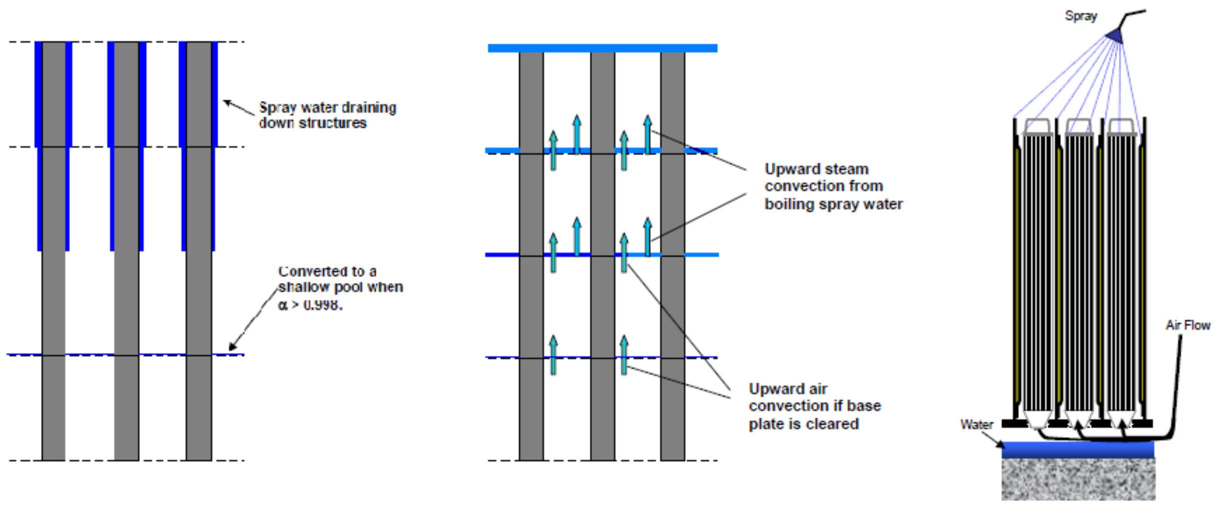

Figure 38 Spray model for SFP analysis

6.1.4 Modeling of Fuel Collapse and Baseplate Failure

Fuel collapse is based on user-defined cumulative fuel damage fraction logic, in which 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 are 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 and 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 RN package models the release and transport of fission product vapors and aerosols (referred to as radionuclides). Release of radionuclides can occur from the fuel-cladding gap by exceeding a failure temperature criterion or losing intact geometry or 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 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 nonradioactive mass to properly model the transport of fission products. The SFP MELCOR model includes 15 default material classes and two user-defined classes to model the behavior of cesium iodide (CsI) and cesium molybdate (Cs_2MoO_4), as shown in Table 23.

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 iodine with the default inventory wholly transferred to Class 16.

- Class 7—Characteristic released compound is molybdenum 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 Cs_2MoO_4. The released vapor pressure and compound mass is consistent with Cs_2MoO_4.

Gauntt (2010) proposes an approach for the estimation of increased ruthenium release under air-oxidation conditions. Ruthenium (Class 6) has the lowest of vapor pressures in the default MELCOR model that prevents prediction of large releases.[13] 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 hyperstoichiometric fuel ($UO_{2.15}$) and release of ruthenium dioxide (RuO_2). The default vapor pressure parameters in MELCOR are adjusted for the ruthenium class to match RuO_2 vapor pressure at 2200 K.[14] The new ruthenium release model is applied only to scenarios involving rapid draindown (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.

Table 23 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 Molybdate	Cs_2MoO_4	Cs_2MoO_4

The gap inventory specified in Table 24 is based on NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” issued February 1995 (NRC, 1995). However, NUREG-1465 states that, for accidents in which long-term cooling is maintained (e.g., postulated spent fuel handling accident), the gap release could be as low as 3 percent, and, in the unmitigated scenarios in this study, the fuel experiences prolonged high temperatures (and even failure in some instances). Therefore, the present work assumes that 5 percent applies to all scenarios.

[13] There is a mass transfer limitation to the release from the fuel.

[14] The rationale for an increased ruthenium class release is based only on increased vapor pressure and requires further experimental validation.

The decay heat calculation was based on security assessment analyses 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 burnup and power history. The utility provided the program and the appropriate input files for the SFP configuration after its last offload (i.e., September 2001) to perform decay heat calculations. Consequently, the decay heat power of every assembly was calculated as a function of time from reactor shutdown.[15] 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.

Table 24 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 burnup. The effective operating power was calculated as the total burnup of all assemblies in the SFP (gigawatt days per metric tons of uranium) divided by the average assembly metric tons of uranium and the total number of days of criticality. Based on the effective operating power, MELCOR calculates the specific time-dependent decay heat and mass inventory for each element. 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 (kilogram (kg)) times the specific element decay heat (watts per kilogram) 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 postprocessing routine is implemented that uses the MELCOR predicted release fractions along with actual inventories calculated for each batch.

15 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 percent). Since the number of old assemblies was increased by 60 (3,055 total in the pool), the decay heat for these assemblies was assumed to be an average of the older assemblies.

To accommodate consequence calculations using MACCS2, an extensive control system was written in the MELCOR input file that tracks the fission product releases from each ring[16] and the subsequent release to the environment. Time-dependent, nondimensional 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 MACCS2. MELCOR activity release for each isotope (e.g., m = Cs-137, Cs-134, Cs-136 for Class 2) is given by the following:

$$A_m(t) = \sum_{r=1}^{6}[RF_{m,r}(t) \times A^0_{m,r}] \qquad (9)$$

MACCS activity release is given by the following:

$$\tilde{A}_m(t) = \widetilde{RF}(t) \times \sum_{r=1}^{6} A^0_{m,r} \qquad (10)$$

where $\widetilde{RF}(t)$ is defined as:

$$\widetilde{RF}(t) \times A^0_1 + \ldots + \widetilde{RF}(t) \times A^0_M = \sum_{m=1}^{M} A_m(t) \qquad (11)$$

$$\widetilde{RF}(t) = \frac{\sum_{m=1}^{M} A_m(t)}{\sum_{m=1}^{M} A^o{}_m} = \frac{\sum_{m=1}^{M} A_m(t)}{\sum_{m=1}^{M}\sum_{r=1}^{6} A^0_{m,r}} \qquad (12)$$

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)
$\widetilde{RF}(t)$ = environmental release fraction (by radionuclide group for MACCS2)
A = released activity (Becquerel (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 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 identification, design type, initial enrichment, discharge burnup, and discharge date. The analysis basis for the high-density SFP inventory was 3,055 assemblies, a number based on the pool capacity of 3,819 assemblies, reduced by 764 assemblies to accommodate a full core offload capability. Information on

[16] A ring is a collection of assemblies in the MELCOR radial nodalization.

assemblies discharged before 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 et al., 2011), maintained within the SCALE nuclear safety analysis code system (Rearden et al., 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 7x7, 8x8, 9x9, and 10x10 General Electric assembly designs (Ilas et al., 2006) used in the reference plant 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 and Gauld, 2008) to be accurate within plus or minus 2 percent.

For the burnup analysis, the irradiation and decay history for each of the 3,055 assemblies in the pool was simulated 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, 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 (1,683 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 5 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.[17] Therefore, the actual inventories from batches were scaled appropriately to correspond to the rings in the 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 SFP results were compiled for each assembly group and all decay times and included activities (Bq) for 69 radionuclides and decay heat.[18]

Results from the present analysis were compared with those generated previously for the reference plant pool using assembly data provided by the utility through 2001 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

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[17] The nodalization was based on the security assessment work. The additional data on later cycles were received after the MELCOR model had been developed and the calculations were started.

[18] The decay heat in the present analysis is based on the past security assessment work.

average discharge burnup increased significantly after Cycle 12. The burnup values used in the present analysis are significantly higher and therefore more representative of modern SFP inventories than earlier analyses. Previous analyses using the 2001 data are representative of discharged fuel up to about 1995.

Other differences are attributed to the specific power of the assemblies which influences the decay heat power and activities of short-lived fission products in the analysis time range. The utility did not provide information on the specific power. Notwithstanding power uprates for the reference plant reactor, the most recent occurring in 2002, the specific power used to calculate inventories for the assemblies in the present analysis was lower than that 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 percent larger than the current results, likely the result of 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 percent over all cooling times, with present results slightly larger than utility values, most likely because of the increase in discharge burnup since 2001.

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 an SFP calculation, ignition is assumed to occur in the reactor building when the hydrogen concentration exceeds 10 percent by volume. In addition, MELCOR checks to determine whether there is sufficient oxygen. The minimum oxygen mole fraction for ignition is 5 percent. The maximum diluents mole fraction for ignition (mole fraction of steam plus mole fraction of carbon dioxide) is 55 percent. If all of these conditions are satisfied, a burn is initiated. Some uncertainty may exist regarding the combustion of hydrogen, especially with regard to the timing of a spontaneous ignition. 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 in which combustion was very likely or very unlikely. Consequently, the SFPS presents the results of cases with and without combustion. However, some cases have conditions in which the occurrence or timing of a combustion event has more uncertainty; these cases were assumed to ignite or not ignite according to the default spontaneous combustion criteria in MELCOR (see Section 9.1). Once a burn is initiated, it can propagate to other control volumes using the default hydrogen concentrations of 4 percent, 6 percent, and 9 percent for upward, horizontal, and downward propagations, respectively.

6.2 Description of MELCOR Models

The SFP, 40 ft (12.2 m) wide by 35.3 ft (10.8 m) long by 38.75 ft (11.8 m) 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-in.- (0.63-cm-) thick stainless steel. The walls and the floor of the

SFP are approximately 6 ft (1.83 m) thick. In the northeast corner of the SFP is a cask area that is 10 square feet (ft^2) (0.93 square meter (m^2)).

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 (7 m) of water for radiation shielding. The SFP racks are freestanding, full length, and top entry and are designed to maintain the spent fuel in a spaced geometry that 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 3,819 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 baseplate, and base support assemblies. Each cell is composed of (1) a full-length enclosure constructed of 0.075-in.- (0.2-cm-) thick stainless steel, (2) sections of Bisco Boraflex, which is a neutron-absorbing material, and (3) wrapper plates constructed of 0.020-in.- (0.05-cm-) thick stainless steel. The inside square dimension of a cell enclosure is 6.07 in. (0.15 m). The cell pitch is 6.28 in. (0.16 m). The baseplate is made from 0.5-in.- (1.27-cm-) thick stainless steel with 3.8 in. (0.1 m) 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 (see Figure 39). 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-in. 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 39 shows the layout of the rack cells. There is the potential for lateral cell-to-cell flow between connected rack cells.

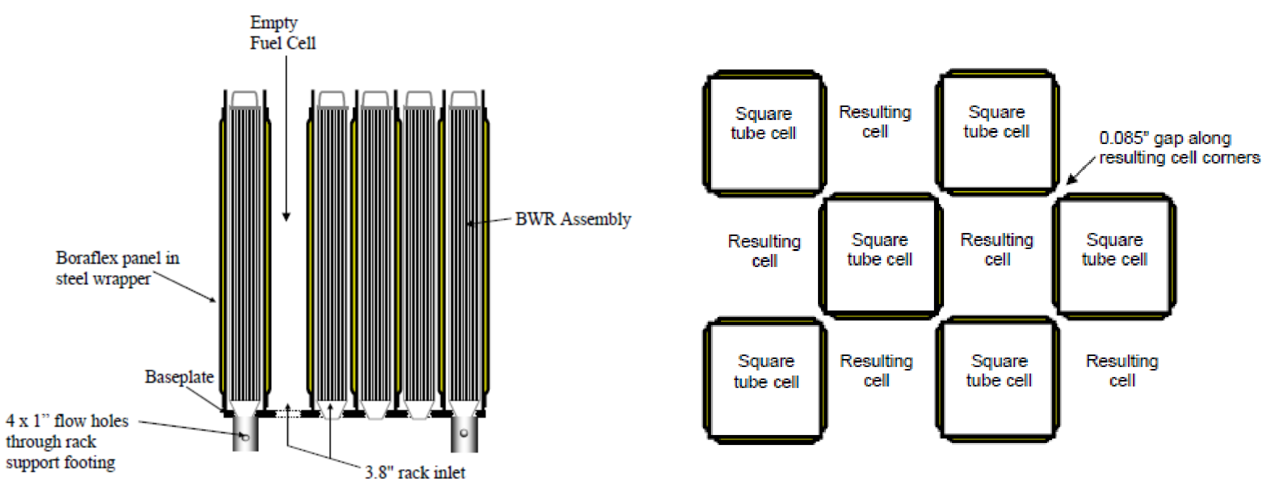

Figure 39 Typical SFP rack cut away cross sections

Figure 40 shows the control volume nodalization of the SFP region of the whole pool model. The bottom of the pool was divided into eight regions. CV299 represents all open regions in the SFP around the racks, including the cask area. The racks are subdivided into the other seven regions. Ring 7 (CV170 and CV171) represents the empty rack cells on the periphery of the SFP. All of the assemblies in the SFP are located in Rings 1 through 6. Each ring with assemblies is further subdivided into 19 control volumes—one control volume below the racks, nine control volumes inside the canister, and nine control volumes 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 41). Similarly, 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 control volume nodalization does not contain a bypass region (between the channel box and rack) as shown on the right side of Figure 41.

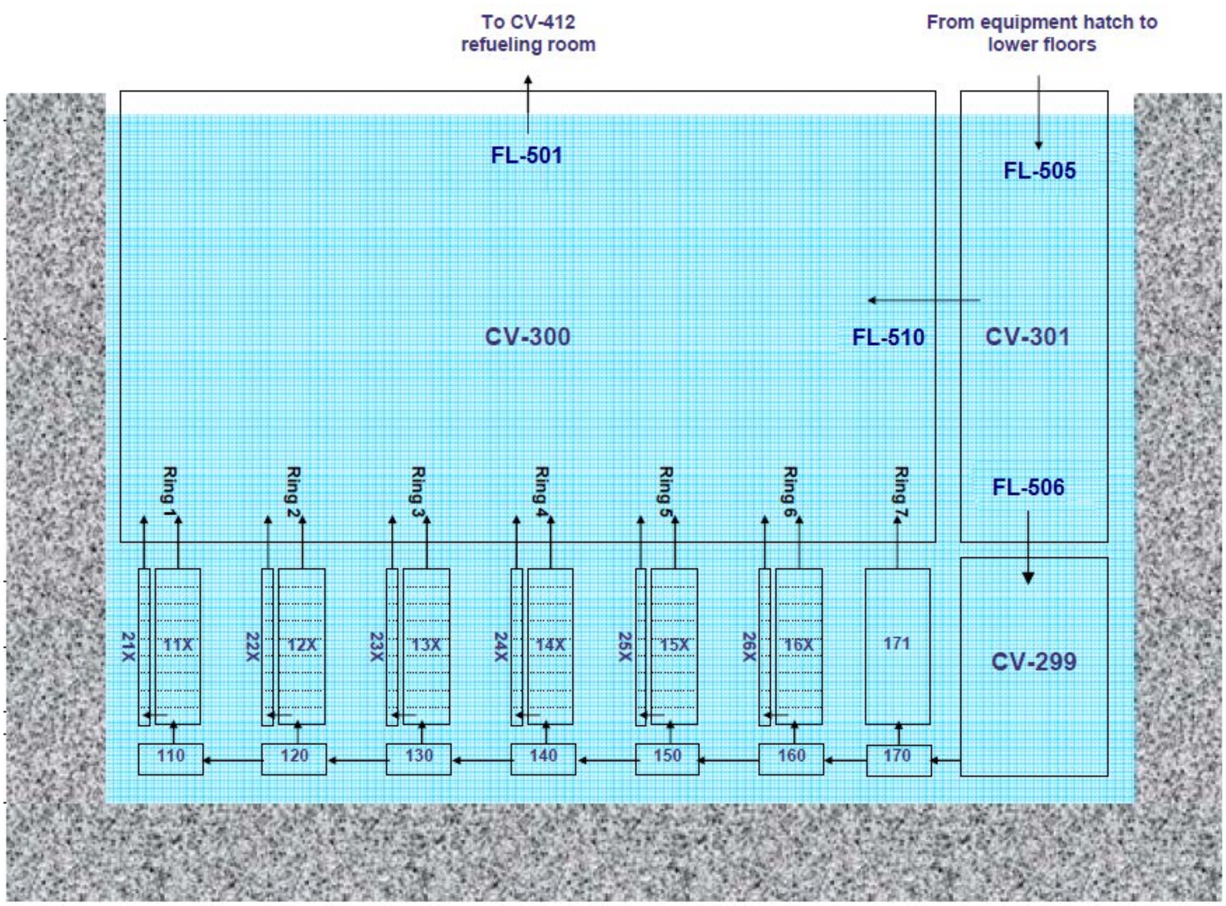

Figure 40 MELCOR nodalization of the whole pool high density model

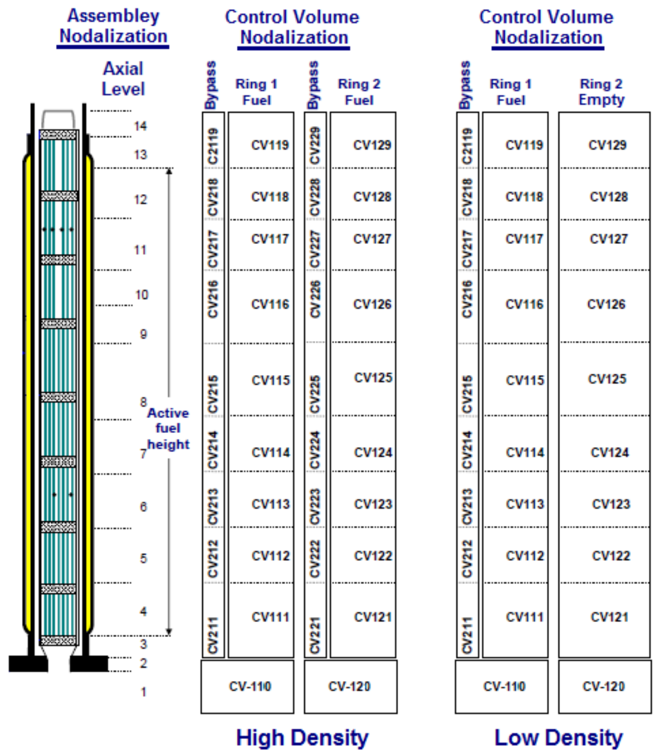

Figure 41 MELCOR nodalization of the assemblies (only two rings shown)

The hydraulic resistance was specified using the results from the SNL experimental test program (Durbin, 2005).[19] For example, for the flowpath connecting CV113 and CV114 in the fully populated region, the MELCOR input values included a form loss coefficient of 3.8, and a friction factor (S_{LAM}) of 31.3 (equal to 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 flowpaths. 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 flowpaths between the bypass region and the assembly. Initially, the canister wall precludes flow. However, if the canister fails, a radial flowpath 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.

19 In the present study, the assembly nodalization is based on the GE14C 10x10 configuration (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 percent compared to the 9x9 design. The hydraulic resistance data are assumed to apply. The frictional loss coefficient for a 10x10 array could be somewhat different since it is a function of hydraulic diameter and grid spaces design.

A complete reactor building has been developed for the reference plant (NRC, 2012d). However, the bulk of the reactor building does not play a significant role in SFP accidents, given that the study does not explicitly model (1) the effect of the SFP accident on reactor systems or (2) specific obstacles to deploying mitigation (e.g., presence of steam on lower elevations). Consequently, the reactor building model was simplified to only model the refueling room (i.e., within the red dashed line in Figure 42).

A single control volume models the refueling bay. An open hatch in the southeast quadrant connects (via a flowpath) 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 overpressure failure flowpaths within the reactor building. The simplified refueling floor model included the two most important flowpaths—(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 overpressure greater than 1,720 pascal (Pa) (0.25 pounds per square inch gauge (psig)). If the reactor building pressure rises above 3,450 Pa (0.5 psig), failure of the roof decking 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 when 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 overpredict the amount of thermal mixing within the building. Based on insights from the computational fluid dynamics calculations for the security assessment work, 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 40) comes directly from the hatch region (see left side of Figure 42). 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 the computational fluid dynamics calculations.

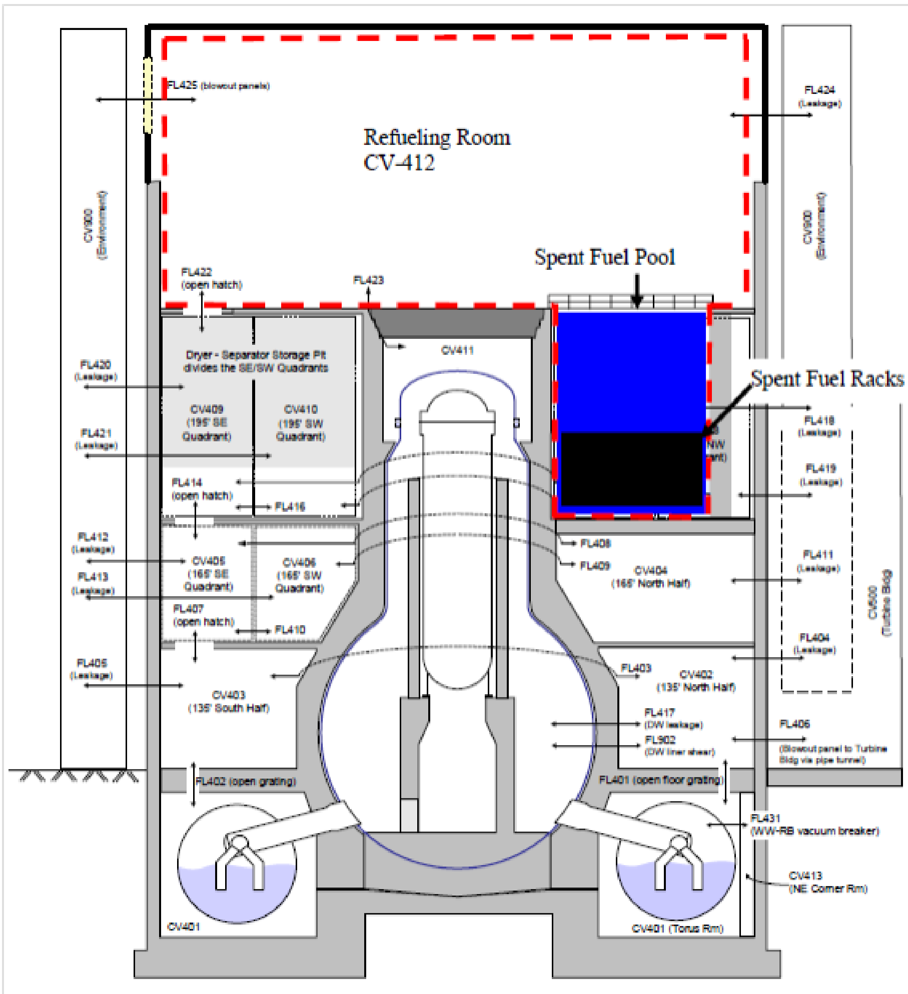

Figure 42 MELCOR reactor building model

6.2.1 High-Density Loading during Outage

During an outage in which the SFP and reactor are hydraulically connected, a single control volume is used to represent both the reactor well and separator/dryer pool, as shown in Figure 43. The total volume of pool in CV601 is about 1,900 m^3 (neglecting the dead-end pool volume of 243 m^3 below the separator/dryer gate elevation). CV601 is hydraulically connected to CV300 (see
Figure 40) using two flowpaths until the water level reaches the SFP gate and no more water can flow into 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 1,400 m^3.

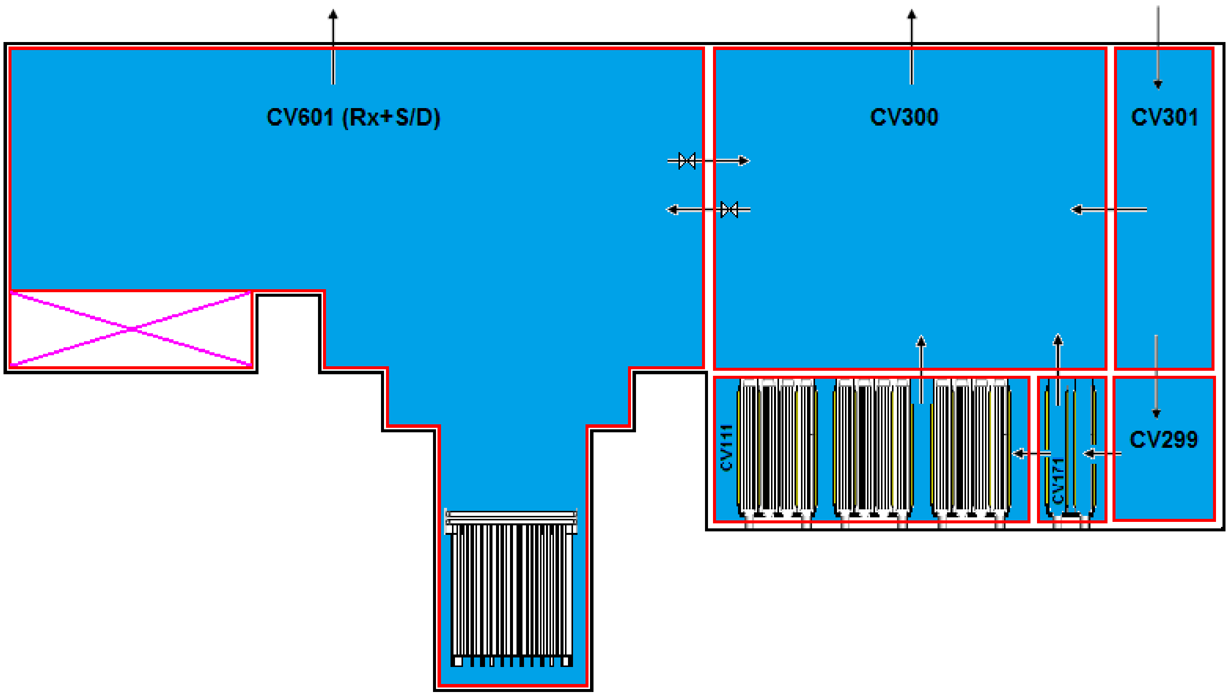

Figure 43 SFP and reactor connection model during outage

For both OCP1 (at 4 days) and OCP2 (at 13 days), CV601 is connected to the detailed model of the SFP (
Figure 40). Figure 44 shows the assembly layout for OCP1 in a 1x4 pattern in which the assemblies are grouped into six 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.[20] Ring 3 is empty during the outage where the assemblies still reside in the reactor.[21] Ring 5 contains the last offload (284 assemblies) with an additional 31 assemblies from previous offloads. Rings 2, 4, and 6 have a total of 2,456 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 41) without the bypass control volume. This will ensure a better representation of flow through the assemblies and modeling of heat transfer between components in various rings.

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20 All of the old assemblies are smeared in MELCOR Rings 2, 4, and 6 (i.e., decay power per assembly is the same).

21 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.

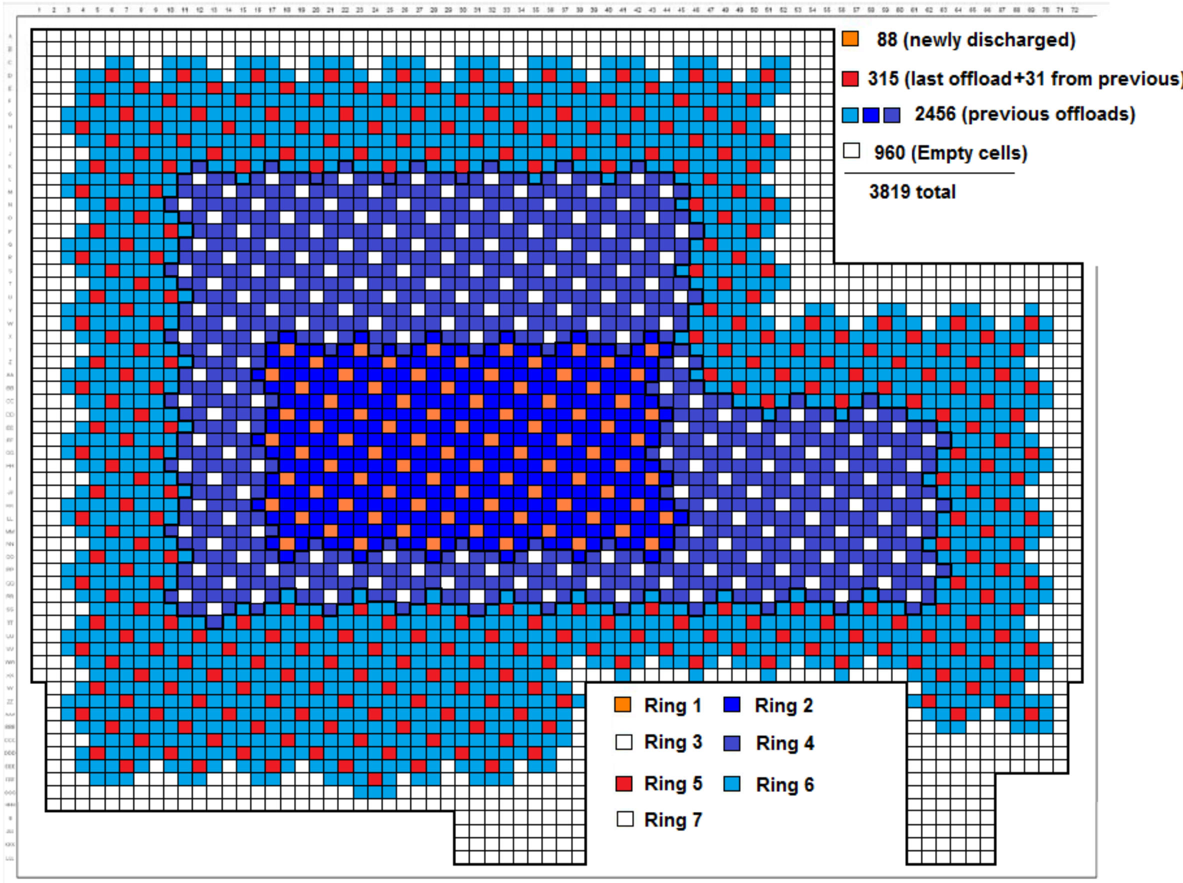

Figure 44 Layout of assemblies for OCP1 high density (1x4) model

Figure 45 shows the cell-wall radiation view factors between the various rings.[22] 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 44, 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, Rings 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.[23] In OCP2, the 196 assemblies have been moved to Ring 3, as shown in Figure 46, and the radial thermal coupling is preserved as in Figure 45.

[22] MELCOR models intracell radiation between concentric rings by default. To disable the radiation model for Rings 2 to 3 and 4 to 5, the radial view factor area is set to zero.

[23] The view factor is assumed to be unity. It should be noted that there is a temperature gradient within each ring, and MELCOR attempts to model a multidimensional geometry with a simplified two-surface radiation model.

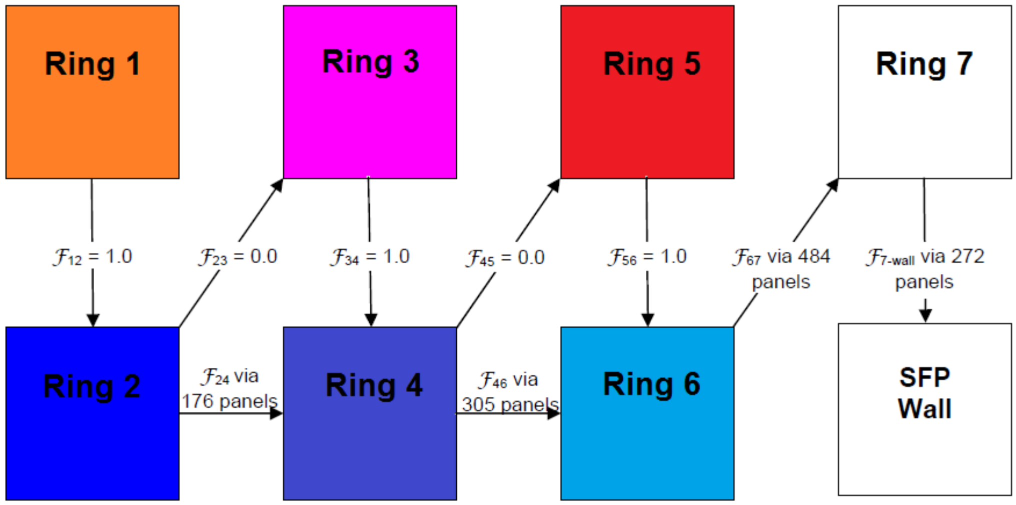

Figure 45 MELCOR radial radiative coupling scheme

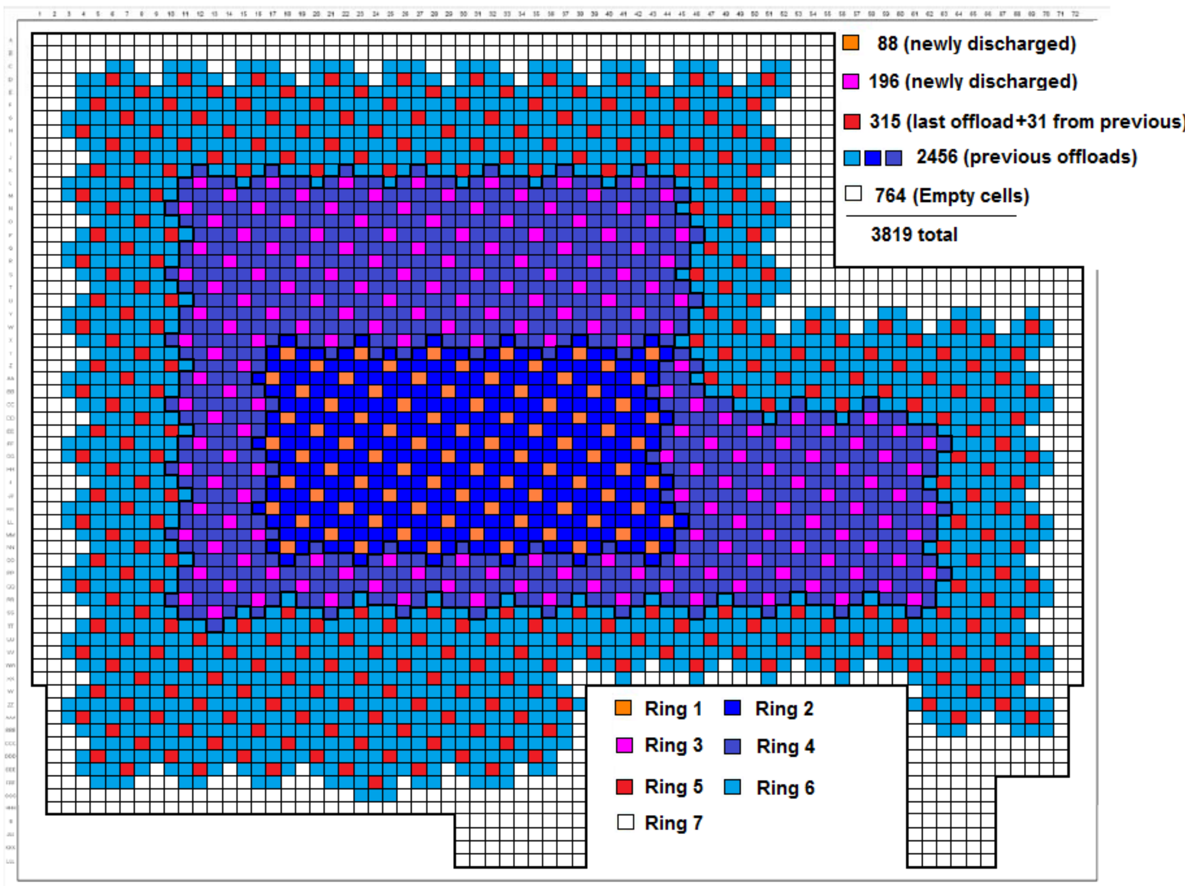

Figure 46 Layout of assemblies for OCP2 high-density (1x4) model

The methodology described in Section 6.1.5 was used to estimate the decay heat power as a function of time for different OCPs. Table 25 shows the results of this analysis. The reactor power was based on the decay power for all assemblies residing in the reference plant reactor (NRC, 2012d) by subtracting the power associated with assemblies that have already been moved to the SFP. For example, for OCP1, the analysis assumed that 88 assemblies are already in the SFP.

Table 25 Distribution of Decay Heat in the Reactor and SFP for High Density Loading

	Reactor (kW)	Spent Fuel Pool (kW)							
		Days	Ring 1 (88)[1]	Ring 3 (0)	Ring 5 (315)	Ring 2 (352)	Ring 4 (784)	Ring 6 (1,320)	Total (2,859)
OCP1	10,216	3.6	1,927	0	465	80	179	301	2,951
	9,915	3.9	1,867	0	452	80	179	301	2,878
	9,006	5.0	1,690	0	417	80	178	300	2,666
	7,406	8.0	1,403	0	358	80	178	300	2,320
	6,710	10.0	1,282	0	334	80	178	300	2,174
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (315)	Ring 2 (352)	Ring 4 (784)	Ring 6 (1,320)	Total (3,055)
OCP2	4,395	13.1	1,144	1,533	332	80	178	300	3,567
	4,117	15.0	1,077	1,444	330	80	178	299	3,409
	3,530	20.0	957	1,294	318	79	176	296	3,120
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (315)	Ring 2 (352)	Ring 4 (784)	Ring 6 (1,320)	Total (3,055)
OCP3		37	720	973	324	79	177	298	2,571
OCP4		107	422	602	301	78	173	292	1,868
OCP5		383	191	315	230	73	162	273	1,245

1. The numbers in parentheses are the number of assemblies.

6.2.2 High-Density Loading Postoutage

The layout for the postoutage high-density loading is similar to OCP2 (see Figure 46). In postoutage, the assemblies are assumed to be in a 1x4 pattern, which applies to OCP3, OCP4, and OCP5. The assembly layout remained constant for these OCPs. However, the decay heat decreased from OCP3 to OCP5 as the aging time since reactor shutdown increased. Table 25 summarizes the decay heat power in each ring.

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 (equal to 284 × 3). 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 47 shows the layout of assemblies in the SFP for OCP1, and Figure 48 shows the layout for OCP2. For both configurations, all of the old fuel has been removed from the pool, and the current offload is in a 1x4 pattern with empties. Because of space limitations, the last two offloads are placed in a checkerboard pattern.[24] For the axial nodalization, Ring 1 contains both the channel (inside the

[24] There is not enough room to place all of the fuel in a 1x4 pattern. The current offload eventually requires 1,420 cells (284 for assemblies and 284 × 4 for empties surrounding them), which would leave only 1,635 cells (excluding Ring 7). The 568 assemblies would require 2,840 cells for storage in a 1x4 pattern.

canister) and the bypass (outside between canister and rack) control volumes, while both volumes are combined for Ring 2 (see Figure 41). The basic radial thermal coupling from Figure 45 still applies, but the boundary area from Ring 6 to Ring 7 is 472 panels. For modeling convenience, Rings 2, 4, and 6 from the high-density layout are still present, but the cells contain only the rack component.

Table 26 provides the distribution of decay heat in the pool. A comparison with the high-density decay heat shows that the total decay heat in the pool for the low-density case is reduced by less than 20 percent. 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 zirconium fire because of reduced mass in the empty assemblies.

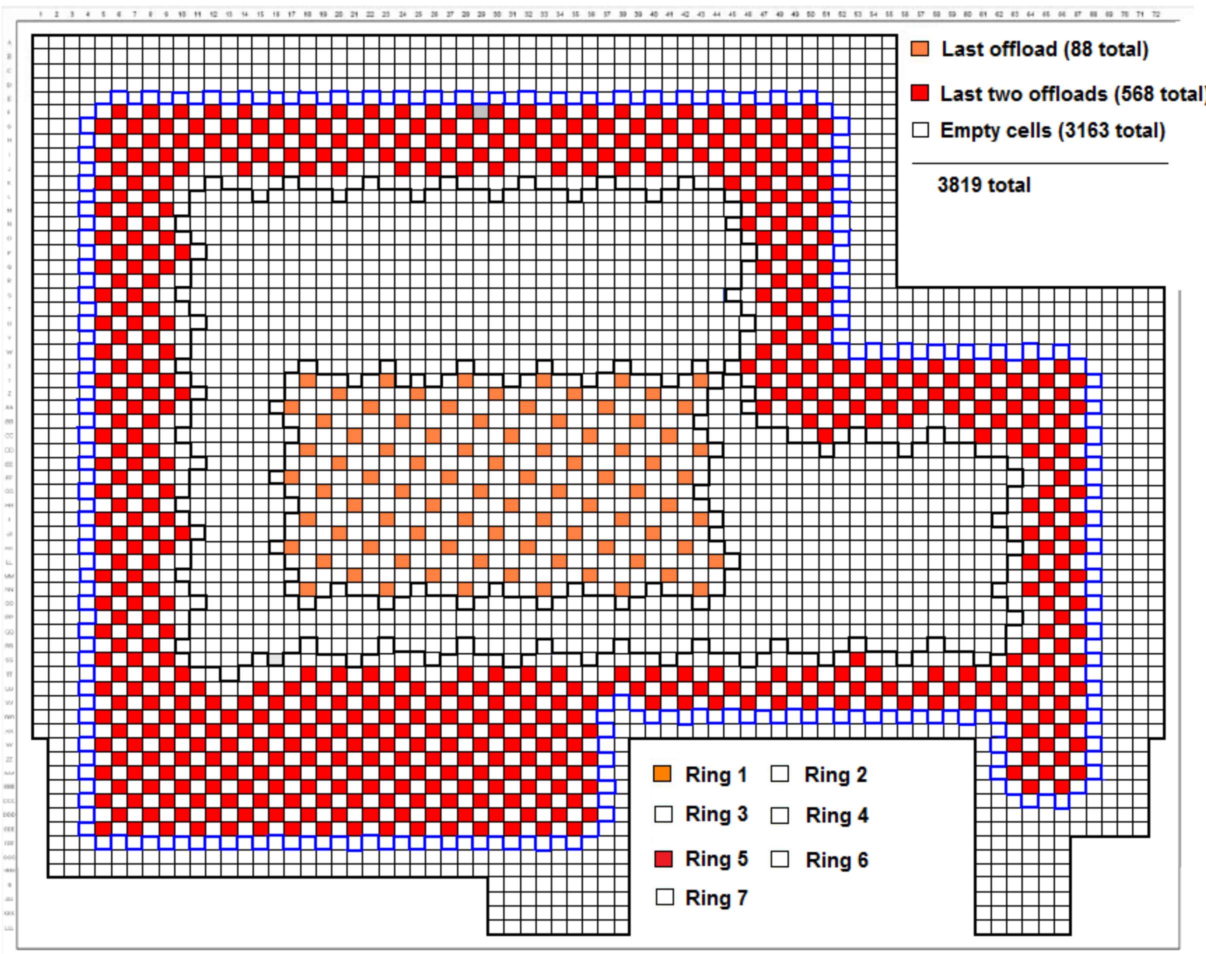

Figure 47 Layout of assemblies for OCP1 low-density model

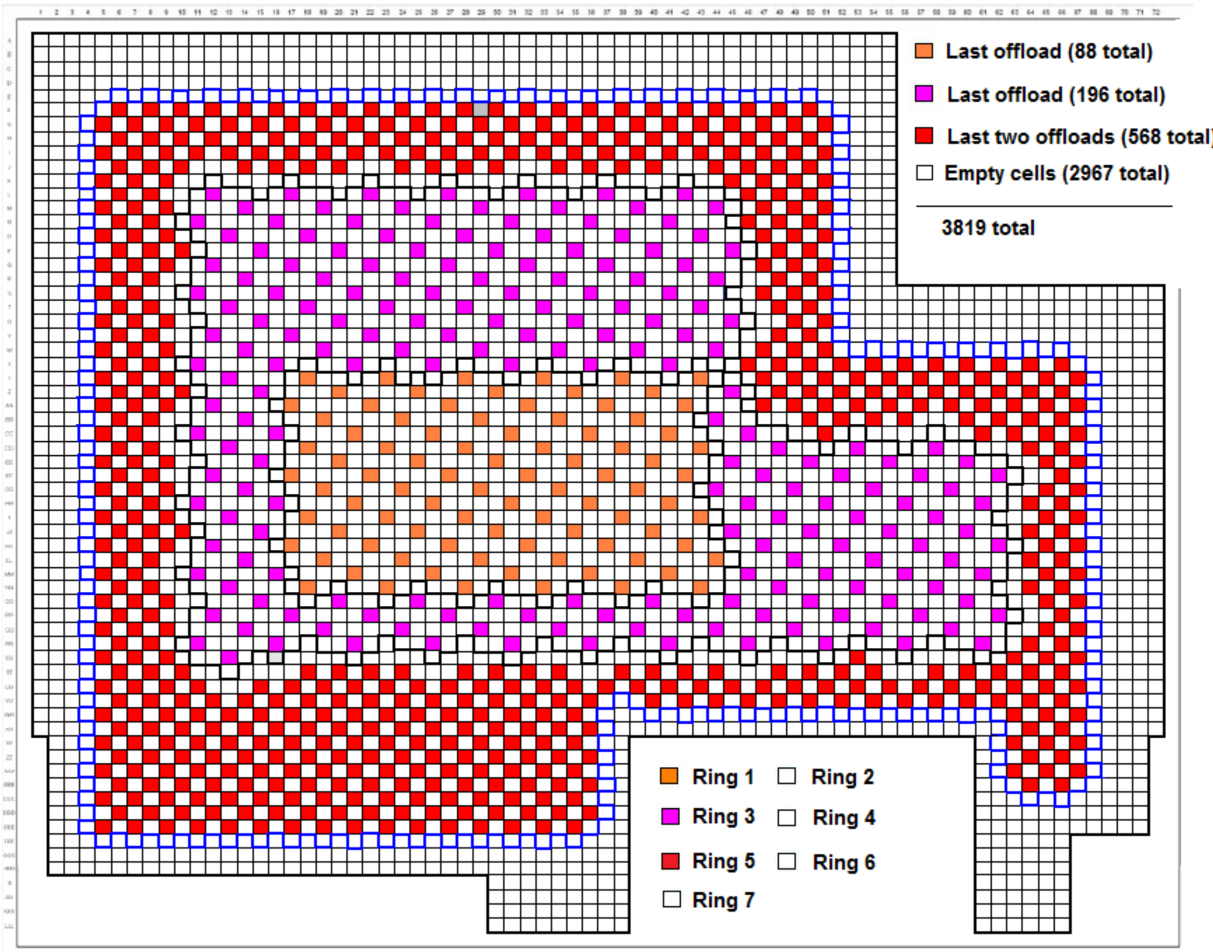

Figure 48 Layout of assemblies for OCP2 low-density model

6.2.4 Low-Density Loading Postoutage

The postoutage low-density layout for OCP3, OCP4, and OCP5 is identical to OCP2 (see Figure 48), and the pool decay heat is provided in Table 26.

Table 26 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)
OCP1	10,216	3.6	1,927	0	599	0	0	0	2,526
	9,915	3.9	1,867	0	587	0	0	0	2,454
	9,006	5.0	1,690	0	551	0	0	0	2,241
	7,406	8.0	1,403	0	492	0	0	0	1,895
	6,710	10.0	1,282	0	468	0	0	0	1,750
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (568)	Ring 2 (0)	Ring 4 (0)	Ring 6 (0)	Total (852)
OCP2	4,395	13.1	1,144	1,533	466	0	0	0	3,143
	4,117	15.0	1,077	1,444	464	0	0	0	2,985
	3,530	20.0	957	1,294	448	0	0	0	2,699
		Days	Ring 1 (88)	Ring 3 (196)	Ring 5 (568)	Ring 2 (0)	Ring 4 (0)	Ring 6 (0)	Total (852)
OCP3		37	720	973	455	0	0	0	2,149
OCP4		107	422	602	427	0	0	0	1,451
OCP5		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, the following four classes of scenarios do not result in a release from the fuel:

1. boiloff scenarios with no SFP leaks
2. mitigated scenarios for small leaks
3. unmitigated scenarios in late phases (OCP4, OCP5)
4. mitigated moderate leak scenarios in OCP2, OCP3, OCP4, and OCP5

Boiloff

For the boiloff scenarios, a simplified model was used to estimate the pool heatup and water level drop. Figure 43 shows this model in which all of 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 25 for high-density cases or Table 26 for low-density cases. The top of the pool is connected to the reactor building (see Figure 42) in the same manner as in the detailed model. This simplified model is used as a screening tool to determine whether more detailed analysis is needed. Figure 49 shows the water level as a function of time for both high- and low-density cases for

OCP1, OCP2, OCP3, and OCP4.[25] Figure 49 also identifies the time required to reach pool saturation. For cases in the same OCP, the high-density cases become saturated sooner since there is less water volume and more decay heat. In the late OCPs following refueling, the difference in the timing directly correlates to the decay heat power. While there are differences in postsaturation water level for OCP3 and OCP4, the water level for OCP1 and OCP2 is similar as a result of mixing assumed between the reactor well water and the SFP water (see Figure 43). For the OCP4 low-density and OCP5 cases, the SFP never becomes saturated in 72 hours. The slight water level increase during the sensible heating period results from the change in pool density as the water heats up. The analysis shows that there is 4.6 m (15 ft) of water above the top of racks in OCP1 at 72 hours.

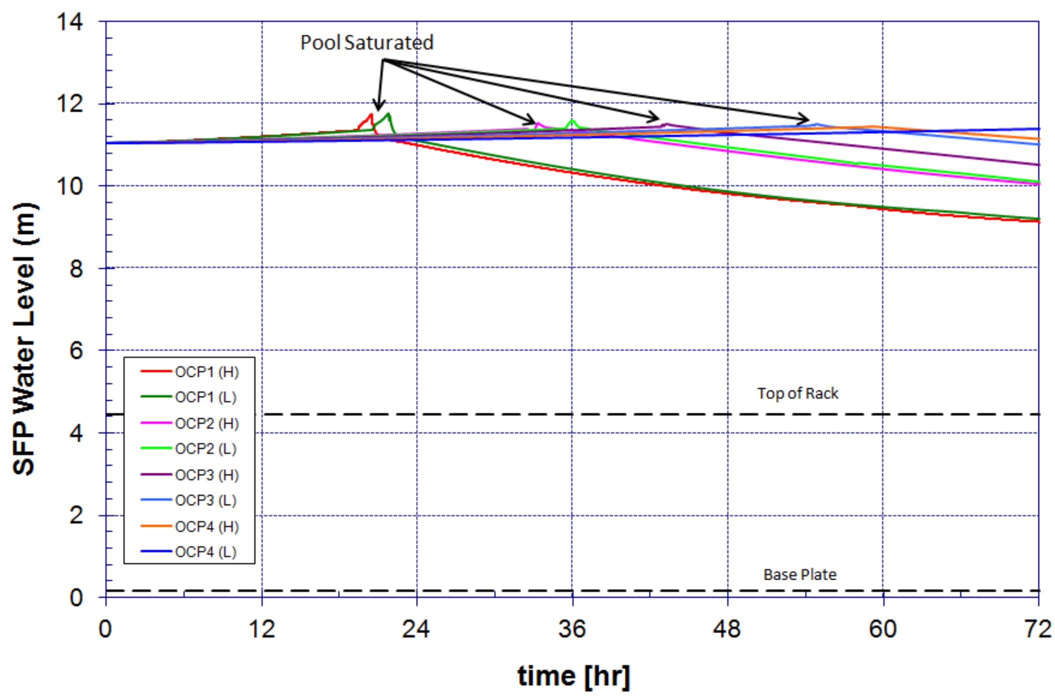

Figure 49 Water level for boiloff scenarios

<u>Mitigated Scenarios (Small Leaks)</u>

The small leak is modeled in MELCOR with a 4.4 cm (1.75-in.) 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). Figure 50 and Figure 51, respectively, show the water level and the injection and leak mass flow rates for the low-density OCP1 case. 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 case, mitigation is direct makeup to the pool

25 The initial water level is assumed to be 11 m. The initial water temperature is 82 degrees F (28 degrees C). Both these initial conditions are applied to all accident scenarios in this report. Based on a teleconference with the licensee held on April 24, 2012, this is the postoutage water temperature under steady-state conditions where the heat exchangers are working (prior to postulated accident). During an outage (OCP1 and OCP2), the water temperature could vary between approximately 80 degrees F and 100 degrees F. The higher temperature affects the sensible heating of the pool and is not expected to change the overall conclusion of boiloff scenarios given the significant margin observed.

(injection) since the water level at the time of deployment is more than 1 m above the top of the racks. For this small leak, the initial water flow rate is about 250 gpm (0.016 cubic meters per second (m^3/s)), 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 leak 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 configuration as long as the water level remains above the top of the racks.

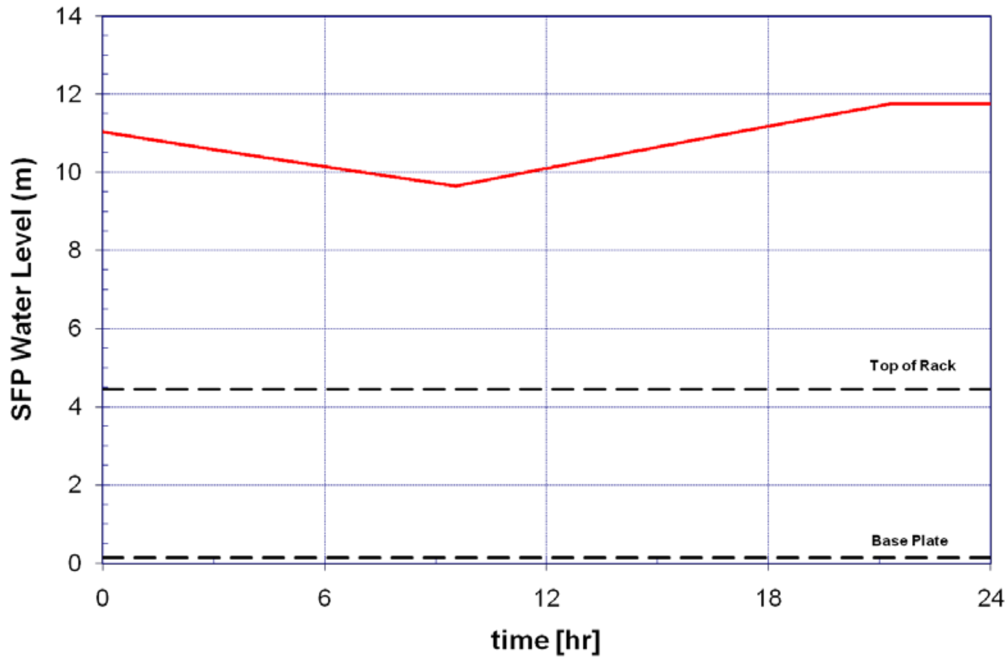

Figure 50 Water level for mitigated low-density OCP1 (small leak) scenario

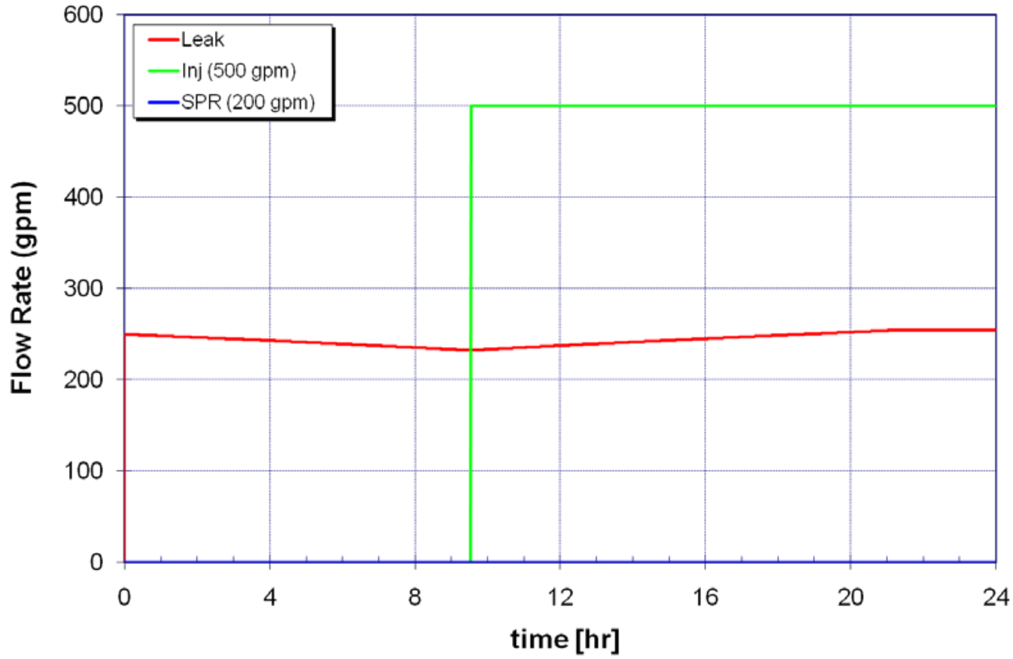

Figure 51 Flow rates for mitigated low-density OCP1 (small leak) scenario

Unsuccessful Deployment of Mitigation for OCP4 and OCP5 Scenarios

For OCP4, the decay heat is between 37 percent to 48 percent lower than for OCP3. None of the scenarios in OCP4 or OCP5 leads to a release from the fuel.[26] Figure 52 through Figure 55 illustrate the thermal-hydraulic response of the high-density pool to a small leak and a moderate leak. It takes less than 6 hours to clear the rack baseplate and initiate airflow for the moderate leak, while for the small leak case, the rack baseplate does not clear until about 39 hours. In both cases, there is a heatup 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 heatup is slower since 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 56 or Figure 57). 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 the total decay heat in the pool for low-density case is only 77 percent of the high-density case, the decay heat in Ring 1 is identical in both cases.

[26] The start of the release of radionuclides from the fuel is modeled based on a temperature of 900 degrees C (1,173 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.

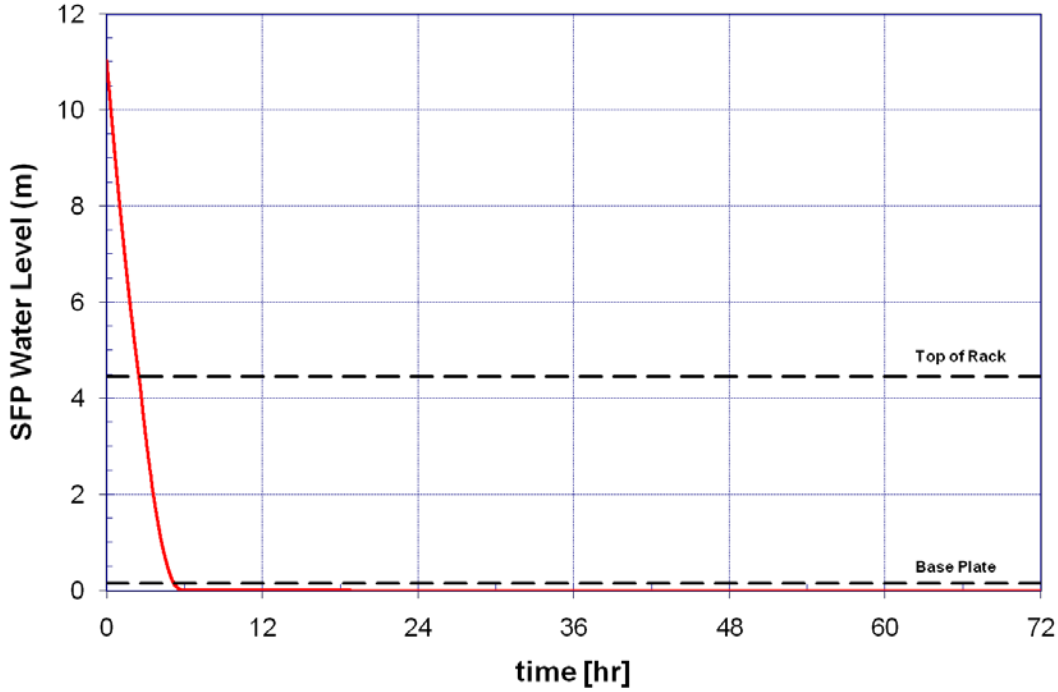

Figure 52 Water level for unmitigated high-density moderate leak (OCP4)

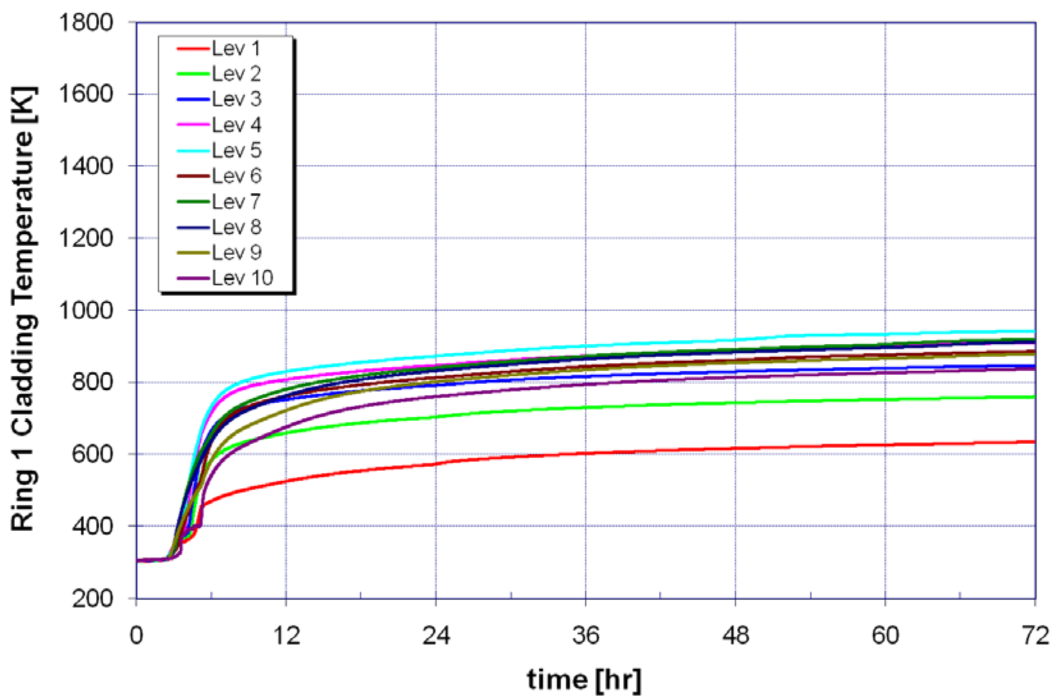

Figure 53 Ring 1 temperature for unmitigated high-density moderate leak (OCP4)

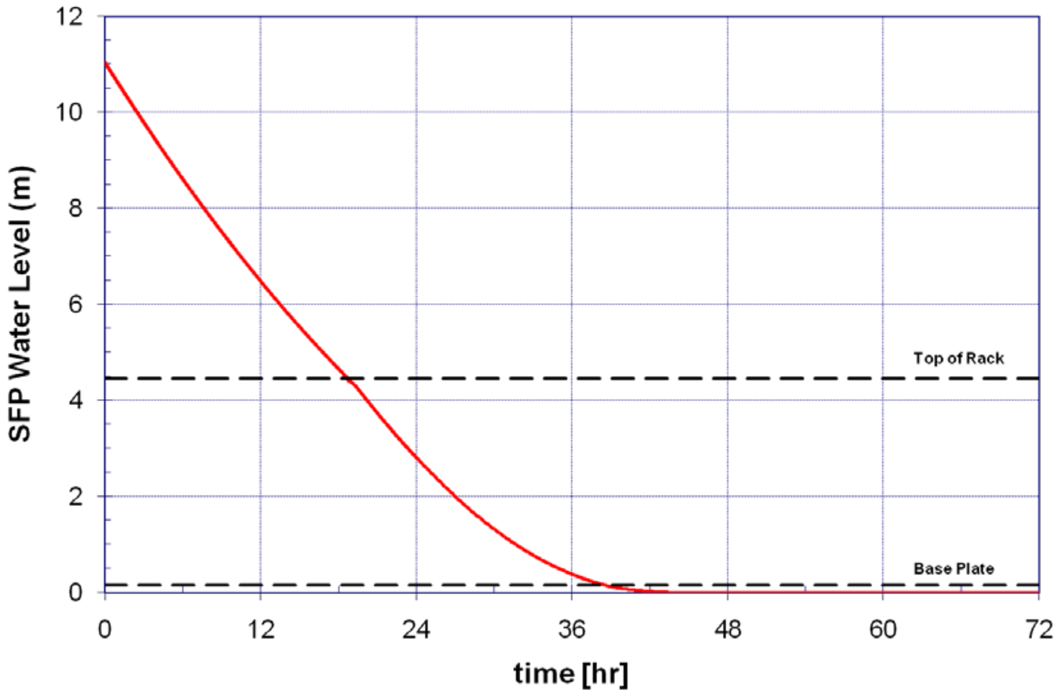

Figure 54 Water level for unmitigated high-density small leak (OCP4)

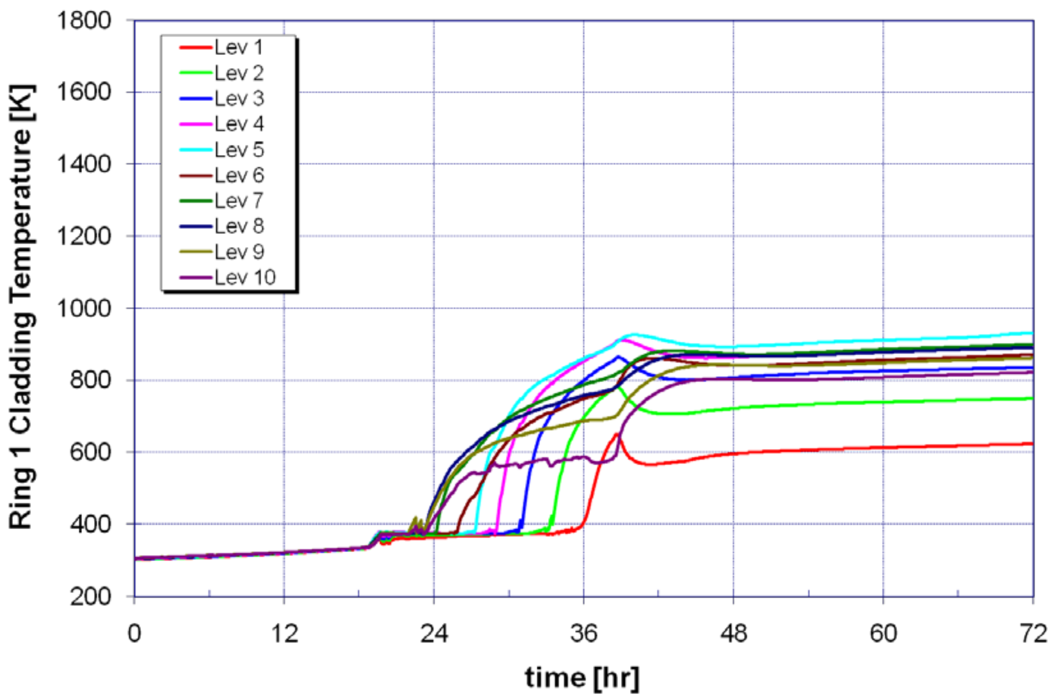

Figure 55 Ring 1 temperature for unmitigated high-density small leak (OCP4)

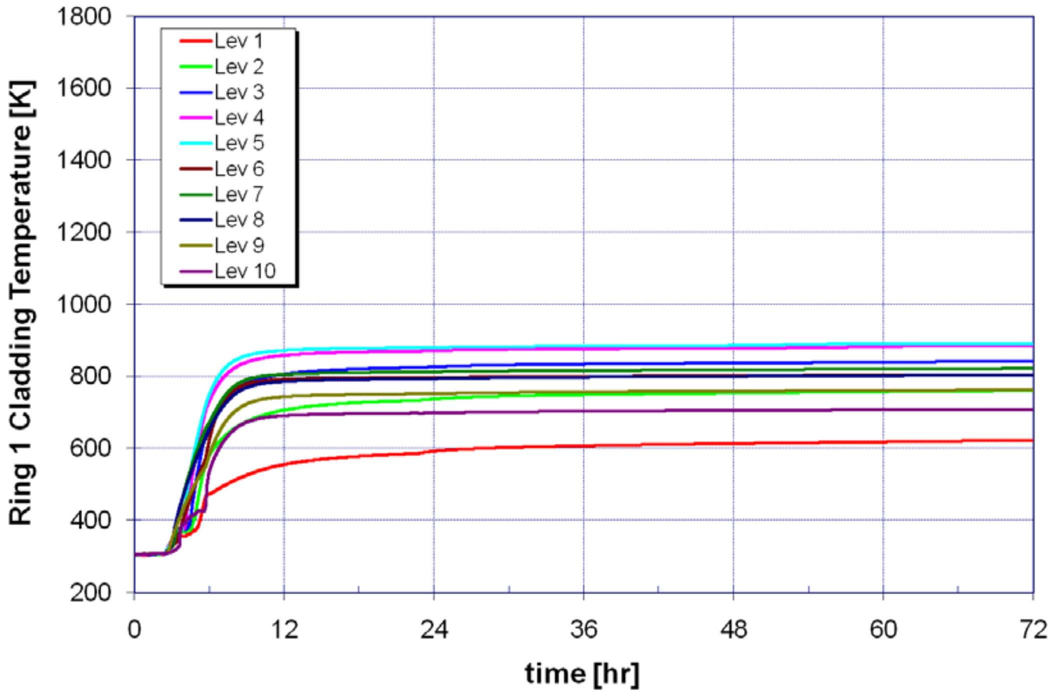

Figure 56 Ring 1 temperature for unmitigated low-density moderate leak (OCP4)

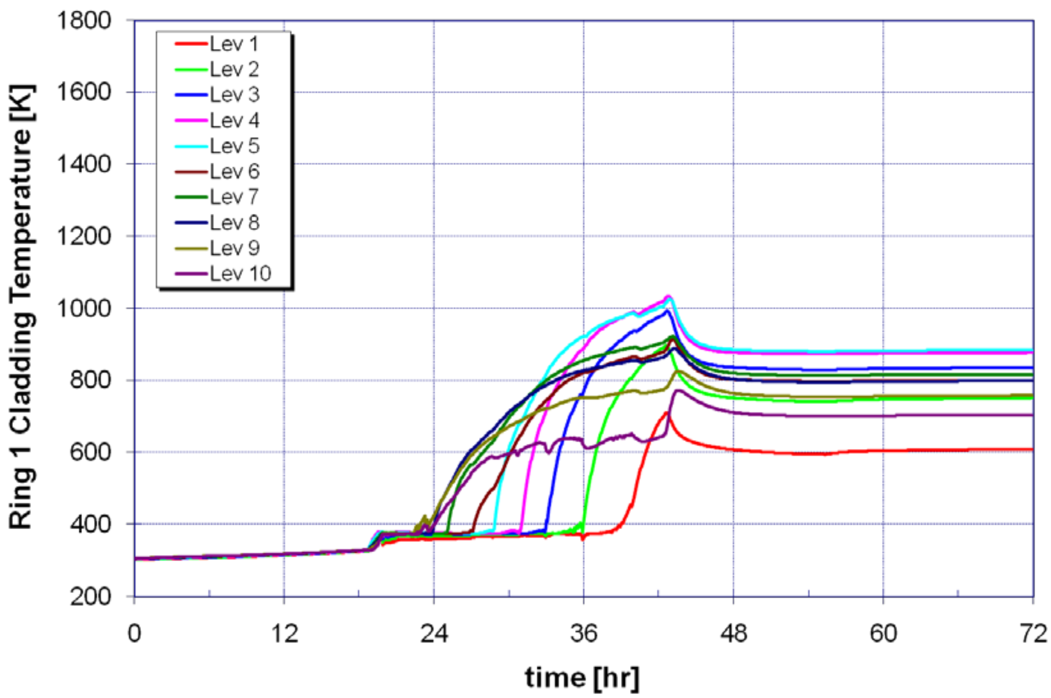

Figure 57 Ring 1 temperature for unmitigated low-density small leak (OCP4)

Mitigated Moderate Leak Scenarios in OCP2, OCP3, OCP4, and OCP5

Mitigation for moderate leak cases involves actuation of the sprays for the postoutage scenarios (OCP3, OCP4, and OCP5) and direct injection in OCP1 and OCP2. The moderate leak is modeled in MELCOR with a 11.4-cm- (4.5-in.-) 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). Section 6.1.3 of this report discussed the MELCOR modeling of the sprays and presented two modeling options (i.e., simple flow regime model on or off). Only high-density OCP3 results[28] are presented since the unmitigated scenarios in later phases do not lead to release, and the moderate leak size is large enough to avoid the baseplate blockage resulting from quasi-steady water level at the bottom of the pool in response to the 200-gpm (0.013-m^3/s) spray water. The results of the OCP2 calculation showed no release from the fuel resulting from various heat transfer mechanisms (see also discussion for OCP1 in Section 6.3.2 of this report).

Figure 58 shows the water level for the moderate leak, high-density OCP3 scenario. Because of the spray activation at 3 hours (see Figure 59), the bottom of the racks clears for natural circulation airflow more than 1 hour later compared to an unmitigated case (see Figure 52). 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 less than 200 gpm (0.013 m^3/s) in Figure 59 because of heat transfer from spray droplets to the atmosphere and fuel rods.[29] Figure 60 shows the response of the clad in Ring 1 for the case in which the simple flow regime model is active. As expected, the top cells experience more cooling as there is more water coverage. The temperatures reach a quasi-steady state by about 10 hours[30] and the maximum clad temperature is about 850 K. Figure 61 shows the clad temperatures for the case in which 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 the 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 differ, the maximum clad temperatures are almost the same and well below the gap release criterion. This is partially because of the importance of the heat removal by natural circulation of air through the racks. If there was no natural circulation of air through the racks, the cooling of the fuel by the spray flow (i.e., modeled with the simple flow regime map) would be the only effective cooling mechanism, and therefore 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 an additional 3-hour delay in the actuation of the spray as shown in Figure 62.[31] Both Figure 63 and Figure 64 show that (for OCP3), following the initial heatup 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.

[28] The low-density case is similar to the high-density case, 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.

[30] The calculation fails shortly after 10 hours from numerical problems.

[31] These cases were actually run based on an earlier logic for spray actuation that assumed a 3-hour additional delay at the end of deployment.

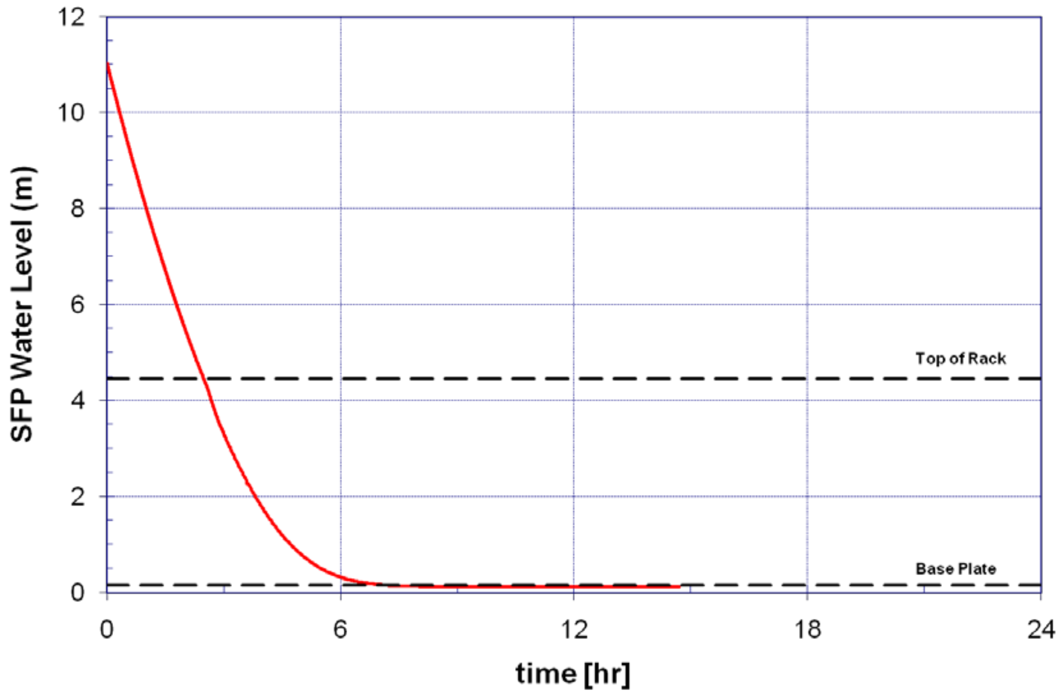

Figure 58 Water level for mitigated high-density moderate leak (OCP3)

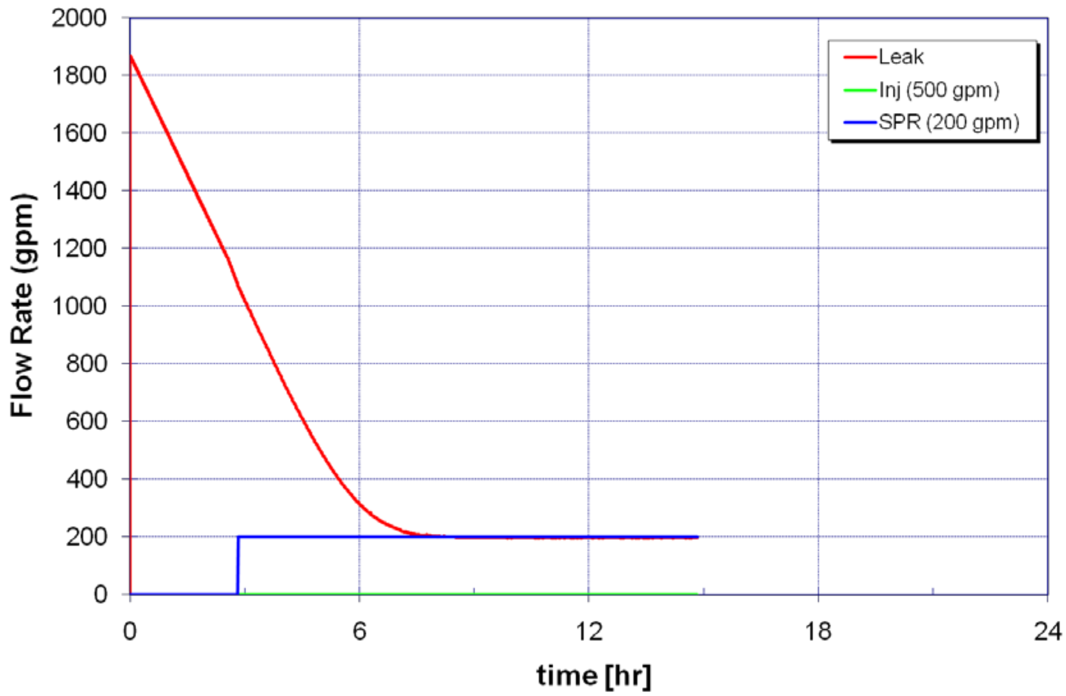

Figure 59 Water flow rates for mitigated high-density moderate leak (OCP3)

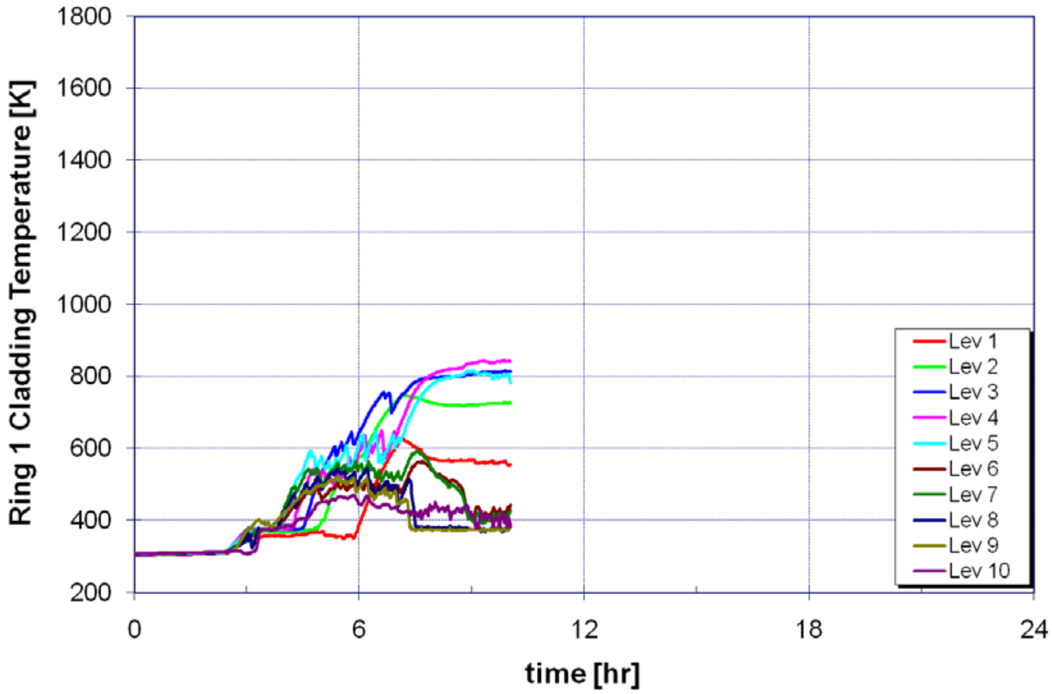

Figure 60 Ring 1 clad temperatures for mitigated (simple flow regime active) high-density moderate leak (OCP3)

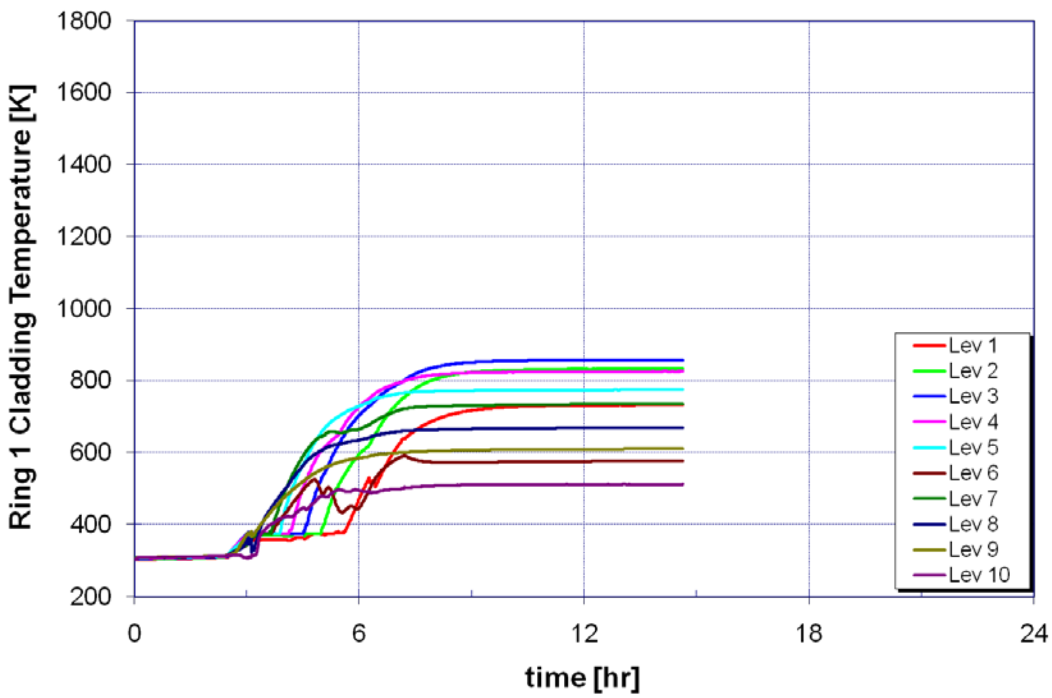

Figure 61 Ring 1 clad temperatures for mitigated (simple flow regime inactive) high-density moderate leak (OCP3)

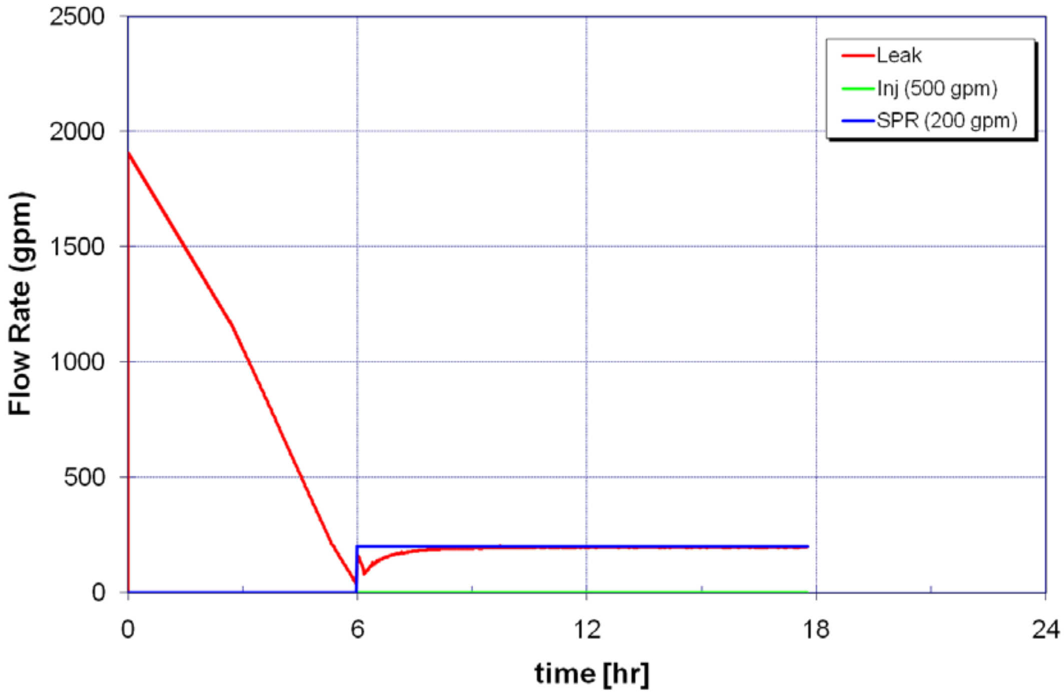

Figure 62 Flow rates for mitigated high-density moderate leak (OCP3) with late actuation of sprays

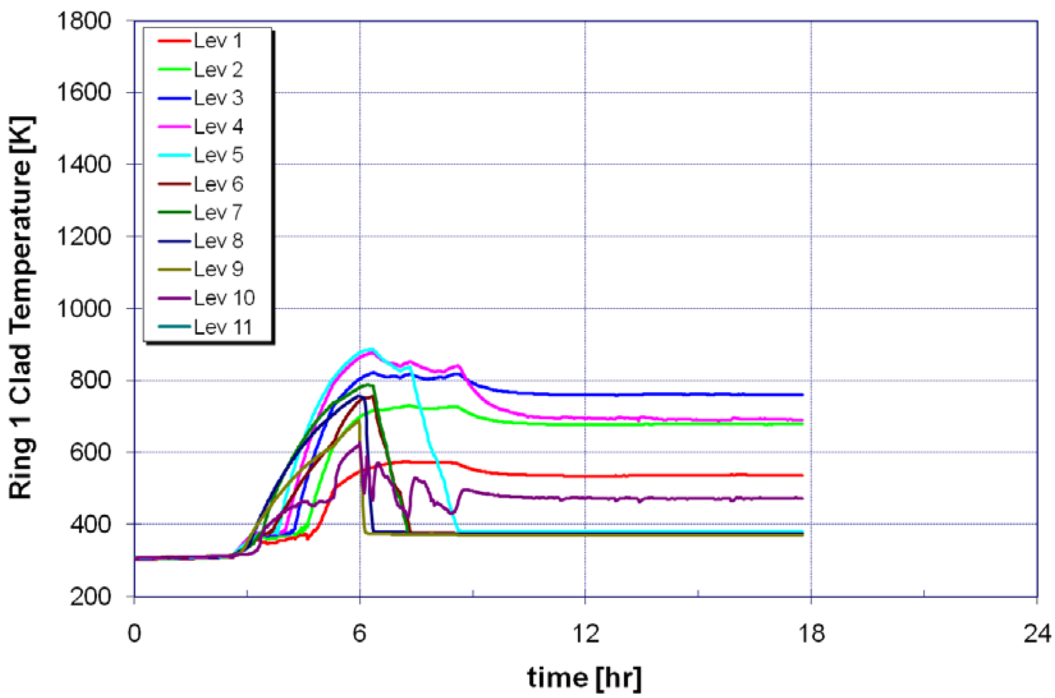

Figure 63 Ring 1 clad temperatures for mitigated (simple flow regime active) high-density moderate leak (OCP3) with late actuation of sprays

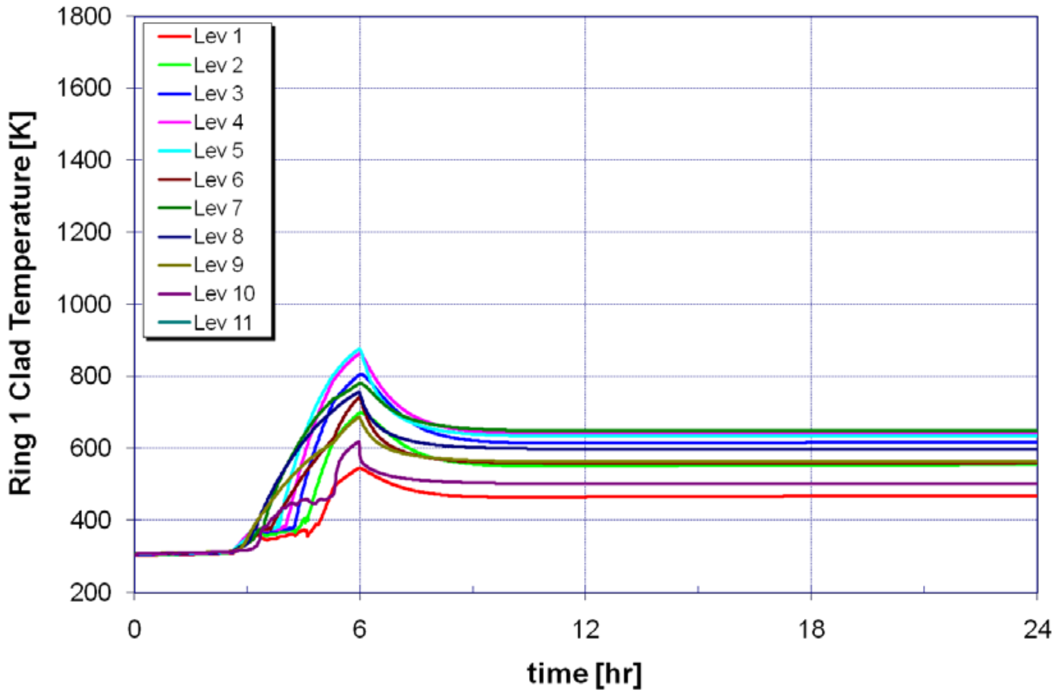

Figure 64 Ring 1 clad temperatures for mitigated (simple flow regime inactive) high-density moderate leak (OCP3) with late actuation of sprays

6.3.2 Sequences That Do Lead to a Release

All the sequences in OCP1, OCP2, and OCP3 lead to release without successful deployment of mitigation. This section will only discuss representative scenarios 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 42), 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-percent concentration if there is adequate oxygen (oxygen concentration is greater than or equal to 5 percent) and no steam inerting (steam concentration is less than or equal to 55 percent). The analysis considered the sensitivity of the ignition assumptions and potential for reactor building refueling bay failure on a case-by-case basis (see Section 9.1 of this report).

<u>Unsuccessful Deployment of Mitigation for Moderate Leak (OCP1) Scenario</u>

The water level for the high-density scenario (Figure 65) 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 postoutage scenarios (see Figure 52) because of the additional water in the reactor well connected to the SFP. The reactor power (Figure 66) 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 67) 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 zirconium fire that starts near the top of the full rod region (see Figure 41) and propagates downward. The heatup in Ring 1 (fuel, cladding, canister, and racks) propagates to Ring 2 assemblies (Figure 68) 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 and 14.2 hours, the channel boxes in Rings 1 and 2 fail, which 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 heat up 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 zirconium fire initiates at the top of the fuel in Ring 4 at about 42 hours in the upper levels which then propagates downward. The second heatup is more intense and involves the other rings as indicated by both the oxidation power (Figure 66) and the clad temperatures in the outer rings (Figure 68).

OCP1 had a relatively rapid draindown in which an air natural circulation flow developed through the racks before 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 concentration in the refueling floor was only 5 percent, 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 refueling bay, which remained 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 42). The reactor building DF is shown in Figure 71.[34] Aerosols also begin to deposit inside the building and the DF for cesium and iodine aerosols remains between 3 to 4 for much of the accident. The DF 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 DF to arrive at the environmental release for each ring. MELCOR mechanistically models all deposition mechanisms; however, because of the mixing within the reactor building, only an overall DF can be defined for all rings.

Figure 72 depicts the cesium environmental release fraction for individual rings. 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 68. The later releases result from the second heatup and involve all of the outer rings (Ring 3 is empty for OCP1). The total release fraction is the input to the MACCS2 code for consequence analysis and is defined by Equation

[32] The initial heatup of the liner is caused by 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 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 preclude steam oxidation.

[34] The integral DF 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.

11.[35] The DF is a dynamic quantity as the outer rings start to release (see the fluctuations in Figure 71); therefore, care is taken to allow the earlier releases from inner rings preserve their release history so that the total release fraction does not decrease at any time as the release progresses.

Figure 73 illustrates the results of the low-density case. Comparing the heatup with the high-density case (see Figure 67), 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, which results in slower heatup of Ring 5, as shown in Figure 74. Even though the racks fail in this low-density case, the canisters remain intact and the zirconium fire moves down initially and then upwards, as shown in Figure 73. The cesium environmental release fraction for Ring 1 shown in Figure 75 is comparable to the high-density case (Figure 72), but since no release occurs from the older assemblies, the total release fraction for the low-density case is lower.

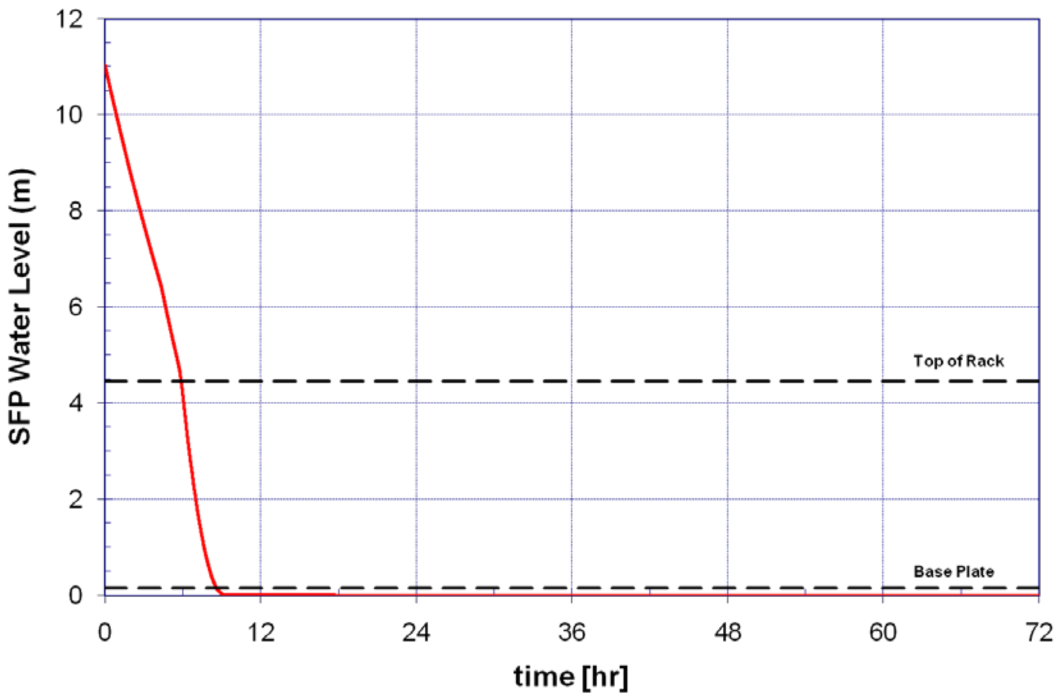

Figure 65 Water level for unmitigated high-density moderate leak (OCP1)

[35] This activity-weighted release 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 is smaller compared to Ring 1.

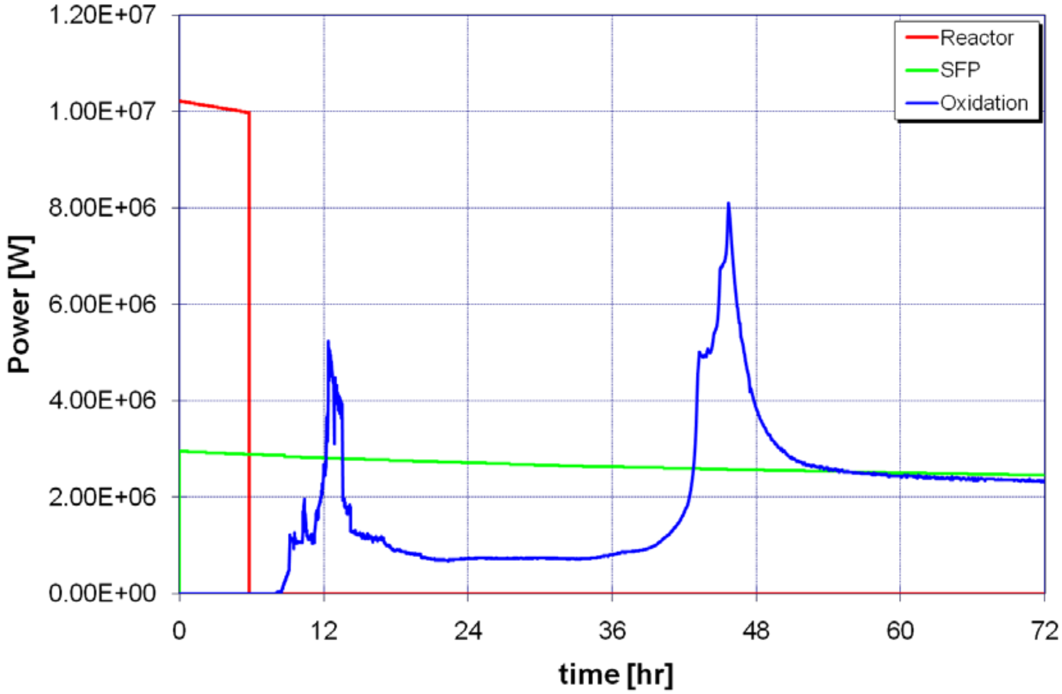

Figure 66 SFP power for unmitigated high-density moderate leak (OCP1)

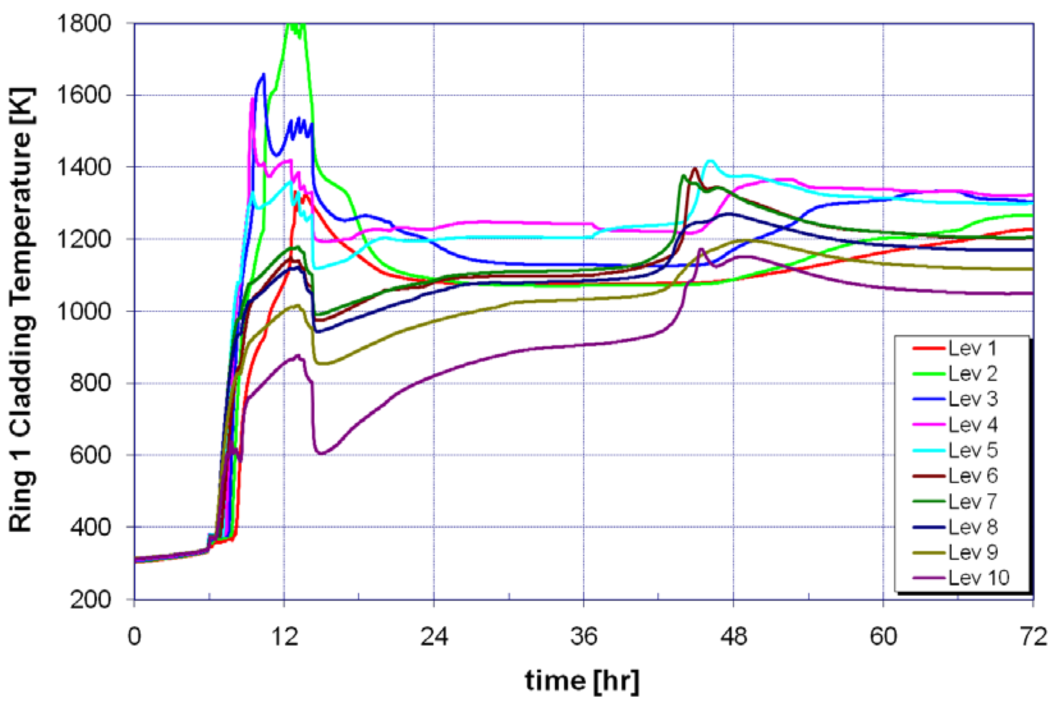

Figure 67 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP1)

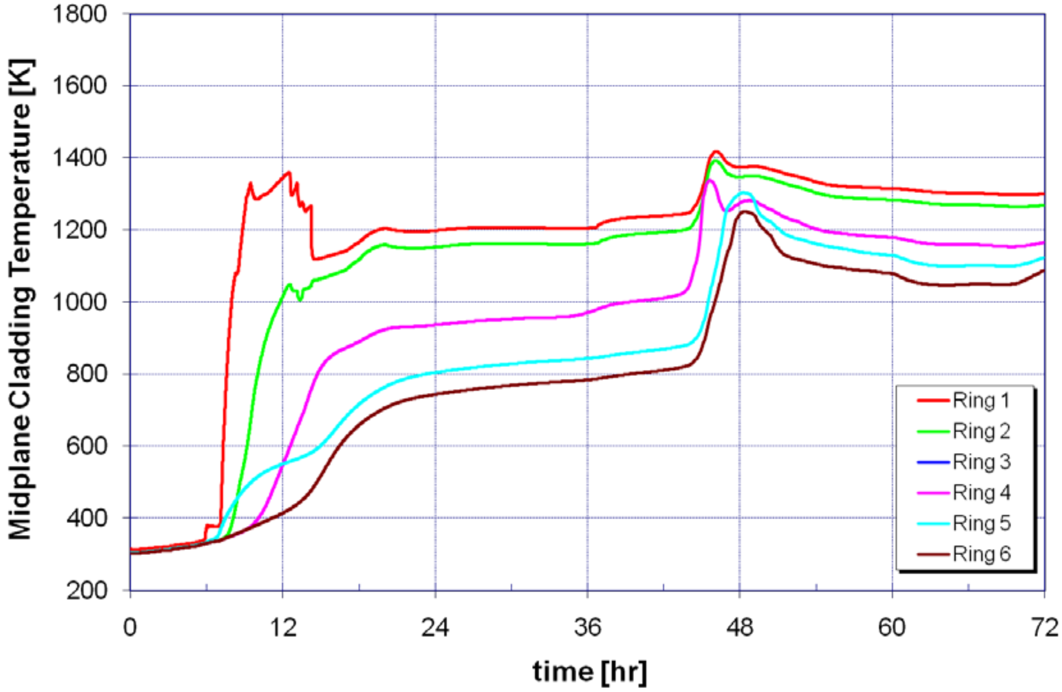

Figure 68 Midplane clad temperature for unmitigated high-density moderate leak (OCP1)

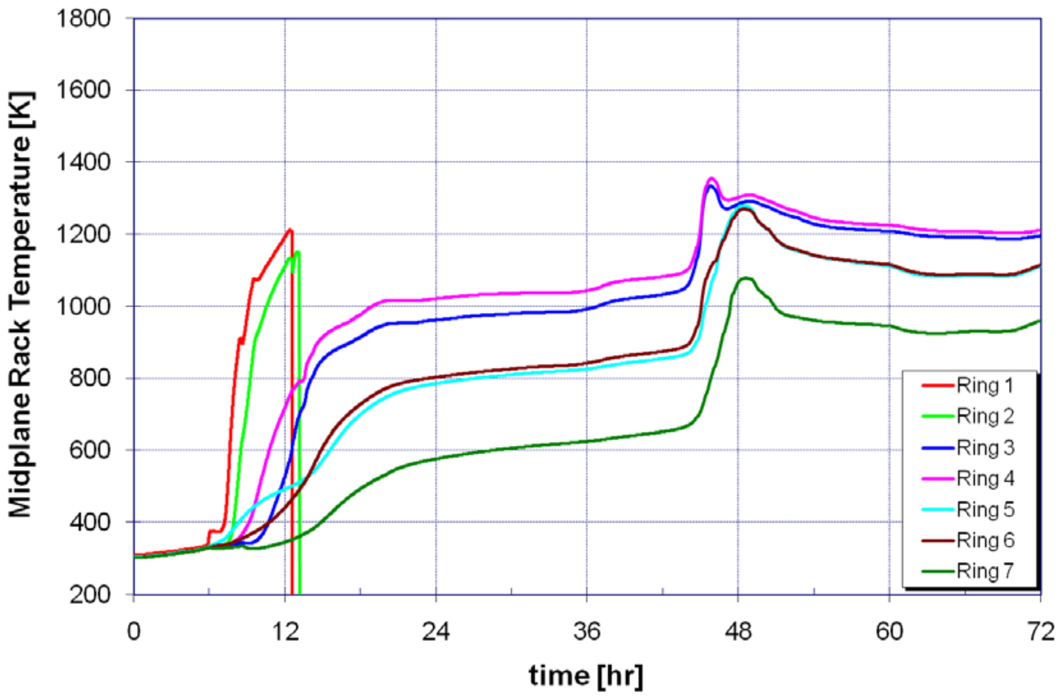

Figure 69 Midplane rack temperature for unmitigated high-density moderate leak (OCP1)

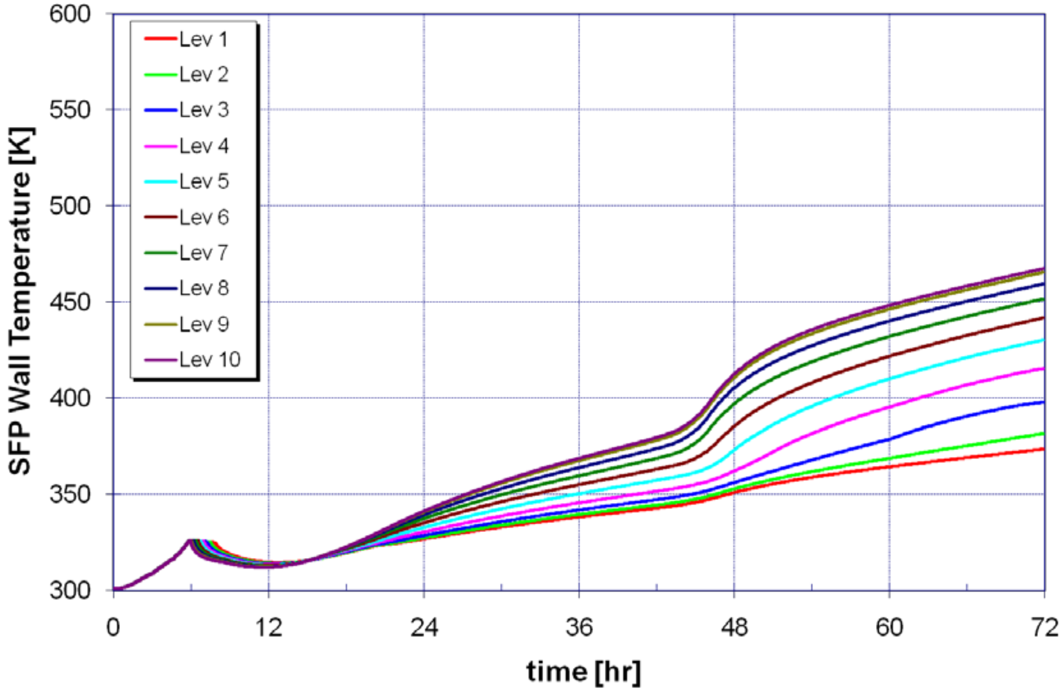

Figure 70 SFP wall liner temperature for unmitigated high-density moderate leak (OCP1)

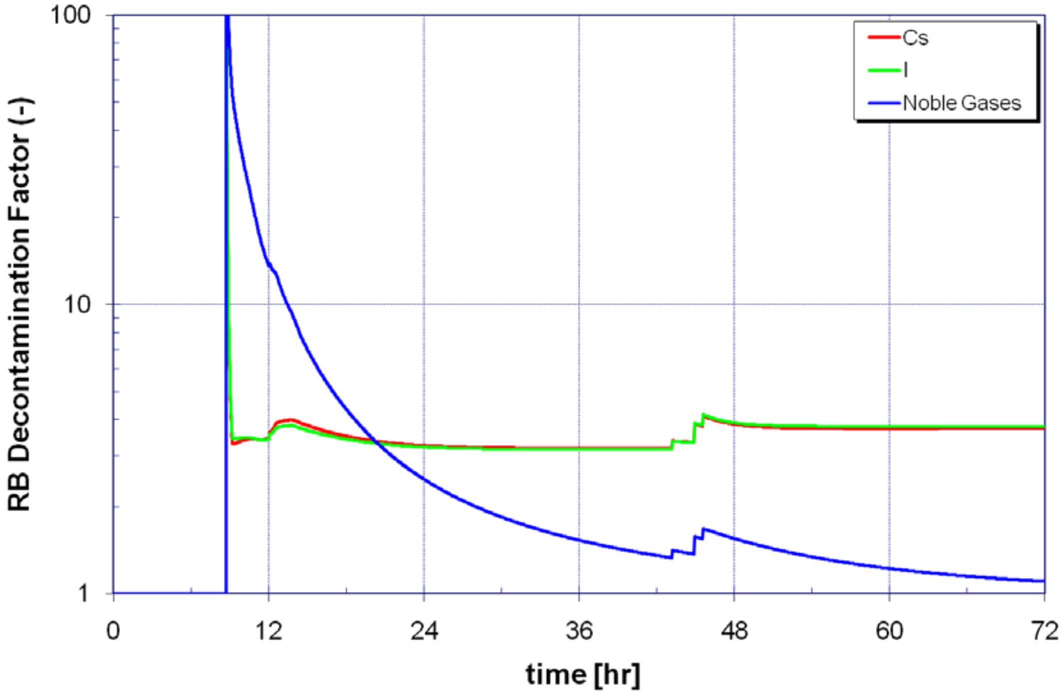

Figure 71 Reactor building DF for unmitigated high-density moderate leak (OCP1)

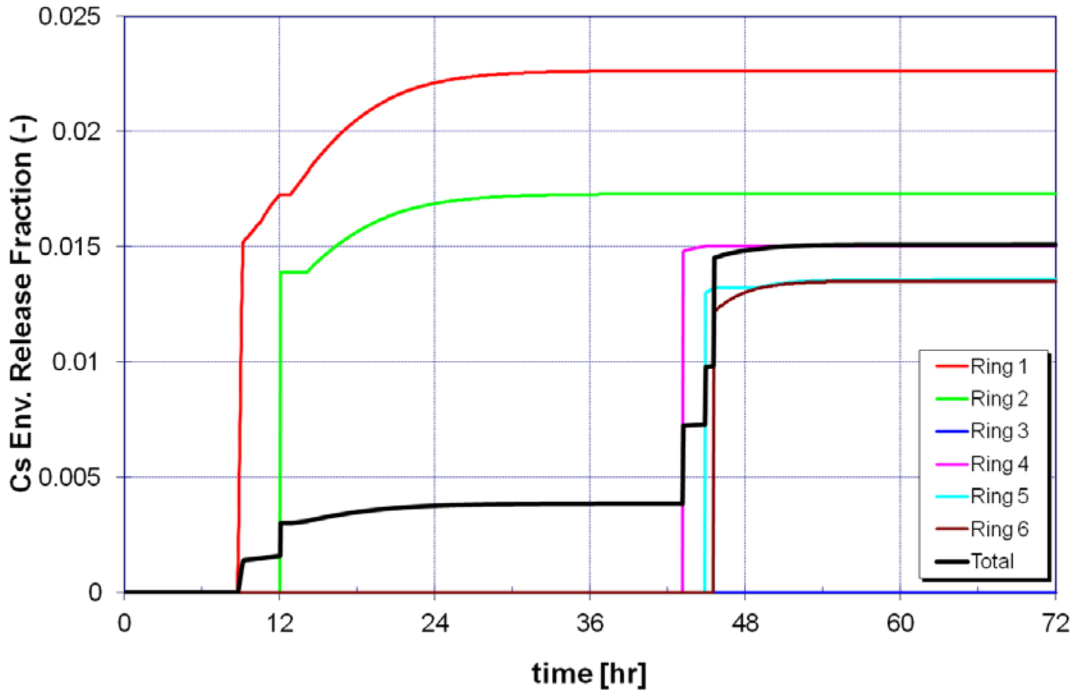

Figure 72 Cesium environmental release fraction for unmitigated high-density moderate leak (OCP1)

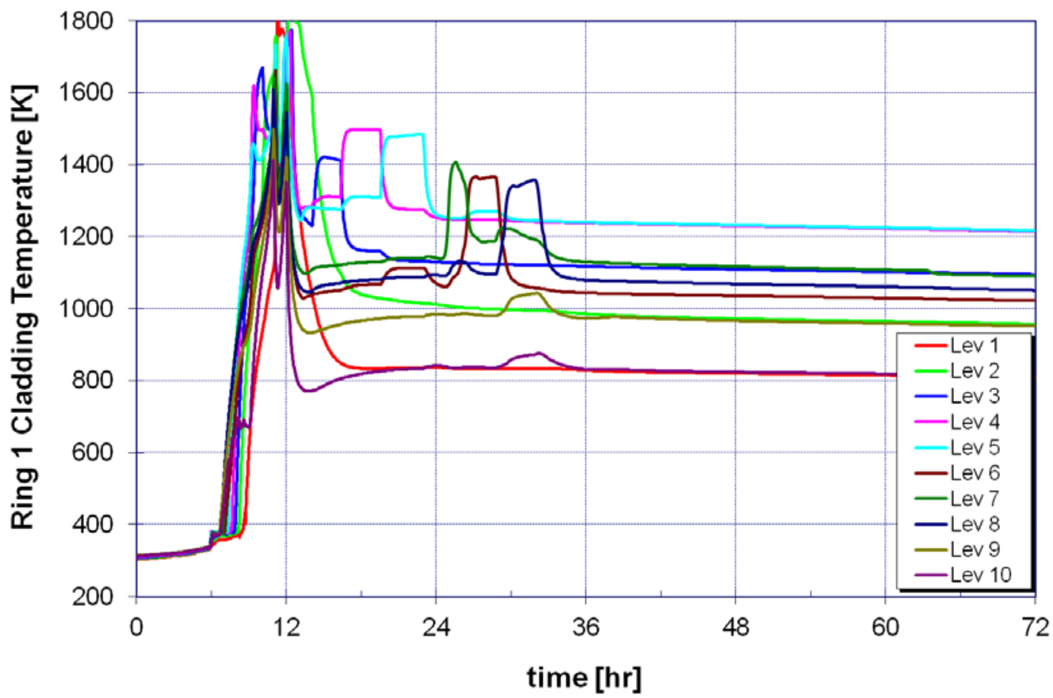

Figure 73 Ring 1 clad temperature for unmitigated low-density moderate leak (OCP1)

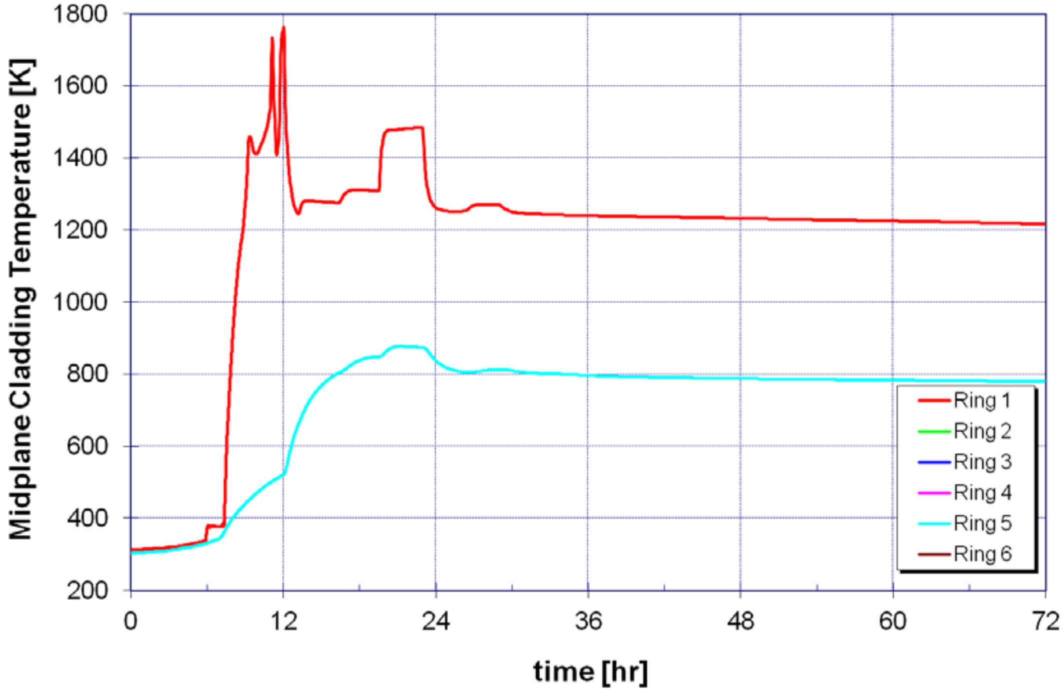

Figure 74 Midplane clad temperature for unmitigated low-density moderate leak (OCP1)

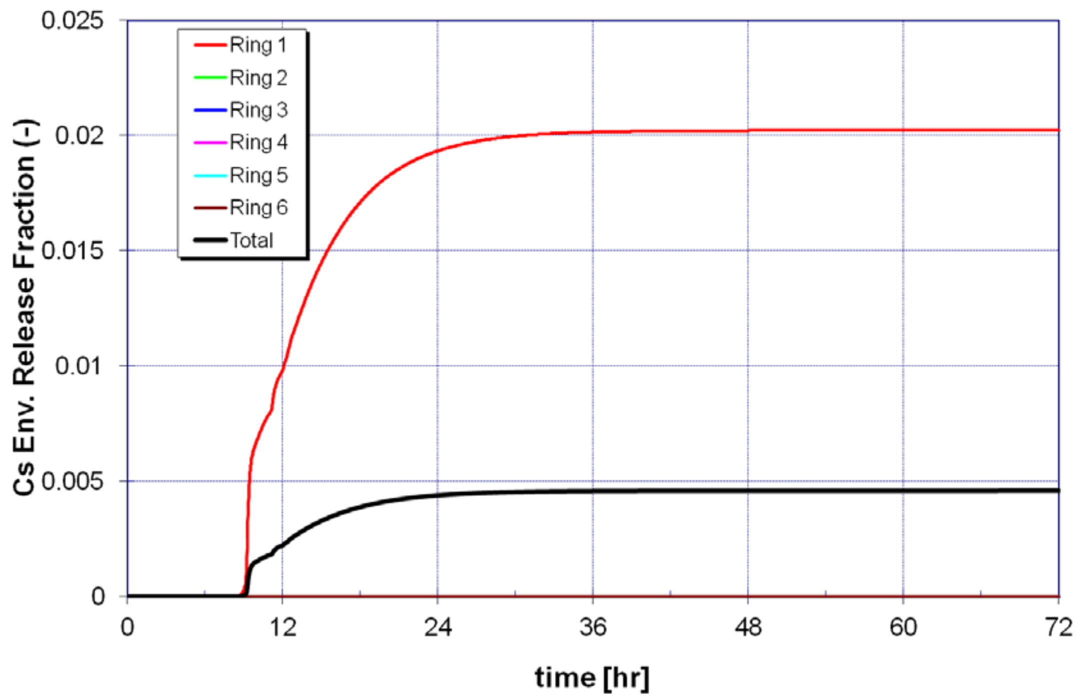

Figure 75 Cesium environmental release fraction for unmitigated low-density moderate leak (OCP1)

Mitigated Moderate Leak (OCP1) Scenario

Figure 76 illustrates the response of the pool to the mitigated scenario. The connectivity between the reactor and the SFP, as well as the additional volume of water, results in a relatively slow draindown. Thus, 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 heatup of the fuel occurs (see Figure 77), there is no indication of a zirconium 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 discharged, high-temperature fuel to adjacent fuel assemblies; and steam cooling from boiling in the bottom of the assemblies between cells keep the fuel temperature near 1200 K. Only Ring 1 had cladding failure and subsequent releases of the gap inventory, as shown in Figure 78. All other fuel was below the threshold for cladding failure and fission product releases.

Figure 79 shows the clad temperature in Ring 1 for the low-density case. The heatup rate for the low-density case is more extreme than the high-density case, as was observed for the 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 heatup. 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. Higher fuel temperatures during the initial oxidation transient led to slightly more release in the low-density case.

[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 the 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.

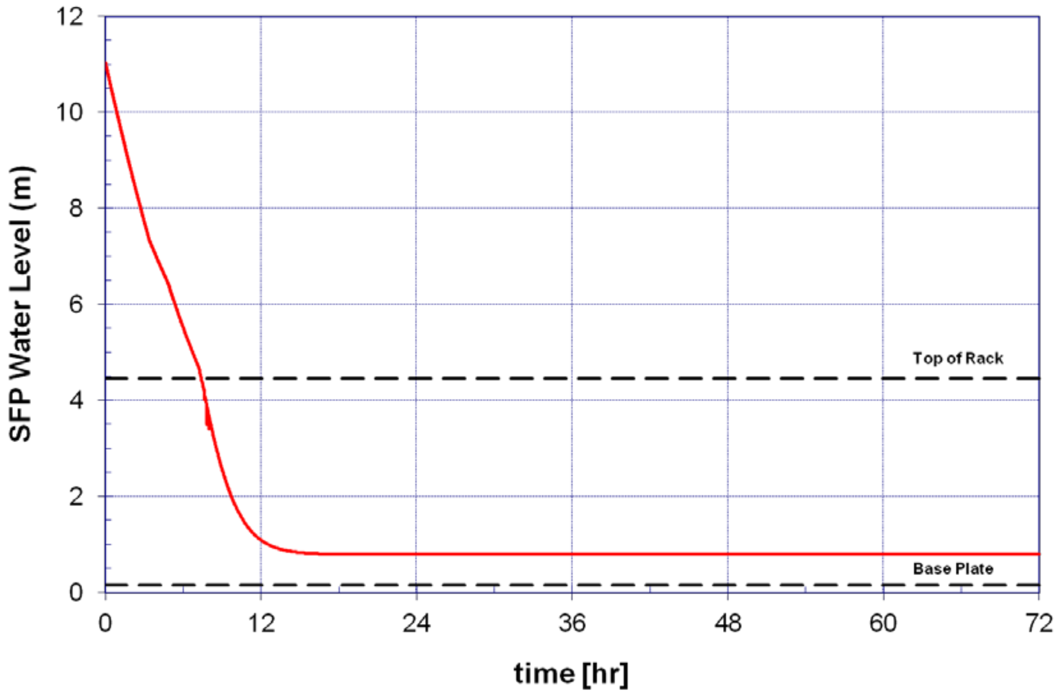

Figure 76 Water level for mitigated high-density moderate leak (OCP1)

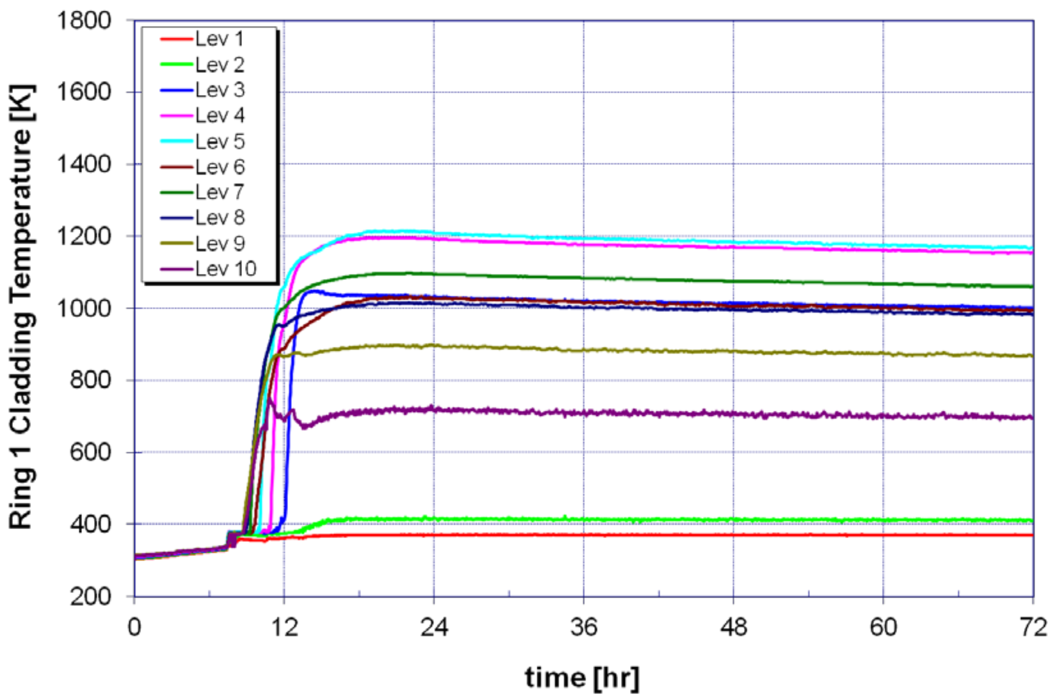

Figure 77 Ring 1 clad temperature for mitigated high density moderate leak (OCP1)

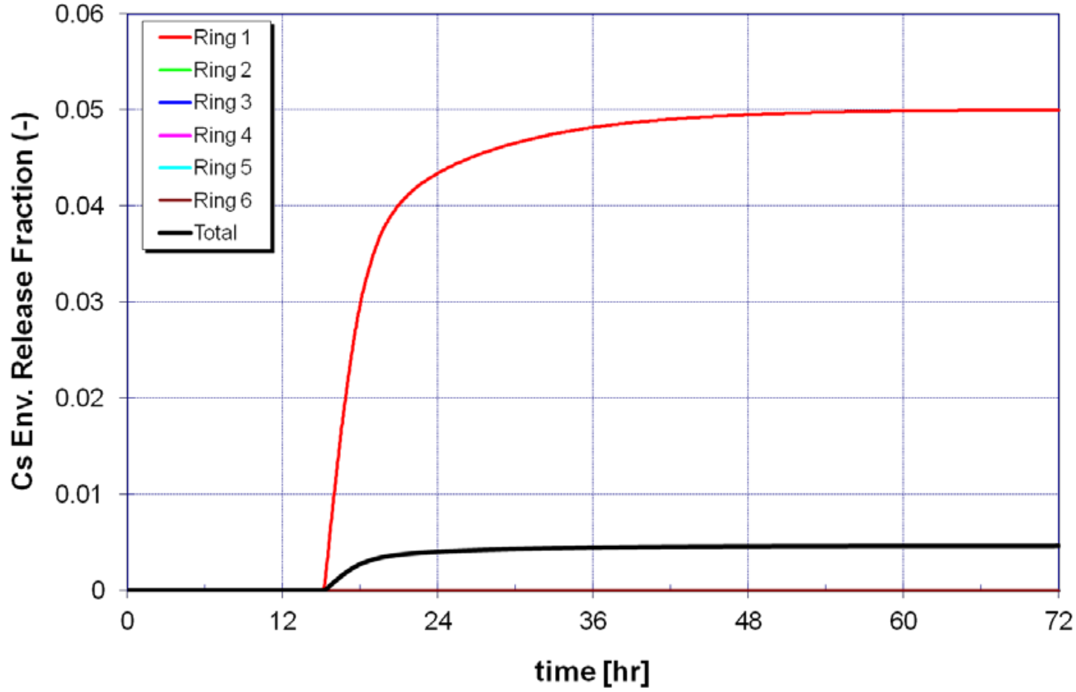

Figure 78 Cesium environmental release fraction for mitigated high density moderate leak (OCP1)

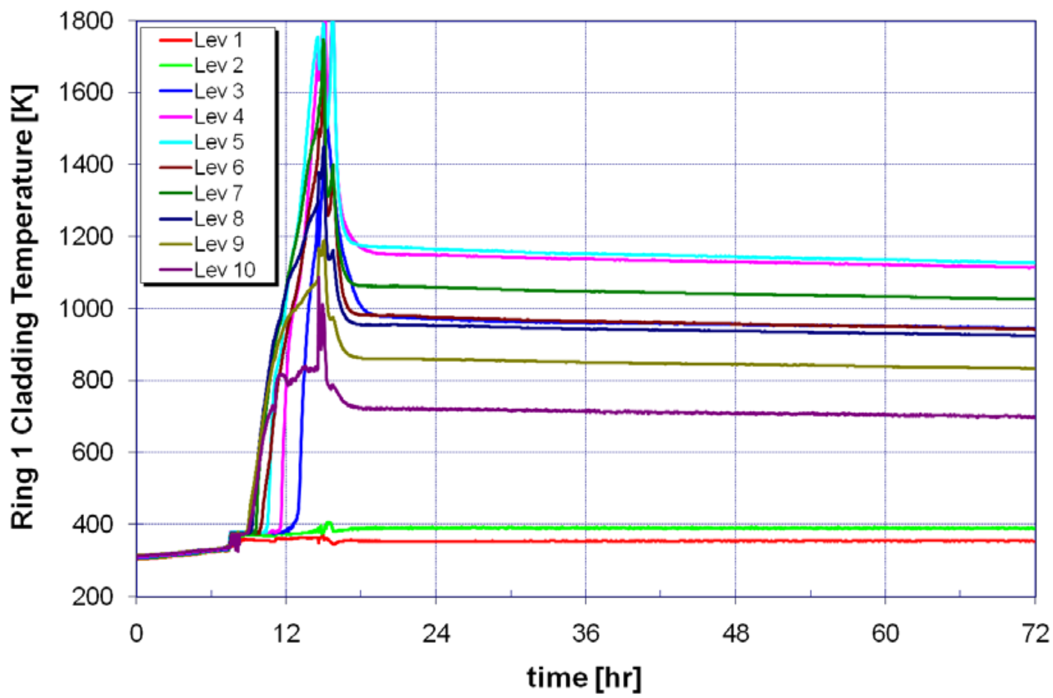

Figure 79 Ring 1 clad temperature for mitigated low density moderate leak (OCP1)

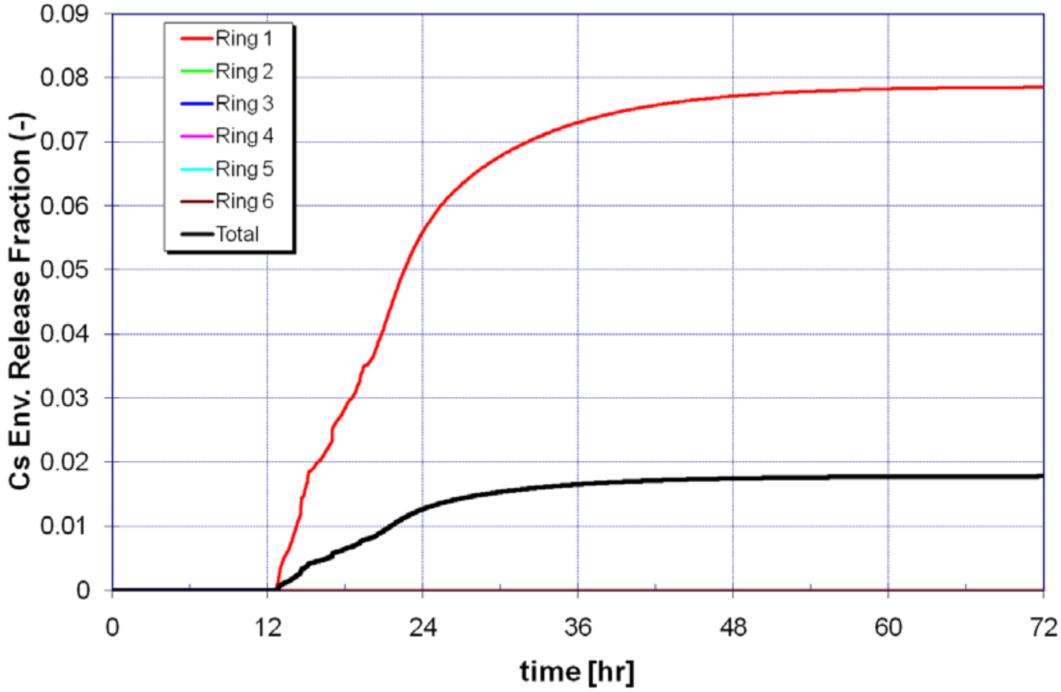

Figure 80 Cesium environmental release fraction for mitigated low density moderate leak (OCP1)

Unsuccessful Deployment of Mitigation for Small Leak (OCP2) Scenario

This scenario involves a hydrogen combustion that caused a late rapid air oxidation. Figure 81 shows the slow draindown of the pool exposing the top of the racks at 42.6 hours compared to 18.7 hours for postoutage scenarios (see Figure 54). Figure 82 illustrates the decay power and the oxidation power. The air oxidation power reaches an order of magnitude higher than the decay heat during the oxidation transient after 60 hours. The fuel heatup begins after the water level reaches about the fuel midplane (see Ring 1 response in Figure 83). The high-temperature fuel in Ring 1 heats the surrounding low decay heat fuel in Ring 2, as shown in Figure 84.

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 structures. The hydrogen concentration builds up until it reaches 10 percent at 65 hours and combusts. At the time of combustion, all the necessary conditions are satisfied; the hydrogen concentration is 10 percent, the oxygen is 10 percent, and the steam is less than the 55-percent 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 Ring 1 and then in Ring 2. The reactor building decontamination factor approaches unity (Figure 86) resulting in about a 17-percent cesium release to the environment (Figure 87).

The response for the low-density case was similar to, 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. Figure 88 shows the response of the highest decay heat assemblies in Ring 1. The peak fuel temperatures were less than 1,400 K. As shown in Figure 89, the fuel in Ring 3 had a similar response, but the fuel in Ring 5 was substantially lower. Fewer fuel assemblies and lower peak temperatures resulted in 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 percent) than in the high-density case.

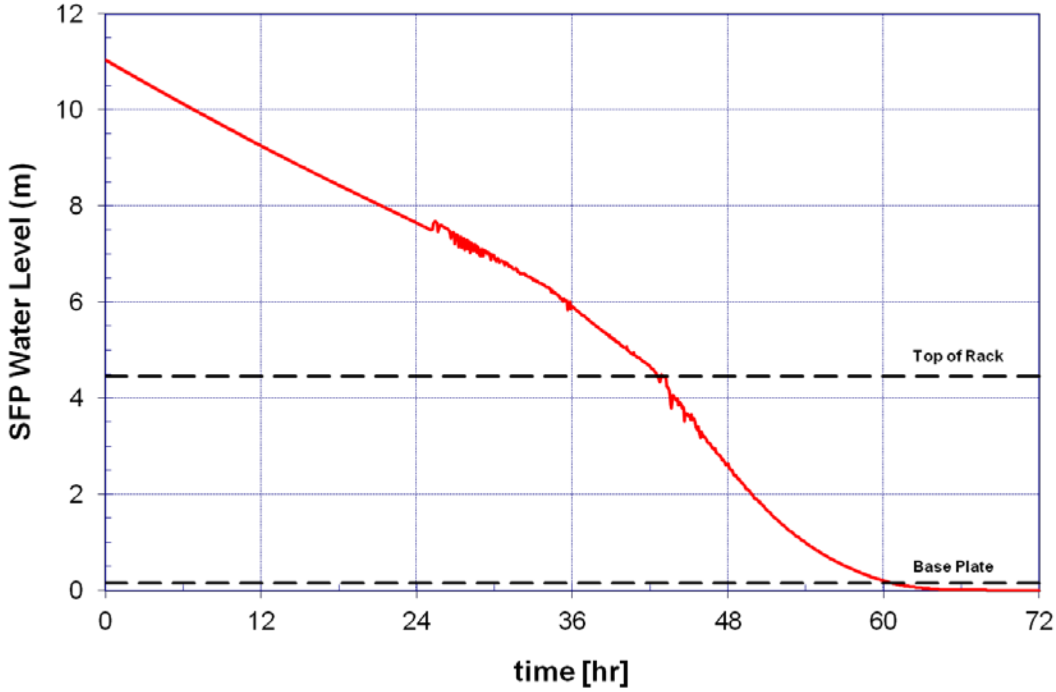

Figure 81 Water level for unmitigated high-density small leak (OCP2)

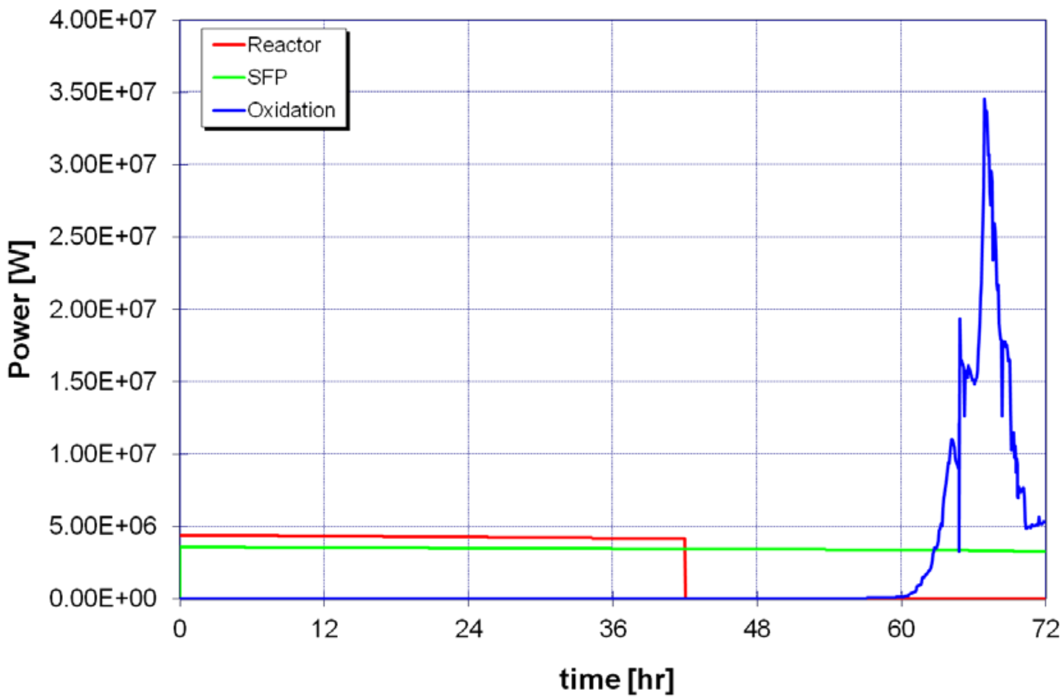

Figure 82 SFP power for unmitigated high-density small leak (OCP2)

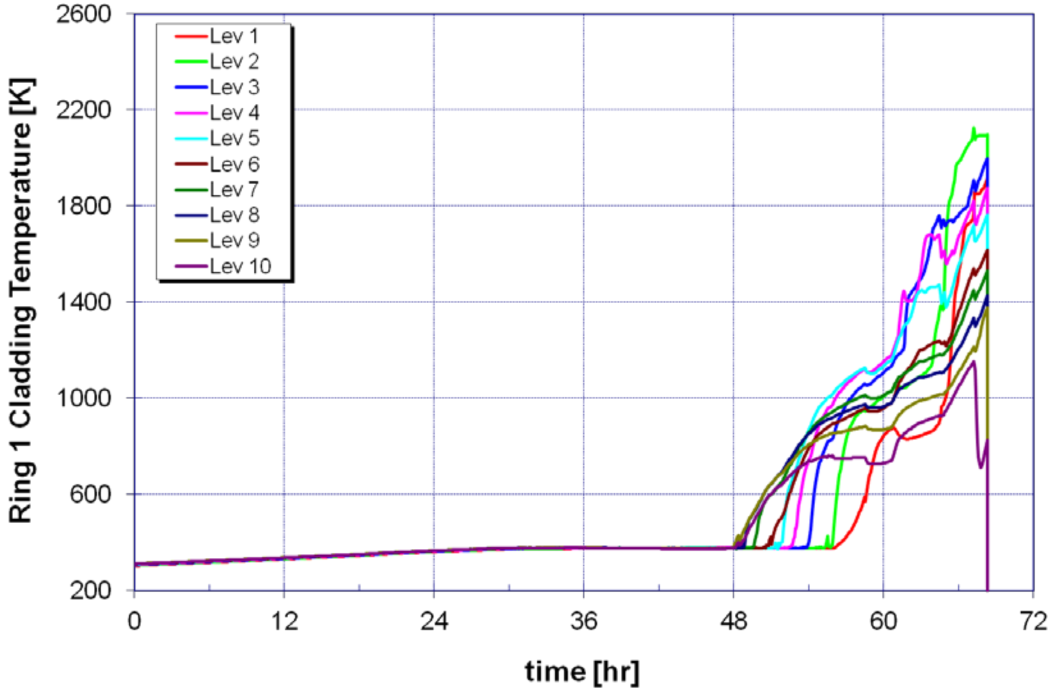

Figure 83 Ring 1 clad temperature for unmitigated high-density small leak (OCP2)

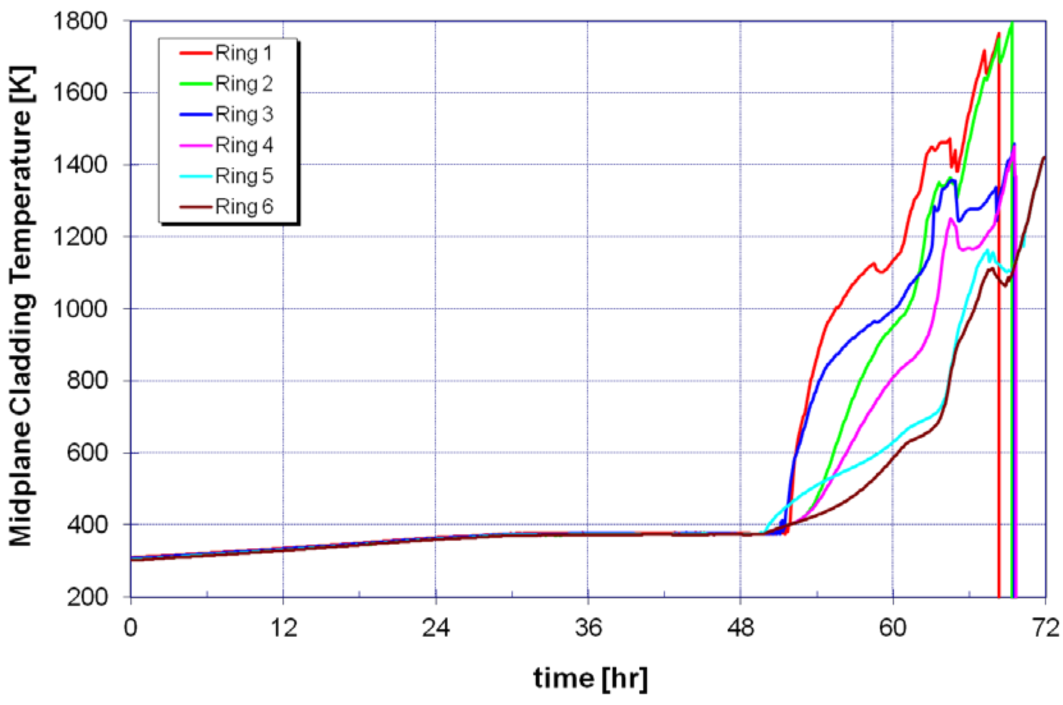

Figure 84 Midplane clad temperature for unmitigated high-density small leak (OCP2)

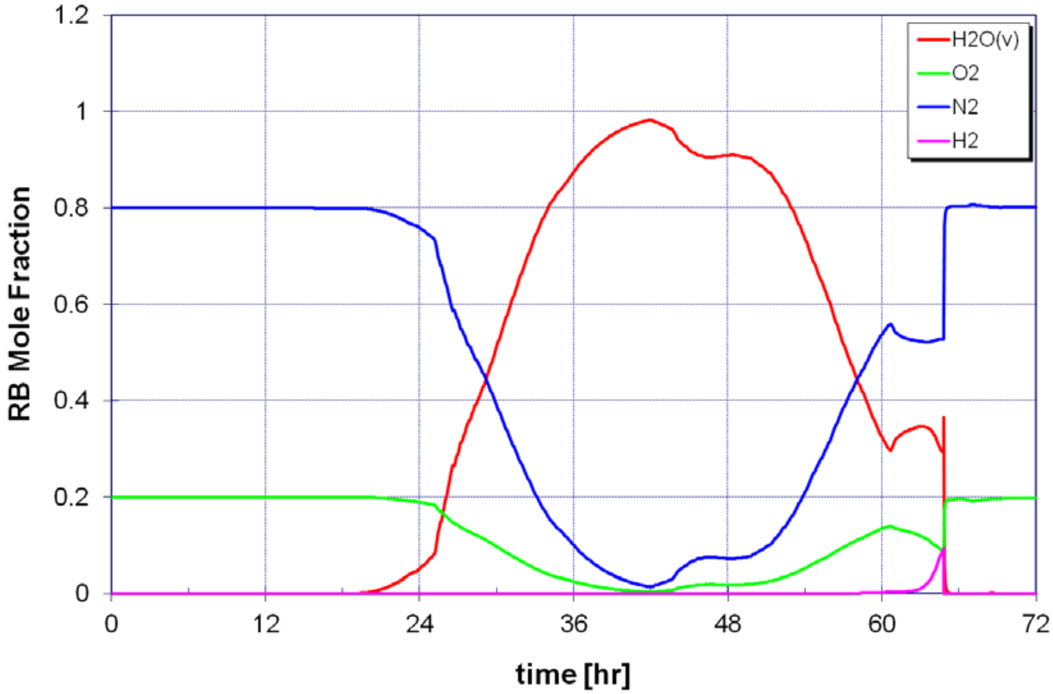

Figure 85 Reactor building mole fraction for unmitigated high-density small leak (OCP2)

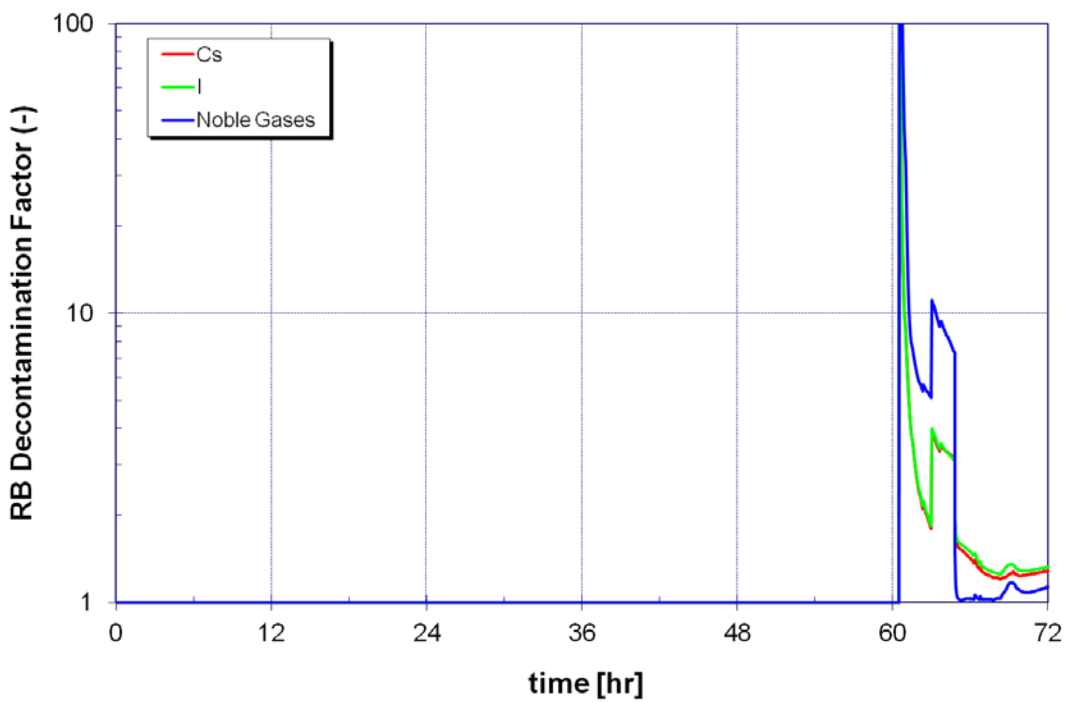

Figure 86 Reactor building DF for unmitigated high-density small leak (OCP2)

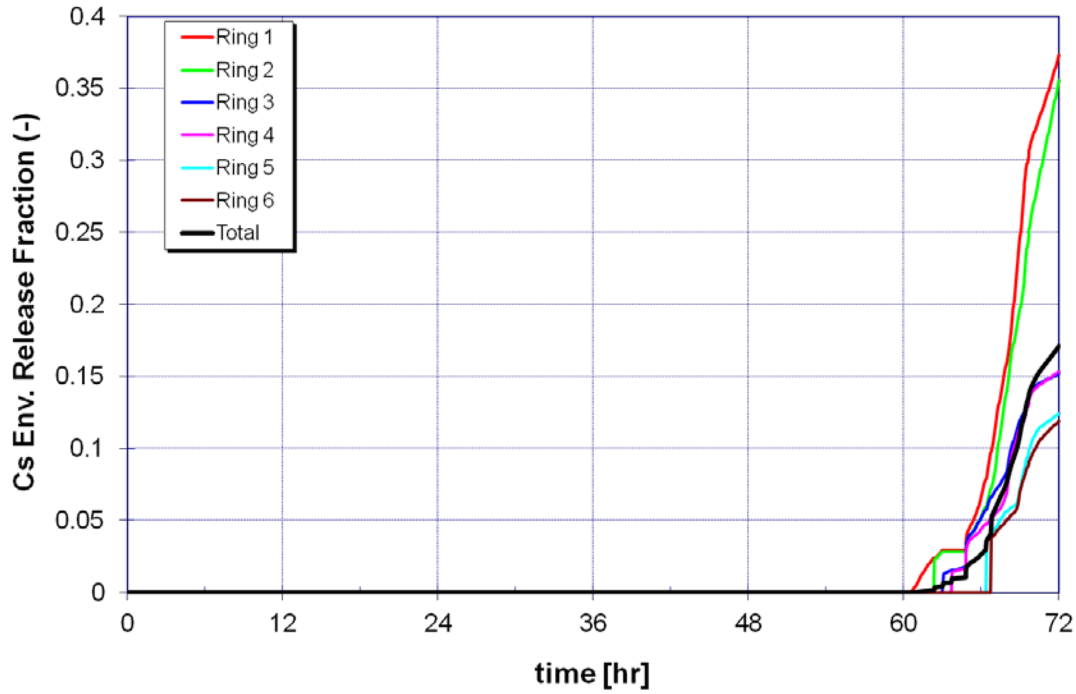

Figure 87 Cesium environmental release fraction for unmitigated high-density small leak (OCP2)

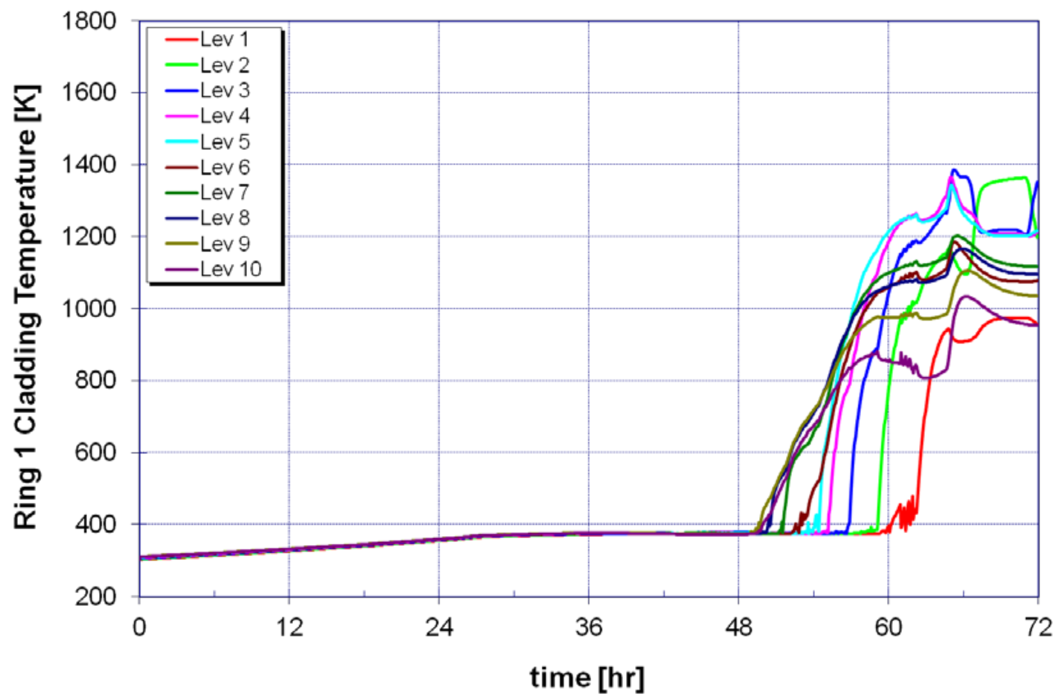

Figure 88 Ring 1 clad temperature for unmitigated low-density small leak (OCP2)

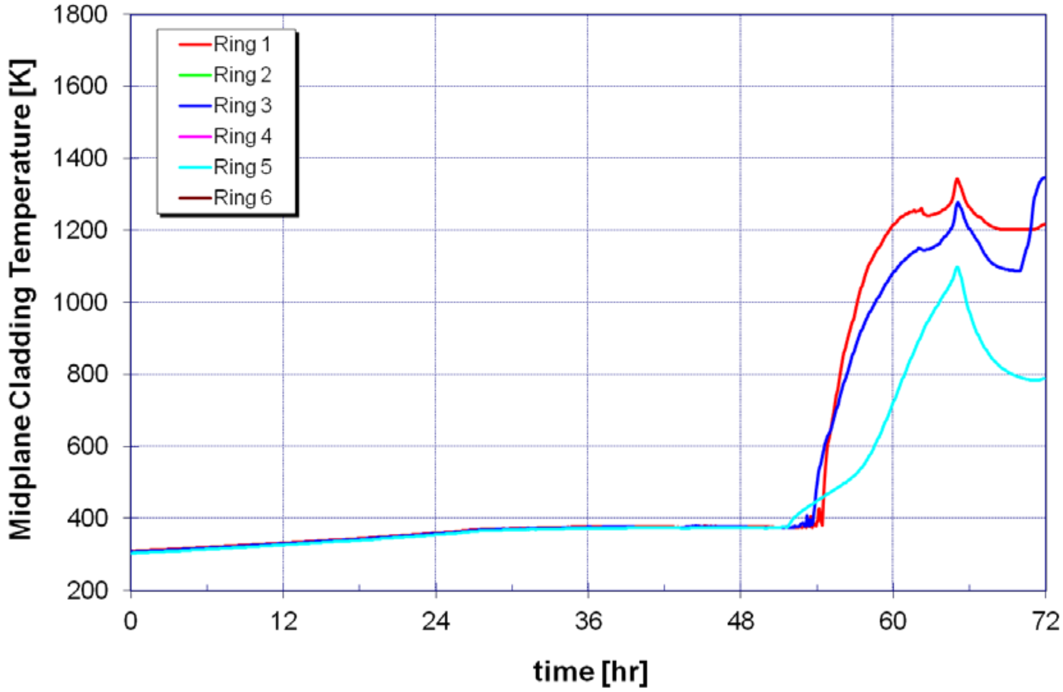

Figure 89 Midplane clad temperature for unmitigated low-density small leak (OCP2)

Unsuccessful Deployment of Mitigation for Moderate Leak (OCP3) Scenario

Figure 90 (compare to Figure 67 for OCP1) depicts the response of the fuel temperature in Ring 1. The heatup starts sooner because there is less water to drain and the approach to a zirconium fire is more gradual because of lower decay heat (i.e., by a factor of 2.5—see Table 25) and the natural circulation of air through the assemblies. However, once the zirconium fire is started, the maximum temperatures are comparable in both cases. As shown in Figure 90, the zirconium fire starts at Level 5 but then moves slowly to Level 4, Level 3, and Level 2. After the peak temperature at Level 4, the peak temperature in the zirconium fire front decreases with each successive level. Radial heat transfer from the fuel racks to the SFP wall (Figure 91), the buildup of the oxide layer on the fuel, and the depletion of the oxygen in the reactor building (Figure 92) cause the clad temperature to decrease. After 24 hours, the fuel temperatures in Ring 1 are relatively stable. There was no hydrogen combustion in this calculation. When the hydrogen concentration peaks at 8 percent, the oxygen concentration is only 3 percent (well below an amount sufficient for combustion as shown in Figure 92).

Figure 93 shows the temperature profiles for the low-density case.. The low-density temperatures are about 400 K lower than the high-density case, with the total cesium release being about 0.1 percent compared to 0.7 percent 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 analysis was performed to examine the effect of higher vapor pressure for the air-oxidizing ruthenium releases. Figure 94 (the default ruthenium release model) and Figure 95 (the enhanced ruthenium release model used in the present study) show that the ruthenium release differs by an order of magnitude.[37] All the calculations with moderate leaks were based on the enhanced ruthenium release model.

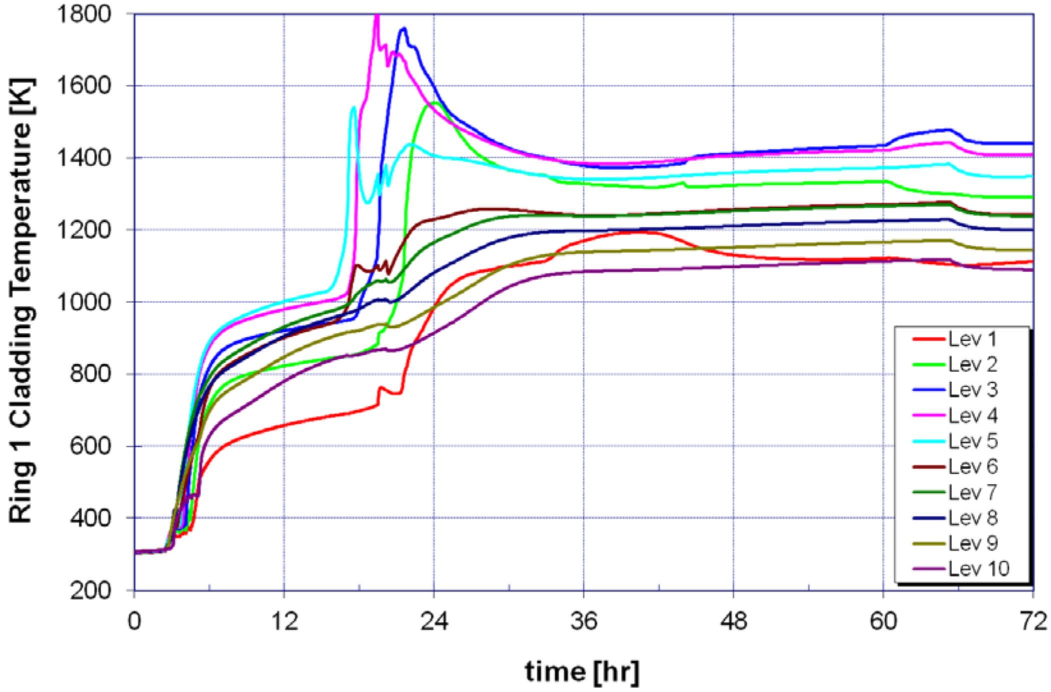

Figure 90 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP3)

[37] However, ruthenium release differences could be higher for scenarios in OCP1 and OCP2.

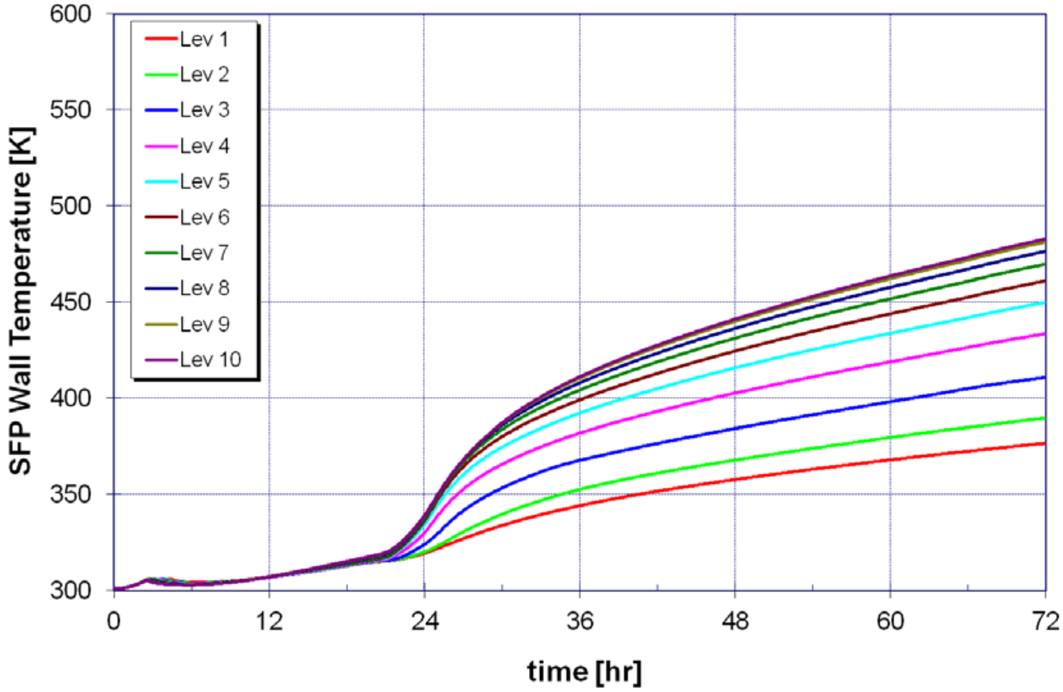

Figure 91 SFP wall liner temperature for unmitigated high-density moderate leak (OCP3)

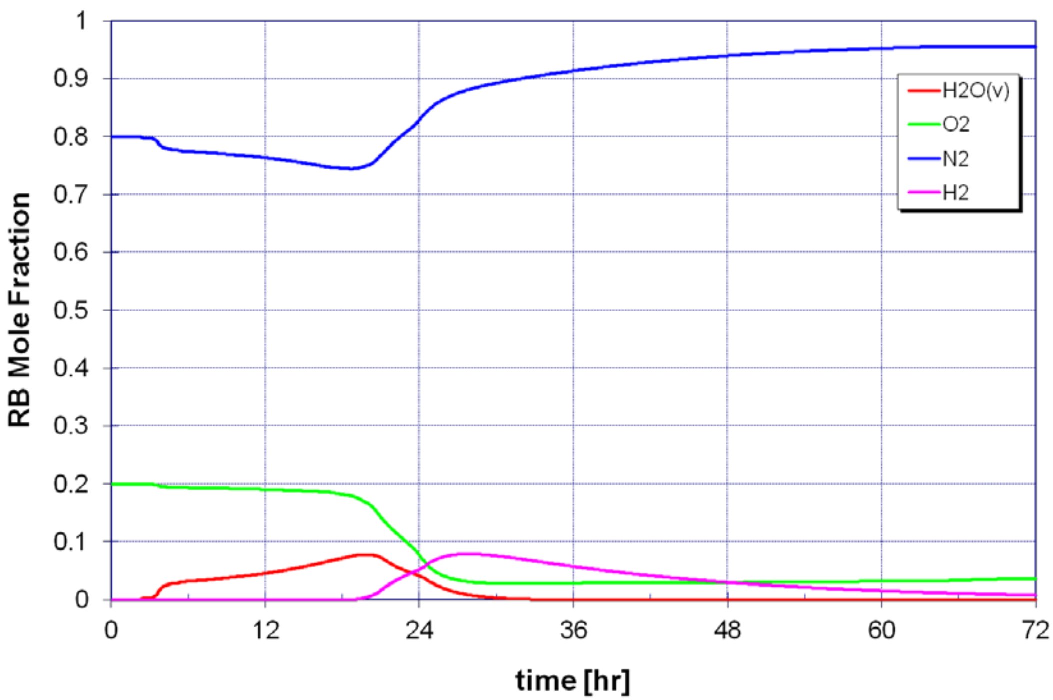

Figure 92 Reactor building mole fractions for unmitigated high-density moderate leak (OCP3)

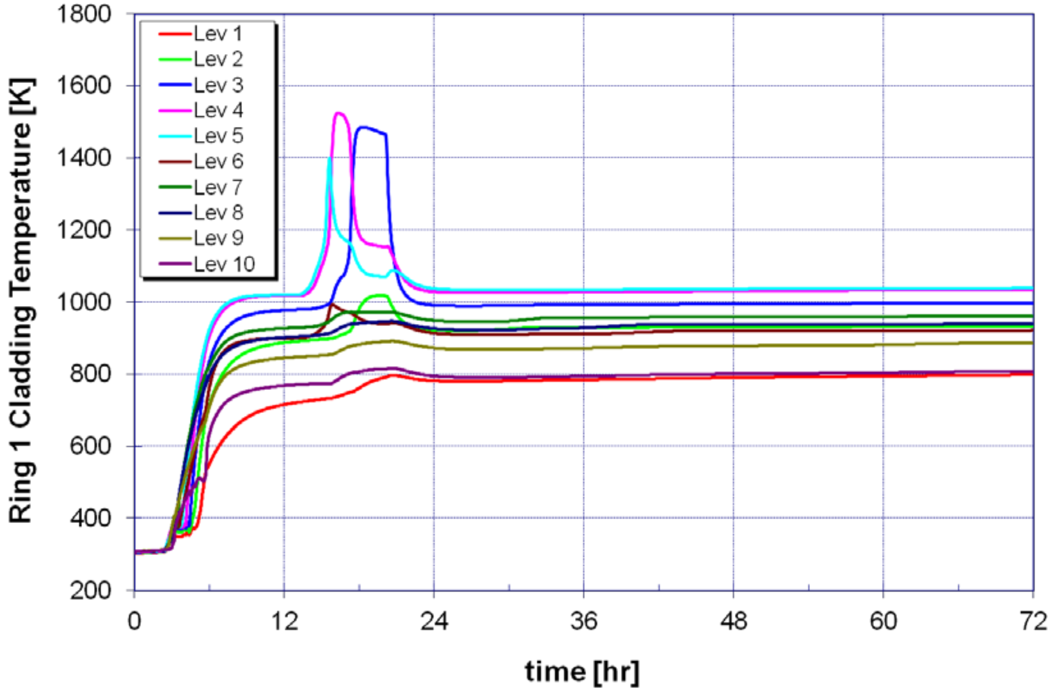

Figure 93 Ring 1 clad temperature for unmitigated low-density moderate leak (OCP3)

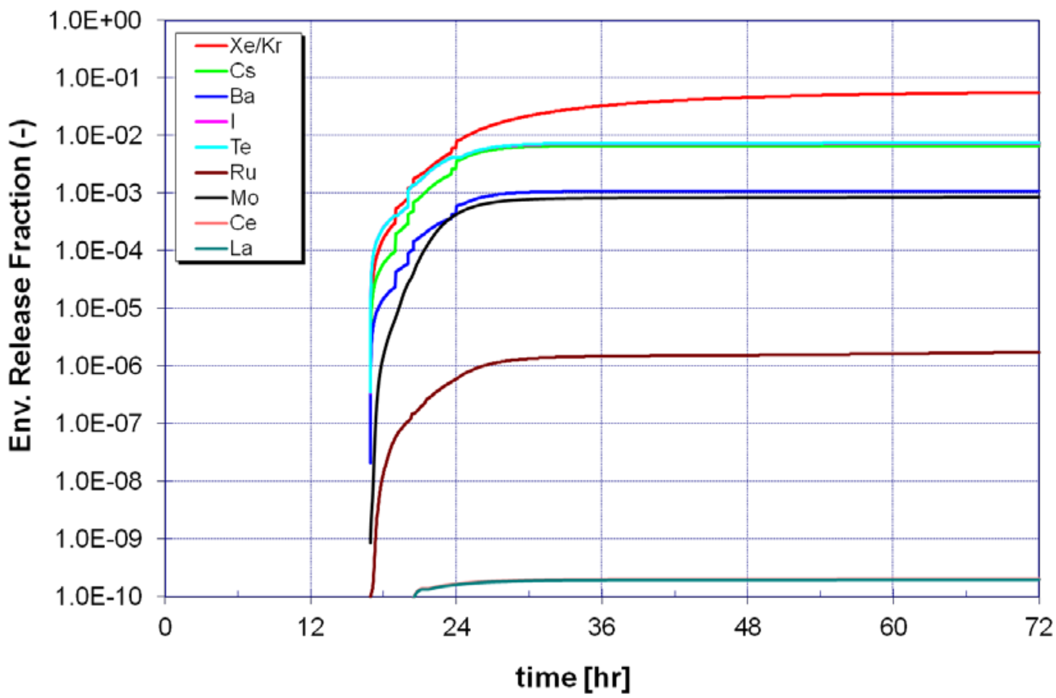

Figure 94 MELCOR default ruthenium release for unmitigated high-density moderate leak (OCP3)

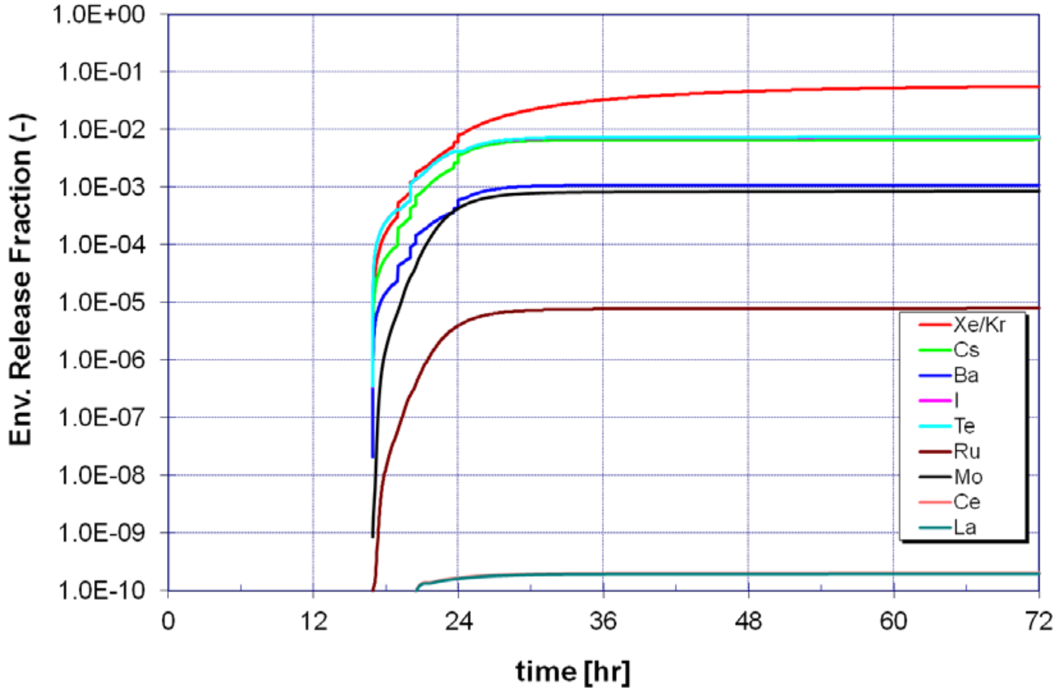

Figure 95 MELCOR enhanced ruthenium release under air oxidizing conditions for unmitigated high density moderate leak (OCP3)

6.3.3 Source Terms for Offsite Consequence Analysis

Table 27 summarizes the release characteristics and key events for the high-density scenarios, and Table 28 summarizes these factors for the low-density scenarios. Previous sections of this report provided a more detailed discussion of key phenomena for selected sequences. The releases are binned for offsite consequence analysis, which Section 7 describes.

For the high-density loading, all of the mitigated scenarios (except OCP1) have no release, either because the makeup exceeds the leak rate, as in the small leak cases, or the mitigation is successful in limiting the fuel heatup and avoiding 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 OCP1 cases which had higher release fractions than the high-density cases in some instances. This difference resulted from more rapid heatup of the fuel in Ring 1 because of less efficient heat transfer to the outer assemblies. As shown above, 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 because of a larger inventory of water (assemblies removed) resulting in longer times for clearing the baseplate. The gap release first occurs in Ring 1 (hot assemblies), which has the same decay heat in both high-density and low-density configurations.

6.3.4 Accumulation of Water Elsewhere in the Reactor Building

There are approximately 50 floor drains on the refueling floor, both at floor level and in the lower, recessed areas of the floor. The two stair towers are fully enclosed and will not be subjected to condensation. The doors to the stair towers are secondary containment doors, and so they have air seals (weather stripping, but not watertight seals). The open crane hatch in the refueling floor has a surrounding 6-in. (0.15-m) curb, so condensation on the floor will run to the floor drains and not the hatch. However, condensation forming directly over the hatch, which is 17 ft 0 in. (5.2 m) by 21 ft 9 in. (6.6 m), will fall to Elevation 135 ft (41.2 m). There is a 4-in. (0.1-m) floor drain directly under the hatchway at Elevation 135 ft (41.2 m), with no equipment in the footprint of the hatch.

If the stainless steel liner plate and 6-ft- (1.83-m-) thick reinforced concrete slab of the SFP leak through, the water will fall onto Elevation 165 ft (50.3 m) of the reactor building. Directly beneath the SFP on Elevation 165 ft (50.3 m) are the three fuel pool cooling pumps, the three fuel pool cooling heat exchangers, and the three fuel pool service water booster pumps. There are several floor drains in this area. Equipment adjacent to this area that could be affected by a large volume of water includes the fuel pool equipment panel and the reactor level and pressure instrument racks. If the floor drains on Elevation 165 ft (50.3 m) cannot keep up with the flow, then the alternate flow paths would be the crane hatch or the door to each of the two stair towers, having the same configuration as described for the refueling floor above. A significant flow rate could also affect the emergency auxiliary load centers on Elevation 165 ft (50.3 m). Water flowing over the curb of the crane hatch would reach Elevation 135 ft (41.2 m), where it would either enter the floor drains, flow through the grating to the torus room floor, or exit the building under the doors of the equipment access lock. Water reaching the stair towers would travel to the bottom of the stair tower. Water in the W stair tower would reach the residual heat removal pump room which has its own floor drains (procedurally controlled, normally closed) and sump pump. Water in the east stair tower would reach the core spray pump room or elevator shaft bottom which have floor drains (procedurally controlled, normally closed) that run to the reactor building main sump.

The reactor building MELCOR model is simplified (see Figure 42). Therefore, all water leakages corresponding to the SFP damage and draindown and overflow from water accumulation from condensation are directed to the environment. The model does not track the flow of the water and accumulation in other parts of the reactor building.

Table 27 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 release at 72 hours	Cs-137 (MCi) Released	I release at 72 hours	I-131 (MCi) Released
OCP1	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
OCP2	None	Yes							
	None	No							
	Small	Yes							
	Small	No	42.6	60.5	64.8	17.1%	7.90	17.1%	1.91
	Moderate	Yes							
	Moderate	No	5.9	11.6	No	1.6%	0.73	2.0%	0.22
OCP3	None	Yes							
	None	No							
	Small	Yes							
	Small	No	18.7	40.6	47.3	42.0%	24.20	51.2%	0.73
	Moderate	Yes							
	Moderate	No	2.5	16.9	No	0.7%	0.39	0.7%	0.01

Table 28 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
OCP1	None	Yes							
	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	0.5%	0.05	1.7%	0.13
OCP2	None	Yes							
	None	No							
	Small	Yes							
	Small	No	43.1	59.2	No	1.7%	0.28	3.3%	0.37
	Moderate	Yes							
	Moderate	No	5.9	10.5	No	0.4%	0.07	0.7%	0.08
OCP3	None	Yes							
	None	No							
	Small	Yes							
	Small	No	18.8	41.6	No	0.6%	0.10	1.2%	0.02
	Moderate	Yes							
	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 SFP (as detailed in the previous sections), a release of radioactive material from the nuclear power plant site into the atmosphere could occur. Such a release of radioactive material is expected to 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 MELCOR Accident Consequence Code System, version 2 (MACCS2) code is used to model offsite release and consequences of radioactive material. MACCS2 (SNL, 1997) has been developed by SNL for the NRC 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 can 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. Along with MACCS2, SNL has also developed WinMACCS for the NRC. WinMACCS is a user friendly graphical interface to MACCS2 that facilitates selection of input parameters and sampling of uncertain input parameters and performs post processing of results.

MACCS2 rev. 3.7.0 was used for the offsite consequence analysis in this study. In addition, many of the input values for offsite release and consequence modeling are based on approaches developed in the "State-of-the-Art Reactor Consequence Analyses" research project (NUREG-1935). These approaches are documented in greater detail in NUREG/CR-7009, "MACCS2 Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses Project", (expected to be published in 2013), The modeling for NUREG-1935 was, in turn, based on previous studies such as NUREG-1150, an expert elicitation of the NRC/Commission of the European Communities (CEC) to update certain transport and dose parameters in NUREG/CR-7161 (NRC, 2013), and an update of the dose coefficients and dose-response modeling to be consistent with the latest Federal guidance report (FGR) at the time (FGR-13, "Cancer Risk Coefficients for Environmental Exposures to Radionuclides," issued in 2002 (EPA, 2002)). Differences between the approaches used in NUREG-1935 and the approaches used in this study are documented below.

7.1 Offsite Consequence Modeling

7.1.1 Radiological Source Term

A source term definition for MACCS2 was created for each accident consequence calculation. The activity levels of different radionuclides from the fuel in the pool were supplied by ORIGEN calculations. The physical state of the plume, including information on the chemical group release rates, aerosol size distributions, density, and mass flow rate was supplied by MELCOR. The MELCOR analyses provided a release rate for each chemical group. Because the amount of release may differ for different sections of the pool, a new methodology was developed for this study to account for the distribution of radionuclides in the pool as well as radionuclide-specific release magnitudes. For instance, recently discharged fuel, which has more short-lived radionuclides, is more likely to release before and to greater magnitudes than older fuel. This process is described in more detail in Section 6.1.5.

Because explicit modeling with MACCS2 of all release sequences generated by MELCOR analyses is computationally expensive, the MELCOR sequences were binned by their Cs-137 and I-131 release activities (see Table 29). The first criterion used to bin the sequences was

Cs-137 release, because Cs-137 is the most significant contributor to long-term consequences. I-131 was also chosen as a criterion 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. The tally into each of these bins can be seen in Table 30.

Table 29 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 30 Release Category Tally

Release Category	RC11	RC12	RC13	RC21	RC22	RC23	RC31	RC32	RC33	Total
Sequence Tally	5	5	0	2	0	0	0	0	2	14

One sequence was chosen from each bin (not including bins with no contributing accident sequences) to represent the entire release category, and the offsite consequences of these sequences were analyzed. The study considered a number of different factors to determine which sequence should represent each bin, including the release frequency, the relative Cs-137 and I-131 release for the bin, the start time of release, the SFP loading configuration, and the availability of the source term data (some accident progression calculations were still ongoing at the time the selection was made). In addition, because of the significant differences in release category 33 relative to the other bins, both of these sequences were analyzed, as identified in Table 31. Then, based on their conditional probabilities, all the main MELCOR sequences and their associated consequences were applied to the scenarios reported in the results, which are high-density and low-density loading both with and without successfully deployed 10 CFR 50.54(hh)(2) equipment. Sequences with no release were not included, as they do not have offsite consequences. Section 6.3 contains more information regarding which sequences do and do not have a release. For all sequences, successful deployment of 10 CFR 50.54(hh)(2) equipment prevents release of radioactive material, except for a moderate size leak during OCP1 (as defined in Section 5.2), which is when newly discharged fuel is first loaded from the reactor. Without successful deployment of 10 CFR 50.54(hh)(2) equipment, the predicted scenario-specific release frequency is 10^{-7} per year.

Table 31 Listing of Scenario-specific Release Sequences

High Density (1x4) Fuel Loading							
Unsuccessful mitigation				Deployed 50.54(hh)(2)			
Sequence		Release Frequency (/yr)[1]	Release Category	Sequence		Release Frequency (/yr)	Release Category
OCP1	small leak	6E-09[2]	RC12*	OCP1	mod leak	6E-09	RC12
	mod leak	6E-09	RC21				
OCP2	small leak	2E-08	RC33*			No Release	
	mod leak	2E-08	RC21*				
OCP3	small leak	4E-08	RC33*				
	mod leak	4E-08	RC11				
	Total	1E-07			Total	6E-09	
Low Density Fuel Loading							
Unsuccessful mitigation				Deployed 50.54(hh)(2)			
Sequence		Release Frequency (/yr)	Release Category	Sequence		Release Frequency (/yr)	Release Category
OCP1	small leak	6E-09	RC12	OCP1	mod leak	6E-09	RC12
	mod leak	6E-09	RC11				
OCP2	small leak	2E-08	RC12			No Release	
	mod leak	2E-08	RC11				
OCP3	small leak	4E-08	RC11*				
	mod leak	4E-08	RC11				
	Total	1E-07			Total	6E-09	

[1] Release frequency = initiating event frequency * ac power fragility * OCP probability * liner fragility for the specified leak size (see Section 5.6.3 for conditional probabilities)
[2] Example calculation: 1.7×10^{-5}/yr · 0.84 · 0.0086 · 0.05 = 6×10^{-9}/yr
* Sequences marked with an (*) were used in MACCS2 analysis

7.1.2 Atmospheric Modeling and Meteorology

The atmospheric transport and dispersion model in MACCS2 is a straight-line Gaussian plume segment dispersion model. For this study, the atmospheric release of radionuclides is discretized into (at longest) 1-hour plume segments. This accounts for variations in the release rate, as well as for changes in wind direction. More plume segments increase the resolution of the dispersion modeling to the point the resolution corresponds to the time resolution of the weather data, because each segment can travel in a compass direction representative of the actual weather data at the time the plume segment is released.

A set of aerosol deposition velocities, combined with the aerosol size distribution from MELCOR, determines the rates aerosols are deposited from the plume to the ground. 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 microns in diameter, which corresponds to a deposition velocity of about 4 or 5 millimeters per second. Dry deposition velocities have been updated to account for a more typical surface roughness of 60 cm for the reference plant site. (A surface roughness of 10 cm was used for NUREG-1935 and a 60-cm surface roughness was considered in a sensitivity calculation.) The relative aerosol deposition velocities, as well as much of the non-site-specific data for acute health effects, are developed from NUREG/CR-7161 (NRC, 2013).

Because the exact weather conditions that would apply in the case of a potential accident in the future cannot be known in advance, MACCS2 accounts for weather variability by analyzing a statistically significant set of weather trials. Thus, the modeled results are ensemble averages of weather that represent of the full spectrum of meteorological conditions. The weather-sampling strategy for this study uses a nonuniform weather-binning approach. Weather binning is an approach used in MACCS2 to categorize similar sets of weather data based on windspeed, stability class, and the intensity and timing of precipitation. This sampling strategy was chosen to represent the statistical variations of the weather. Further discussion on this approach can be found in NUREG/CR-7009.

Meteorological data used for this project consisted of one year of hourly meteorological data (8,760 data points for each meteorological parameter). The data are from onsite meteorological tower observations are the same as those used in NUREG-1935. The site selected for the reference plant provided two 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 meteorological data chosen for the reference plant was 2006, which was based on data recovery (greater than 99 percent being desirable) as documented in NUREG/CR-7009. Different trends (e.g., wind rose pattern and hours of precipitation) between the years were estimated to have a relatively minor (less than 25 percent) effect on the final NUREG-1935 results. More specific details of the weather data can be found in NUREG/CR-7009.

7.1.3 Exposure, Dosimetry, and Health Effects Modeling

MACCS2 considers groundshine, cloudshine, inhalation, and ingestion exposure pathways. The principal exposure pathway to members of the public occupying land contaminated by atmospheric deposition of radioactive materials is expected to be exposure of the whole body to external gamma radiation. Although it is normally expected to be of lesser importance, the inhalation pathway contributes additional doses to internal organs (EPA, 1992), especially during the emergency phase of the accident. The MACCS2 outputs for health effects and population dose include doses from exposure via the ingestion pathway. However, the MACCS2 code does not represent these consequences in the individual LCF risk results[38]. Food ingestion parameters were chosen to be consistent with Sample Problem A, as documented in NUREG/CR-6613, Vol. 1 (Chanin and Young, 1998). Sample Problem A is based on NUREG-1150, with the exception that newer options not included in the older MACCS model were used to demonstrate new capabilities in MACCS2 (e.g., that the food ingestion model is updated to use the newer COMIDA2 rather than the original MACCS food-chain model). NUREG-1935 did not include exposure to contaminated food because staff judged it not to be a significant contributor to individual risk.

NUREG/CR-7009 reviews the shielding factors applied to evacuation, normal activity, and sheltering for each dose pathway (e.g. groundshine) used in NUREG-1150 (NRC, 1990) and NUREG/CR 6953, Volume 1, “Review of NUREG-0654, Supplement 3, ‘Criteria for Protective Action Recommendations for Severe Accidents’—Focus Group and Telephone Survey,” issued

[38] Including the ingestion pathway is predicted to increase health effect risks in this study by about 5% with an LNT dose response model, depending on the scenario.

October 2008 (NRC, 2008c). This study uses the same shielding factors as updated in NUREG/CR-7009.

The site file containing population and economic data was created for 16 compass sectors and then interpolated onto a 64 compass-sector grid for better spatial resolution for the consequence analysis. Site population data have been projected to the target year 2011 using the latest version of the computer code SECPOP2000 (SNL, 2003). SECPOP2000 uses 2000 census data and applies a multiplier to account for population growth and an economic multiplier to account for the value of the dollar to create site data for MACCS2. A multiplier value of 1.1051 from the U.S. Census Bureau was used to account for the average population growth in the United States from 2000 to 2011. Consistent with the approach used in NUREG-1935, the economic values from the database in SECPOP2000 (which uses an economic database based on the year 2002) were scaled to account for price escalation between the years 2002 and 2011. A scaling factor of 1.250 was derived based on the Consumer Price Index (CPI).

Consistent with NUREG-1935, the dose and risk coefficients and relative biological effectiveness values used in this study are based on FGR-13 (EPA, 2002). The dose coefficients allow organ-specific doses to be calculated from exposure to radiation. The risk factors in FGR-13 are based on the risk coefficients for the U.S. population detailed in the BEIR V report (NAS, 1990). As implemented in MACCS2, these factors include seven organ-specific cancers plus residual cancers not accounted for directly. The inhalation factors in FGR-13 were processed to account for a distribution of particle sizes. An activity median aerodynamic diameter of 1 micron was assumed with a log-normal form for the distribution and with a geometric standard deviation of about 2.5. Parameters that relate to acute health effects in this study, as well as much of the nonsite-specific data used for consequence analysis were taken from NUREG/CR-7161 (NRC 2013). All of the input parameters extracted from the expert elicitation are median values.

The FGR-13 coefficients, as implemented in MACCS2, include a dose and dose rate effectiveness factor (DDREF), which has been incorporated in the dose-response modeling for the long-term phase of the offsite consequences and to the dose-response modeling for the early-phase (i.e. the first week) for doses less than 20 rem. This factor accounts for the fact that protracted, low doses are estimated to be less effective in causing cancer than more acute doses. The DDREFs for all cancer types, except for breast, were 2.0; the DDREF for the breast was 1.0, as in NUREG/CR-7009.

To provide perspective on uncertain low-dose health effects, the results also include dose truncations that limit the quantified health effects to those arising from higher doses. Dose truncation values used here include 620 mrem/year (representative background radiation including average annual medical exposures), and 5 rem/year with a 10-rem lifetime cap (based on the Health Physics Society's position that there is a dose below which, because of uncertainties, a quantified risk should not be assigned). This approach is consistent with the approach used in NUREG-1935.

7.1.4 Emergency Response Modeling

The MACCS2 models were set up to calculate exposures in two distinct phases: the emergency phase and the long-term phase. The emergency-phase models calculate the dose and associated health effects to the public, as well as the effects of emergency preparedness actions that protect the public. The emergency phase is defined as the seven day period following the start of the release.

The staff modeled offsite response organization (ORO) decision making based upon the accident sequences, timing, radiological release, and knowledge of response activities and the availability of emergency response technical support. Since actions beyond the emergency planning zone (EPZ)[39] would be 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 SFPS 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.

For each of the accident sequences, staff determined that a General Emergency 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. A release from a SFP with a moderate leak begins earlier than a damage state with a small leak, but these still do not begin until evacuation is well underway or completed within the EPZ.

A number of different protective actions can be modeled in MACCS2. The residents are modeled as groups (known as cohorts) and have different types of protective actions and associated response timings. The actions that can be taken include:

<u>Shelter-in-place (SIP):</u> For certain areas where dose may be reduced below the PAG through sheltering, SIP is modeled as an expected protective action consistent with the emergency plans. In other areas, sheltering can occur prior to evacuation.

<u>General Public Evacuation:</u> Residents evacuate the affected area when the official order to evacuate is received.

<u>Early Evacuation:</u> Residents evacuate after the earthquake, but before the official order to evacuate is received.

<u>Shadow Evacuation</u>: Residents evacuate from areas that are not under an official evacuation order. A shadow evacuation typically begins when a large scale evacuation is ordered (NRC, 2005b). 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, 2008c). In the SFP project, the initiating event is an earthquake that would be felt by residents of the EPZ. The event would be followed with media information related to an accident at the nuclear power plant, widespread loss of power and damage to some buildings. It was assumed that these factors would increase the shadow evacuation to 30 percent of the public in the environs of the plant.

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[39] EPZ in this study refers to the plume exposure pathway EPZ with a radius of about 10 miles from the reactor site. This should not be confused with the ingestion exposure pathway EPZ with a radius of about 50 miles from the reactor site.

Hotspot and Normal Relocation: Models are included in the MACCS2 code to reflect emergency relocation of people from areas that were not included in the evacuation order where the dose exceeds emergency-phase PAGs. Within the MACCS2 calculation, individuals who would be relocated because their projected total committed dose is projected to exceed the protective action criteria are assumed to not receive any additional dose following relocation for the duration of the emergency phase. This relocation dose criterion is applied at a specified time after plume arrival within the affected area and is applied to the entire population within the analysis area, including the nonevacuating cohort (0.5% of the population) within the EPZ. The hotspot and normal dose and time values were developed for each evacuation model. They were established with the assumption that relocation begins after the evacuation is substantially complete which depends on the timing of the first plume for each sequence. For the larger release sequences which affect areas beyond 30 miles, the normal relocation time was assumed to be 12 hours after the hotspot relocation time. This assumption provides time for offsite response organizations to address the higher priority hotspot areas first.

The detailed emergency plans developed for the EPZ provide a substantial basis for expansion of response efforts if necessary (NRC, 1980a). This study identified many potential accident sequences and performed preliminary consequence modeling to establish baseline dose projections as a function of distance. This information was used to develop the appropriate emergency response parameters for the release being modeled. The distance to which the PAG may be exceeded assisted in determining the extent of offsite protective actions and the type of protective actions (sheltering or evacuation) that would be implemented. In the event of model predictions of elevated doses at distances beyond the plume exposure pathway EPZ, a review of the State emergency response plans was performed to determine the types of protective actions that would be implemented in these areas. The results of the dose projections were binned based on the EPA's PAGs (EPA, 1992) to support an efficient use of detailed consequence modeling to determine the potential effects of such accidents. For this analysis, the PAG was considered to be exceeded if the four day projected dose is expected to exceed one rem for a member of the public. Using the dose projections, the three evacuation models presented in the table below were developed for analysis. Detailed information on the implementation of these evacuation models is provided in APPENDIX A: .

Table 32 Summary of Evacuation Models

Evacuation Model	4-Day Dose Projection	EPZ	Area beyond EPZ
1	Small: Does not exceed PAG beyond EPZ.	General public evacuation, including early evacuation of 30% of the public.	Shadow evacuation of 30% of the public from immediately beyond the evacuation area. Hotspot relocation is 5 rem at 4 hours after plume arrival. Normal relocation is 1 rem at 8 hours after plume arrival.
2	Large (48 hour): Exceeds PAG beyond EPZ.	General public evacuation, including early evacuation of 30% of the public.	Shadow evacuation of 30% of the public from immediately beyond the evacuation area. Delayed evacuation to a distance of 30 miles. Shelter in place (SIP) for the 30 to 40-mile area. Shadow evacuation of 20% of the public from the SIP area. Hotspot relocation is 5 rem at 4 hours after plume arrival. Normal relocation is 1 rem at 16 hours after plume arrival. (Rapid implementation of relocation is based on having 48 hours to prepare before release begins)
3	Large (24 hour): Exceeds PAG beyond EPZ.	General public evacuation, including early evacuation of 30% of the public.	Shadow evacuation of 30% of the public from immediately beyond the evacuation area. Delayed evacuation to a distance of 30 miles. Shelter in place of 30 to 40-mile area. Shadow evacuation of 20% of the public from this Shelter-in-place (SIP) area. Hotspot relocation is 5 rem at 26 hours after plume arrival. Normal relocation 1 rem at 38 hours after plume arrival.

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 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. The evacuation time estimate was used to inform the development mobilization times and travel speeds for the public. To model evacuation in MACCS2, each cohort was loaded onto the roadway network at a specified time, and a set of speed values were applied per cohort for the early, middle and late periods of the evacuation. 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. The rate of evacuating the public is 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) and was modeled as a separate cohort.

An assessment of travel distance and time was initially 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 necessarily oriented directly outward from the plant. Consistent with the location of the reference plant, 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 offsite response organizations from neighboring states would adopt the same protective action decisions.

The following general assumptions were applied in this analysis:

- The EPZ is modeled as the area within 10 miles of the site, as an approximation.
- Protective actions would be implemented within the EPZ were an accident to occur.
- Protective actions would be expanded beyond the EPZ as necessary.
- Dose projections would be developed and available to support protective action decisions.
- Residents would expect they cannot return and would take more belongings with them, than what was considered in the past, e.g. NUREG/CR-7009, thereby increasing mobilization times.
- Residents would generally be aware of an impending emergency through media broadcasts.
- For the delayed release sequences in which a releases do not start for more than 24 hours, schools beyond the EPZ would be closed rather than evacuated.
- Evacuees are transported to safe distances.
- There is no loss of power beyond 20 miles. Communications, traffic signals, and emergency alert system messaging are not impacted in this area.

The chosen time period for the emergency phase begins with the initiating event and continues for one week following the initial release. This assumption gives time for the plume to pass and deposit radioactive material onto the ground so that all the calculated acute exposures are captured. The one-week period for the emergency phase is different than the four-day period used for emergency-phase dose projections, which were used to inform the evacuation models. The four-day period was chosen to be consistent with the EPA PAGs (EPA, 1992).

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 chosen to be radially outward to simplify modeling of evacuation in these areas. Speed adjustment factors were applied at the grid element level to speed up vehicles in the rural uncongested areas and to slow vehicles in more urban settings in which the modeling indicates that speeds are lower than the average values used in the analyses.

The MACCS2 potassium iodide (KI) model used in this analysis assumes that KI would be distributed only within the EPZ. Half the residents within the EPZ are assumed to have access to their KI and to take it within the specified timeframe.

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 an evacuation-speed multiplier to reduce travel speed when precipitation is occurring (indicated from the meteorological data file). The evacuation speed multiplier factor was set to be 0.7, which effectively slows down the evacuating public to 70 percent of the fair-weather travel speed when precipitation exists.

The analyses of the effect of the seismic event on emergency response developed for NUREG-1935 were applied in this analysis, as the reference plant in this study was one of the plants studied in NUREG-1935. The evaluations of the potential failure of roadway infrastructure conducted for NUREG-1935 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 roads would have 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 is assumed to cause the loss of all onsite and offsite power within the EPZ, which can affect the response timing and actions of the public. Sirens would be sounded following the GE declaration, and because the reference plant will have a fully backed up siren system in 2013, it is assumed sirens sound for this analysis. The residents within the EPZ would have felt the earthquake, which effectively serves as the initial warning; however, the loss of power would affect the number of residents receiving instructions via emergency alert system messaging. It is expected that the residents use multiple methods of communication, such as cell phones, telephones, websites (where power is available), 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 and that most intersections are controlled with stop signs. The loss of power would cause traffic signals to default to a four-way stop mode, which is less efficient than normal signalization. It is expected that emergency response personnel would respond to these intersections and direct traffic as indicated in the site evacuation time estimate. Therefore, the loss of signalization should have a limited impact on the evacuation. It is assumed that at distances beyond 20 miles, there is no loss of power and traffic signals, and emergency alert system messaging is not impacted.

7.1.5 Long-Term Protective Action Modeling

The long-term phase is the period following the seven-day emergency phase and is modeled for 50 years. The 50 year duration of the long-term phase has been chosen to provide a reasonable time period for calculating consequences from exposure for the average person. Exposure is mainly from external radiation from trace contaminants that remain after the land is decontaminated, or in lightly contaminated areas where no decontamination was required. However, internal exposures may also occur during this period, including inhalation of resuspended radionuclides and ingestion of food and water with trace contaminants. Depending on the relevant PAGs and the level of radiation, food and water below a certain limit could be considered adequately safe for ingestion, and lightly contaminated areas could be considered habitable.

A long-term cleanup policy for recovery after a severe accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be decided by a number of local, state, and federal jurisdictions and would most likely 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 may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations. Three protective actions were modeled to occur for contaminated land during the long-term phase: interdiction, decontamination, and condemnation. As used in the MACCS2 model, interdiction and condemnation refer to the relocation of people from contaminated areas according to the habitability criterion. Interdiction is defined in the MACCS2 model as temporary relocation during which the contamination levels are reduced by decontamination, natural weathering, and radioactive decay. Condemnation is defined in the MACCS2 model as a permanent relocation when contamination levels cannot be adequately reduced by decontamination, natural weathering, and radioactive decay.

Decontamination is modeled in a manner consistent with both NUREG-1150 and NUREG-1935. Two levels of decontamination (a decontamination factor of 3 and 15) are each assumed to take one year, but the cost of the higher decontamination factor (15) is assumed to be greater, reflecting the greater effort needed to achieve the higher level of decontamination. This study uses the values in NUREG/CR-7009 for the cost of decontamination. During the decontamination period, the land is temporarily interdicted (e.g. the population is temporarily relocated), and may be interdicted for an additional period to allow for radioactive decay and natural weathering to reduce contamination levels if needed to restore habitability. If land cannot be restored to habitability in 30 years, the MACCS2 model defines the land as condemned and residents are modeled to not return during the long-term phase. The MACCS2 models assume that decontamination will only take place if it is projected to make 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 is projected to be higher than the land value, the individuals on that land are assumed to be permanently relocated. Because both the land values and the level of decontamination affect decisions on whether contaminated areas can be restored to habitability, they affect predicted long-term doses, health effects, and economic costs.

Site-specific values are used to determine long-term habitability, whereas farmability is defined to be consistent with NUREG-1150. For 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, consistent with the location of the reference plant, the analysis includes the State of Pennsylvania position using a habitability criterion of 500 mrem per year beginning in the first year, which is the value that is used for this study. For consistency and practicality reasons, the same standard for estimating habitability is applied to all affected areas in this study. The values used to define farmability were taken from NUREG-1150. During the year of the accident, the allowable committed dose equivalent from consumption of dairy products to an organ or tissue is 2.5 rem (7.5 rem for the thyroid), as well as an additional dose of 2.5 rem (7.5 rem for the thyroid) for all other foods. In subsequent years, the maximum allowable dose to the organ or tissue from all foods, including dairy products, is 500 mrem (1.5 rem for the thyroid). Agricultural lands projected to be contaminated to such an extent that agricultural products would exceed these levels are defined to be unfarmable, and the crops growing on these lands at the time of the accident are assumed to be disposed. No farming is allowed until the farmability criterion is satisfied.

7.2 Offsite Consequence Results

Several consequence metrics have been selected to characterize the impacts resulting from a severe spent fuel pool accident. Individual risk of early and latent cancer fatality, as well as societal risk of latent cancer fatalities, are measures of the radiological health impact of the accident and consistent with NRC's safety goals (NRC, 1986). In this study, collective dose is used as a surrogate for the societal impact of latent cancer fatalities. In addition, certain metrics that would influence the values considered by the NRC in regulatory analysis and documented in NUREG/BR-0058, such as measures of offsite property damages, the number of displaced individuals (either temporarily or permanently), and the extent over which such actions may be needed, are also presented. These metrics provide a benchmark for understanding the nature and extent of a severe spent fuel pool accident. These measures are subject to considerable uncertainty, as the details of how long-term protective actions would be carried out would have a significant effect on the actual values reported herein.

All results presented in this section are conditional upon a pool leak following a specified severe (0.71g peak ground acceleration) seismic event on the SFP at the reference plant. In the event of a pool leak following a severe seismic event, a number of potential outcomes could occur, depending upon when in the operating cycle the event occurred, the severity of the leak, and whether effective mitigation (in the form of either makeup water or pool sprays) was able to be successfully deployed prior to the beginning of the release. Staff has evaluated the likelihood of these different conditions. The relative likelihood of a seismic event during a particular operating cycle phase is simply proportional to the duration of the phase. The relative likelihood of significantly different leak rates is discussed in Section 4. Because these probabilities can be quantified with a reasonable degree of certainty, the offsite consequence results are weighted by the relative likelihood of these factors to yield an average over the different operating cycle phases and leak rates.

In contrast, the likelihood of successful deployment of 10 CFR 50.45(hh)(2) mitigation has not been quantified. NRC staff judgment is that the likelihood of successful mitigation can in many cases be high, but that it is affected by a number of factors that are difficult to quantify (see Section 5.3). Related to this, a human reliability assessment (HRA) is provided in Section 8. Although the HRA does not provide a quantitative value required to determine the overall likelihood of mitigation, it does provide significant insights into the likelihood of mitigation during this seismic event for certain damage states. To quantify the overall likelihood of successful mitigation, a PRA type analysis would be required. For this reason, the results of the study are presented as a range of mitigation effects related to successfully deployed mitigation and mitigation that is unsuccessful for 3 days.

This analysis examines the relative effects of a low-density and a high-density fuel loading configuration. Therefore, results are reported for two configurations, those being a high-density loading case with a 1x4 loading pattern and for a low-density loading case with a mixture of 1x4 and checkerboard loading patterns, as portrayed in Figure 44 through Figure 48.

In this chapter, the results for each selected metric are discussed for each loading configuration (high-density and low-density). In addition, the factors that affect the results and how those results vary with dose truncation assumptions and with distance are discussed. To the extent possible, the relationship between the results presented here and the results obtained in previous studies is discussed.

Table 33 Overall Consequence Results

SFP Fuel Loading	High Density (1x4)		Low Density	
Seismic Hazard Frequency[1] (/yr) (PGA of 0.5 to 1.0g)	1.7E-05		1.7E-05	
50.54(hh)(2) Mitigation Credited	Yes	No	Yes	No
Conditional[2] Probability of Release	0.036%	0.69%	0.036%	0.69%
Hydrogen Combustion Event	"Not Predicted"	"Possible"	"Not Predicted"	"Not Predicted"
Conditional[3] Consequences (Release Frequency-Averaged[4])				
Cumulative Cs-137 Release at 72 hours (MCi)	0.26	8.8[(8)]	0.19[(7)]	0.11
	Measures Related to Health and Safety of Individuals			
Individual Early Fatality Risk	0	0	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles	3.4E-04	4.4E-04	3.4E-04	2.0E-04
	Measures Related to Cost Benefit Analysis			
Collective Dose (Person-Sv)	47k	350k	47k	27k
Land Interdiction[6] (mi^2)	230	9,400	230	170
Long-term Displaced Individuals[6]	120k	4,100k	120k	81k
Consequences per year (Release Frequency-Weighted[4])				
Release Frequency (/yr)	6.1E-09	1.2E-07	6.1E-09	1.2E-07
	Measures Related to Health and Safety of Individuals			
Individual Early Fatality Risk (/yr)	0	0	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles (/yr)	2.1E-12	5.2E-11	2.1E-12	2.4E-11
	Measures Related to Cost Benefit Analysis			
Collective Dose (Person-Sv/yr)	2.9E-04	4.1E-02	2.9E-04	3.2E-03
Land Interdiction[6] (mi^2/yr)	1.4E-06	1.1E-03	1.4E-06	2.0E-05
Long-term Displaced Individuals[6] (Persons/yr)	7.1E-04	4.9E-01	7.1E-04	9.5E-03

[1] Seismic hazard model from USGS (Peterson et al., 2008)
[2] Given specified seismic-event occurs
[3] Given atmospheric release occurs
[4] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions (as applicable); additionally, "release frequency-weighted" results are multiplied by the release frequency.
[5] LNT and population-weighted (i.e., total amount of latent cancer fatalities predicted in a specified area, divided by the population that resides within that area.)
[6] 1st year post-accident; calculation uses a dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51
[7] Mitigation can moderately increase release size (see Section 6.3); the effect is small compared to the reduction in release frequency.
[8] Largest releases here are associated with small leaks (although sensitivity results show large releases are possible from moderate leaks). Assuming no complications from other SFPs/reactors or shortage of available equipment/staff, Section 8 shows that there is a good chance to mitigate the small leak event.

7.2.1 Individual Early Fatality Risk

For all scenarios, no offsite early fatalities attributable to acute radiation exposure are predicted to occur. Due to radioactive decay, spent fuel pools tend to have significantly less shorter-lived radionuclides (e.g. I-131) than reactors. Despite this, in at least one case that was analyzed, doses close to the site did reach levels that can induce early fatalities. Therefore, the potential (although remote) for early fatalities exists. However, emergency response as treated in this study effectively prevents any early fatality risk from acute radiation exposure, at least in part because the modeled accident progression results in releases that are long compared to the implementation of emergency response in the areas of most concern.

The projections of no early fatalities in this study is lower than that reported in some previous studies of risks from spent fuel pool accidents, such as NUREG/CR-6451 and NUREG-1738, and consistent with the earlier studies documented in NUREG-1353. Tables 4.1 and 4.2 of NUREG/CR-6451 project anywhere from approximately one to one hundred early fatalities within a 500 mile radius in the event of an accident involving the full spent fuel pool, with the higher values associated with high release fractions. NUREG-1738 (Table 3.7-1 and Table 3.7-2) reported similar values, ranging from no fatalities for low Ruthenium source terms with early evacuation to up to 192 early fatalities for an accident shortly (30 days) after shutdown with high Ruthenium source terms and late evacuation. NUREG-1353 does not provide quantitative estimates of early fatality risk but states that "…there are no "early" fatalities and the risk of early injury is negligible". On balance, the scenarios analyzed here are consistent with the lower end of the reported range from previous studies, in that no early fatalities are projected to occur.

7.2.2 Individual Latent Cancer Fatality Risk

Despite the large releases in certain circumstances, the risk of latent cancer fatality to the average individual within 10 miles of the plant is low. When averaged over the likelihood of different event timings and leak sizes, the conditional risks within 10 miles are in the 1E-04 to 1E-03 range for cases both with and without successful 50.54(hh)(2) mitigation and for high-density and low-density cases, when using an LNT dose response model. This range does not appreciably increase even if the releases for different leak sizes or operating cycle phases are shown separately.

Individual latent cancer fatality risk is low because:

- The predicted release frequency of this event is very small
- Protective actions, especially those for long-term chronic doses, are estimated to avert significant amounts of public exposure.

Because of the nature of the event, this risk is predominantly from long-term chronic exposures. With effective long-term protective measures (e.g. temporary and permanent land interdiction), essentially no individuals receive any long-term risks greater than those associated with the dose limits for protective actions. Therefore, independent of the release magnitude of the event, these dose limits form an upper limit to individual long-term risk. In addition, emergency response is assumed to be very effective in evacuating and relocating the public. For instance, individuals within the 0-10 mile distance (representative of the plume exposure pathway EPZ) essentially only receive LCF risk if they return to low risk, habitable areas. The conditional individual LCF risks within ten miles are comparable to or lower than the projections from earlier studies of spent fuel pool accident risk. For example, NUREG-1738 reports conditional individual latent cancer fatality risks ranging from 8E-4 to 8E-2 for a range of initiating events where large seismic events contributed the most to the overall estimate of risk. These conditional risks were driven largely by the previous estimates of ruthenium volatility and by the effectiveness of evacuation.

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 and related to a single spent fuel pool, staff concludes that since these risks are several orders of magnitude smaller than 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).

Because the health effects that would be induced by low dose radiation are uncertain, staff performed a sensitivity analysis to understand how the risks would change if computed health risks were limited to those arising from higher doses. The upper truncation level used in this sensitivity analysis corresponds to a treatment consistent with the HPS position statement (5 rem annually and 10 rem lifetime). The second truncation level corresponds to the average annual dose to the public from medical and background radiation exposures in the United States (620 mrem annually).

Using truncation levels that do not quantify the effects of doses below the dose levels chosen here significantly reduces the estimated individual LCF risk. This is because individual LCF risk using an LNT dose response model mainly comes from doses less than those specified in protective action guidelines. Table 34 (which shows risk to residents living within ten miles, not including risk from ingestion or risk to decontamination workers) shows the use of the dose truncations that are analyzed here lowers the estimated individual LCF risk within 10 miles by several orders of magnitude. Because the dose truncations are greater than the dose limits for land interdiction, it is difficult for doses from the long-term phase to contribute to the quantified LCF risk. Therefore, emergency-phase exposures play a more significant role in the doses that exceed the truncation levels. However, the amount of early phase exposures that exceed the dose truncations is very small within 10 miles because emergency response is effective in protecting the evacuees.

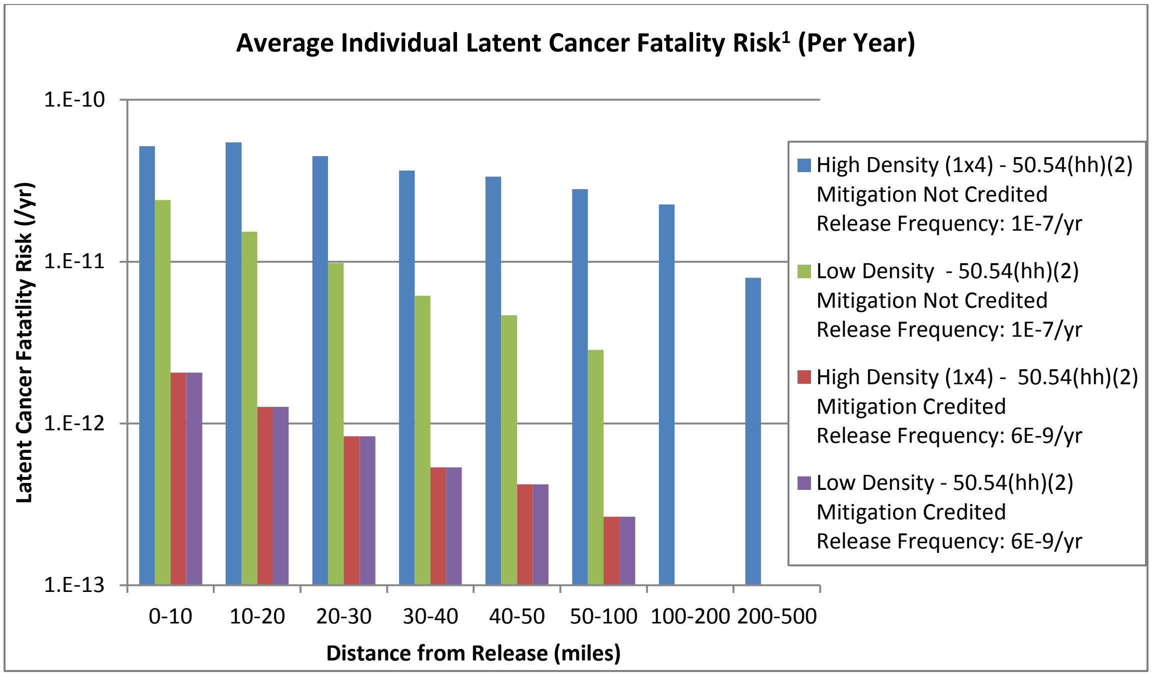
A number of factors can affect quantified individual LCF risks, particularly the very small values from dose truncation results. These include potential variations of the real application of protective actions, different protective action levels, or consideration of ingestion doses. Nevertheless, the overall conclusions that with an LNT calculation, individual LCF risk is mainly from long-term chronic exposures, and that dose truncations significantly lower the estimated individual LCF risk, remain valid.

Table 34 Dose-Response Model Results (LNT) and Dose Truncation Comparison

Dose-Response	High Density (1x4)		Low Density	
50.54(hh)(2) Mitigation Credited	Yes	No	Yes	No
Conditional[1] Individual Latent Cancer Fatality Risk Within 10 Miles (Release Frequency-Averaged[2])				
Linear, No Threshold	3.4E-04	4.4E-04	3.4E-04[(3)]	2.0E-04
620 mrem/yr truncation	6.1E-08	1.2E-07	6.1E-08[(3)]	3.4E-08
5rem/yr or 10rem lifetime truncation	1.8E-08	1.4E-07	1.8E-08[(3)]	5.6E-09
Individual Latent Cancer Fatality Risk Within 10 Miles (/yr) (Release Frequency-Weighted[2])				
Linear, No Threshold	2.1E-12	5.2E-11	2.1E-12	2.4E-11
620 mrem/yr truncation	3.8E-16	1.4E-14	3.8E-16	4.0E-15
5 rem/yr or 10 rem lifetime truncation	1.1E-16	1.6E-14	1.1E-16	6.6E-16

[1] Conditional on a release occurring
[2] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions; additionally, “release frequency-weighted” results are multiplied by the release frequency.
[3] Mitigation can moderately increase release size (see Section 6.3); the effect is small compared to the reduction in release frequency.

While individual latent cancer fatality risk is very low, it decreases slowly with distance, particularly for large releases such as may occur from an accident in a high-density pool with unsuccessful mitigation for 3 days. This is because offsite release models predict significant spread of contamination to far distances, mainly because of the slow deposition of aerosols from the plume. Increasing the magnitude of the release extends the range over which a plume can travel before the radioactive inventory of the plume is significantly depleted by deposition. Furthermore, because protective actions such as land interdiction are modeled to occur wherever the model predicts that the dose limits are exceeded, most distances are held to comparably low levels of individual LCF risk regardless of the magnitude of the deposition, as was seen in the results for individual LCF risks in Table 33. This can be seen in Figure **96**, which like the table, is also weighted by the release frequency.



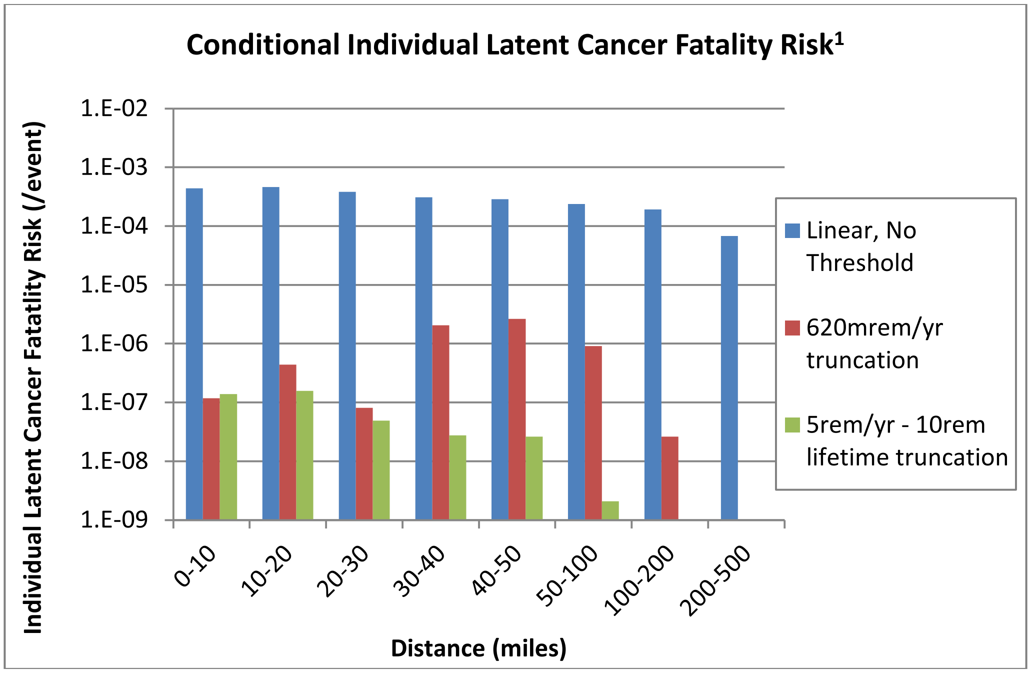
[1]Linear-no threshold, weather-averaged, release frequency-weighted, and population-weighted

Figure 96 Individual Latent Cancer Fatality Risk (per year)

The accuracy of atmospheric transport and deposition (ATD) models (e.g., the Gaussian plume segment model used in MACCS2) tend to decrease with distance, and therefore the results should be viewed with caution at longer distances. However, MACCS2 has been benchmarked against other ATD models, and the staff considers the broad conclusion remains valid—that risks would be small but drop slowly with distance in the event of large releases.

For affected areas with large populations, severe accidents can result in significant numbers of latent cancer fatalities. However, this should be weighed against the likelihood of the accident. Furthermore, because the individual doses are relatively small, this would be a small fraction of all cancer fatalities from all causes. This risk mainly comes from doses that are constrained to be less than dose limits for protective actions from an LNT dose response model; dose truncations predict significantly fewer latent cancer fatalities.

Figure 97 compares the quantified individual LCF risk for different dose truncations and for a variety of reported distance ranges for a high-density (1x4) configuration with unsuccessful mitigation for 3 days. The figure shows that dose truncation significantly lowers the quantified LCF risk. This is similar to Table 34; however this figure shows risks for a range of distances.



[1] High Density (1x4)—Unsuccessful Mitigation, weather-averaged, release frequency-averaged, and population-weighted

Figure 97 Conditional Individual LCF Risk for Different Dose Truncations and Distances

The effect of protective actions can be observed from Figure 97. For the release modeled in this scenario, the LCF risk within 10 miles is slightly less than at the 10–20 mile range. This small variation in risk with distance is because different modeled protective actions (such as evacuation, sheltering, early relocation, decontamination, temporary interdiction, and permanent condemnation) will depend on the level of contamination expected at a particular location. For example, the higher contamination levels closer to the source may result in relatively longer periods of relocation. Because no exposure to these populations would occur during this period these individuals could have lower overall doses than individuals further away under some situations. The 620 mrem annual dose truncation in particular demonstrates the effect of reduced individual LCF risk at these distances compared to longer distances. In Figure 97, the 620 mrem annual dose truncation best illustrates the effect of emergency response because this dose truncation does not quantify the significant contributions from chronic, long-term exposures.

7.2.3 Land Contamination

As the values in Table 33 suggest, conditional on a release (with a frequency of 1E-7 per year, or lower) occurring, the total land contamination area can be considerable. The low-frequency, large releases are significantly affected by hydrogen combustion events, which are currently predicted in some high-density loading situations without successful mitigation for 3 days, but not in other scenarios. For relatively small releases from a SFP, the extent of contaminated land could range to hundreds of square miles. For a large release, such as a release from a high-density pool without successful deployment of 50.54(hh)(2) mitigation that leads to a hydrogen combustion event, the amount of contaminated land can be two orders of magnitude higher (Table 35 partially reflects this range, although it reports average values). The levels of potential land contamination in the event of a release should be weighed against the likelihood of the accident. When the amount of contaminated land is weighted by the annual likelihood of occurrence (as seen in Table 33), the expected impact is relatively low. In addition, only a small portion of these interdicted areas are expected to be permanently interdicted, as the level of contamination is expected to significantly decrease with time as decontamination, radioactive decay, and weathering occur.

The amount of land affected depends on the dose criterion selected. For the purposes of this study, land contamination is defined as the area impacted by protective actions, specifically either temporary or permanent land interdiction. Because of the location of the reference plant, the particular protective action level the study uses is the Pennsylvania standard for habitability (dose limit of 500 mrem each year). The study uses this measure to estimate land contamination starting in the first year after a potential severe accident. In reality, the annual dose limit for what is considered “habitable” can change when crossing a state boundary. However, for consistency and practicality reasons, the same standard for estimating land contamination area is applied to all affected areas in this study, and the measure chosen for this study is only meant to be an indicator of land contamination.

Consistent with the observations of a relatively slow decline in individual latent cancer fatality risk with distance, the results of the analysis indicate that protective actions such as temporary relocation may be needed at long distances. The table below displays an average amount of interdicted land within different distances for high- (1x4) and low-density fuel loading.

Table 35 Average Land Interdiction* (square miles per event)

SFP Loading Pattern	High Density (1x4)		Low Density	
10 CFR 50.54(hh)(2) mitigation credited	Yes	No	Yes	No
Release Frequency (/yr)	6.1E-9	1.2E-7	6.1E-9	1.2E-7
0-50 miles	210	1,200	210**	160
0-100 miles	230	3,100	230**	170
0-500 miles	230	9,400	230**	170

* Weather-averaged and release frequency-averaged; Dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51

** Mitigation can moderately increase release size (see Section 6.3); the effect is small compared to the reduction in release frequency.

A release in the high-density fuel loading situation without successful 50.54(hh)(2) mitigation is capable of large releases, and therefore an average release from this situation is capable of causing significantly more land contamination at longer distances than in the other situations. In contrast, releases from situations with low density fuel loading (and/or successfully deployed 50.54(hh)(2) mitigation equipment) cause a relatively smaller amount of land contamination beyond 50 miles, and none beyond 100 miles when using land interdiction as a measurement of land contamination. This is because on average, a release in these situations contaminates significantly less area. However, because of the release magnitude of any of the analyzed SFP releases, the total amount of land contamination that remains within ten miles is relatively small.

On land contamination, past results are expected to be broadly consistent with this study. However some previous studies did not report land contamination and some reported different metrics for estimating areas, so a direct comparison is not possible. NUREG/CR-6451 reports values for condemned farmland that includes hundreds of square miles within a 50-mile radius and thousands of square miles within a 500 mile radius, albeit for a full core off-load. NUREG-

1353 reports values for land contamination based on NUREG/CR-4982 that range into the hundreds of square miles, albeit largely within a 50-mile radius of the plant. These differences, as well as different choices for the land contamination criteria that can significantly affect the estimated areas, make a quantitative comparison less meaningful. However, it is clear that both this study and past studies have predicted that SFP accidents can lead to significant land contamination.

7.2.4 Displaced Individuals

Consistent with the results for land contamination, relatively large numbers of people may be impacted following a large release from a spent fuel pool. Displaced individuals, also known as relocated individuals, are people who are predicted to be temporarily or permanently relocated due to interdiction of contaminated land, based on the dose limit for land interdiction starting in the first year following an accident. These individuals are not necessarily the same as evacuees, who evacuate during the emergency phase (although an individual could be both of these).

Conditional on a release (with a frequency of 1E-7 per year or lower) occurring, the total number of temporarily relocated individuals could be considerable. For relatively small releases of an SFP, the number of displaced individuals could range into the hundreds of thousands. For a large release, which is predicted in some high-density loading situations early in the operating cycle without successful 50.54(hh)(2) mitigation, the number of displaced individuals can be two orders of magnitude higher. (Table 36 partially reflects this range, although it reports average values).

Also consistent with the observations related to the amount of land contamination with distance, the results of the analysis indicate that protective actions such as temporary relocation may be needed at long distances. The table below displays the average number of displaced individuals for different distances for high (1x4) and low density fuel loading.

Table 36 Average Number of Long-term Displaced Individuals* (per event)

SFP Loading Pattern	High Density (1x4)		Low Density	
10 CFR 50.54(hh)(2) mitigation credited	Yes	No	Yes	No
Release Frequency (/yr)	6.1E-09	1.2E-07	6.1E-09	1.2E-07
0-50 miles	100k	780k	100k**	72k
0-100 miles	120k	2,000k	120k**	81k
0-500 miles	120k	4,100k	120k**	81k

* Weather-averaged and release frequency-averaged; Dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51

** Mitigation can moderately increase release size (see Section 6.3); the effect is small compared to the reduction in release frequency.

These estimates should be weighed against the likelihood of the accident. When the number of displaced individuals is weighted by the annual likelihood of occurrence (with a frequency of 1E-7 per year or lower; as seen in Table 33), the expected impact is relatively low. An average release in the high-density fuel loading situation without successful 50.54(hh)(2) mitigation causes significantly more relocation at longer distances than in the other situations because it is capable of larger releases. In contrast, releases from situations with low density fuel loading (and/or successfully deployed 50.54(hh)(2) mitigation equipment) cause a relatively small amount of relocation beyond 50 miles, and none beyond 100 miles because on average, a release from these scenarios contaminates significantly less area. For all situations, the number of displaced persons from the 0 to 10 mile area is relatively small because the number of people living on this area is relatively small.

7.3 Offsite Consequence Comparison

A goal of the study is to compare the results of the scenario-specific, high- and low- density fuel loading seismic events. To facilitate the comparison, results of different scenarios are compared to each other by dividing the results from one scenario by another scenario, for a variety of consequence metrics. The ratios of the consequence metrics are indicators of the scenario specific safety benefit between the two scenarios.

These comparisons should consider the scenario release frequency as well as conditional on a release occurring, appropriate. In the first comparison below, the high-density (1x4) fuel loading and low-density fuel loading had the same release frequency. Therefore, for this comparison, there is no additional reduction when the likelihood of occurrence is also considered.

Table 37 Consequence[1] Comparison – High (1x4) Density / Low Density Loading without Successful 50.54(hh)(2) Mitigation

SFP Fuel Loading	High Density (1x4)	Low Density	Reduction Factor (dimensionless)
Release Frequency	1.2E-07	1.2E-07	1.0
Individual Latent Cancer Fatality Risk[2] within 10 Miles	4.4E-04	2.0E-04	2.1
Collective Dose (Person-Sv)	350k	27k	13
Land Interdiction[3] (mi^2)	9,400	170	56
Long-term Displaced Individuals[3] (Persons)	4,100k	81k	51

[1] Conditional on a release occurring (frequency of 1E-7 per year, or lower); results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions
[2] Linear-no threshold, population-weighted
[3] 1st year post-accident; calculation uses a dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51

The most significant reduction factor in a low-density loading pattern is in the reduction in land interdiction and associated displaced individuals. This is because these consequences are more closely related to the size of release than the other results. In addition, a small amount of contamination can occur before land reaches the dose level for interdiction. This dose threshold effect means smaller releases more-than-proportionally reduce the amount of land interdiction.

The reduction in collective dose (and latent cancer fatalities) in a low density loading pattern is also due to the smaller release magnitude. This reduction is significant, although not as significant as the reduction in land interdiction. This is because larger releases are predicted to have considerably more temporary and permanent interdiction to protect the public. This is especially true at shorter distances, as indicated by the reduction factor for LCF risk for 0-10 miles. One significant reason a smaller release magnitude is expected in the low-density loading situations is because hydrogen combustions are currently not predicted in these situations.

The next table reports the reduction of the consequences with successful deployment of 50.54(hh)(2) mitigation equipment. Because successfully deployed mitigation can prevent fuel release, it affects the reduction factors for release frequency-weighted consequences (per year) differently than consequences conditional on a release occurring. For brevity, the consequence values are not displayed here, although can be seen in the previous section.

Table 38 Consequence Comparison – Unsuccessful/Successful Deployment of 50.54(hh)(2) Equipment

Fuel Loading Density	High Density (1x4)	Low Density
	Reduction Factor (dimensionless)	
Change in Release Frequency (/yr)	19	19
	Conditional[1] Consequences (Release Frequency-Averaged[2])	
Type of Consequence	Reduction Factor (dimensionless)	
Individual Latent Cancer Fatality Risk[3] within 10 Miles	1.3	0.61
Collective Dose (Person-Sv)	7.4	0.59
Land Interdiction[4] (mi^2)	40	0.72
Long-term Displaced Individuals[4] (Persons)	36	0.70
	Consequences per year (Release Frequency-Weighted[2])	
Type of Consequence	Reduction Factor (dimensionless)	
Individual Latent Cancer Fatality Risk[3] within 10 Miles (/yr)	25	12
Collective Dose (Person-Sv/yr)	140	11
Land Interdiction[4] (mi^2/yr)	780	14
Long-term Displaced Individuals[4] (Persons/yr)	690	13

[1] Conditional on a release occurring (frequency of 1E-7 per year, or lower)
[2] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions; additionally, "release frequency-weighted" results are multiplied by the release frequency.
[3] Linear-no threshold, population-weighted
[4] 1st year post-accident; calculation uses a dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51

For both high- and low-density fuel loading, the release frequency was reduced by about a factor of 20 with successful deployment of 50.54(hh)(2) mitigation.

Conditional on a release occurring (middle portion of Table 38), successful deployment of 50.54(hh)(2) mitigation reduces all of the average consequences of the high-density fuel loading pattern, although to varying degrees. These varying degrees of consequence reductions are similar to that predicted in Table 37 for using a low-density loading pattern, although to a somewhat lesser extent. A significant portion of this reduction may be attributable to the fact that hydrogen combustions are not predicted with successful deployment of 50.54(hh)(2) equipment.

Contrary to what might be expected, 50.54(hh)(2) mitigation is predicted to slightly increase the average conditional consequences of a release from a low-density fuel loading pattern. While successful deployment of 50.54(hh)(2) equipment is usually effective at preventing releases, it is not as effective at mitigating release from the low-density fuel loading pattern when deployed in a capacity specifically to provide makeup water through injection, as sometimes assumed. In these conditions, release from a SFP can sometimes be somewhat larger with deployed mitigation. In addition, the situations for which 50.54(hh)(2) equipment prevented release for the low-density loading events were the situations with the smallest release magnitudes, which has the non-intuitive effect of increasing the average consequence of a release.

The bottom section of Table 38 shows the combined benefit of prevention and mitigation from successfully deployed 50.54(hh)(2) equipment, which combines the reduction factors of a lower release frequency with the changes in the average consequences of a release. In the high-density loading situation, the overall benefit of 50.54(hh)(2) equipment is very significant (more than a factor of 100 reduction in most of the risk metrics) if successfully deployed. For low-density loading, the deployment of the 50.54(hh)(2) equipment has a somewhat negative effect on the average conditional consequence; however, this is far outweighed by the benefit it provides in preventing release.

8. HUMAN RELIABILITY ANALYSIS

Consistent with the limited scope of the SFPS, a limited scope human reliability analysis (HRA) was performed, to develop initial insights into the likelihood of successful operator actions to prevent spent fuel damage for the specific seismic event and consequence scenarios studied. A full scope HRA would primarily be useful as part of a PRA analysis. A PRA would necessarily consider a much broader scope than the SFPS. Such a scope would include the likelihoods of all initiating events, the plant damage states for the two reactors and spent fuel pools, the availability of all installed or portable mitigation equipment, and the availability of on-site (and possibly off-site) personnel. Thus the limited scope HRA results presented here must be viewed from the context of their specific assumptions, including assumptions that remove likely complexities that impact operator performance.

In this context, to perform an HRA for this study, successful mitigation must be defined. For this HRA, mitigation success is defined as preventing radioactive release from the fuel rods of the Unit 3 SFP fuel (or gap release). The reference plant site has two reactor units (Unit 2 and Unit 3) in operation. The status of the Unit 2 and 3 reactors, the Unit 2 SFP, and the other plant SSCs would affect Unit 3 SFP mitigation, but successful mitigation, as defined in this analysis, is only determined by the Unit 3 SFP fuel status.

The effective SFP mitigation strategies, to prevent fuel overheating and release of radioactive material from the damaged fuel rods, are to either inject or spray water into the SFP from the refueling floor. The refueling floor on top of the reactor cavity is part of the primary containment that insulates the refueling floor from the reactor. In situations involving reactor damage with intact primary containment, access to the refueling floor is still possible. Over the refueling floor is the secondary containment which is a light-weight steel structure. During an SFP accident, the secondary containment can reduce the radioactivity released from the SFP to the environment. During refueling, the primary containment head is removed to expose the reactor cavity. The reactor vessel head is also removed for defueling and refueling. Therefore, during a refueling outage, the refueling floor is no longer insulated from the reactor. Heat and radiation generated from the reactor would directly affect the work environment on the refueling floor. In addition to the strategies of spraying water from the refueling floor to the SFP, strategies to spray water from outside of the secondary containment (e.g., by ladder fire trucks) to the secondary containment or the SFP (through containment breaches) are available. However, as these strategies are aimed at mitigating releases to outside of the secondary containment and not at preventing fuel overheating, they are not credited in this HRA study.

The SFPS ran a number of computer simulations to understand the effects of a set of factors affecting SFP fuel radioactive release after an earthquake damaged the normal SFP cooling system. The set of factors include SFP leak size, spent fuel loading pattern, OCP, mitigation deployment, mitigation flow rate, and types of mitigation (i.e., injection or spray). These simulations generated information that served as the foundation for the HRA study. Section 8.1 summarizes the SFPS results relevant to the HRA study and discusses their implications to the HRA study. Section 8.2 discusses the staffing, mitigation equipment, strategies, and procedures of the reference plant relevant to the SFP mitigation. Section 8.3 discusses the HRA study framework, scope, and approach. The conduct of an HRA is normally done in conjunction with a PRA to identify each event sequence (i.e., scenario) following an initiating event. For each event sequence, the PRA model would explicitly specify the status (i.e., success or failure) of each component, system, and human action that affects the event sequence's progression and end consequence. For this reason, the development of a PRA model would require significant effort. For this limited scope HRA study, a detailed PRA (i.e., using event trees branched to represent various possible scenarios) was not performed. Instead, scenarios are classified based on the status of a few key SSCs (e.g., electric power availability, and the status of the Unit 3 reactor and primary containment). The Unit 3 reactor and primary containment status are included in the HRA study because of their significant effects on the Unit 3 refueling floor work environment (i.e., where the SFP mitigation strategies are performed). Table 39 summarizes the scope and assumptions applied to the HRA study. Section 8.4 summarizes the insights of this study.

Table 39 The scope and assumptions of the HRA study

#	Scope and assumptions	Notes
1	Success criterion: prevent radioactive release from the fuel rods of Unit 3 SFP fuel	- Do not include strategies designed to reduce radioactivity released to the environment. The effective mitigation is to inject or spray water into the SFP. - The status of fuel in the Unit 2 and 3 reactors and Unit 2 SFP are not considered in the success criterion.
2	Classify plant damage states as a result of the earthquake and estimate the mitigation failure probability for each plant state.	The probabilities of the plant damage states as a result of the earthquake were not estimated.
3	The installed equipment for SFP mitigation is not available. Operators have to use the 10 CFR 50.54(hh)(2) equipment for mitigation.	If the installed equipment (e.g., fire system and residual heat removal system) is available, the SFP mitigation would have a much higher success likelihood than this study's estimates.
4	The SFP mitigation uses the minimum flow rate specified in NEI 06-12 guidance for complying with 10 CFR 50.54(hh)(2).	The actual flow rate is expected to be higher than the minimum NEI recommended flow rate (i.e., 500 gpm of injection or 200 gpm of spray)
5	10 CFR 50.54(hh)(2) equipment and water sources for Unit 3 SFP mitigation is available.	Earthquake-caused damage to the 10 CFR 50.54(hh)(2) equipment is not included in the study. Further, dividing equipment to mitigate multiple reactor and SFP problems is not considered.
6	Sufficient plant staff is available to perform the Unit 3 SFP mitigation.	Staffing information is discussed but different staffing scenarios are not factored into the analysis. For multiple reactors and SFPs damaged by the hypothetical earthquake, the personnel sufficiency would be a key factor affecting mitigation success.
7	Non-plant, off-site support (e.g., off site fire trucks) are not considered.	For an SFP event, the primary function of off-site support is to keep radioactivity release within the plant site. Off-site support for preventing SFP fuel rod damage is not credited.

8.1 Summary of Spent Fuel Pool Study Analysis Results Relevant to Human Reliability Analysis

8.1.1 High Level Scenarios Classification

The SFPS concludes that the following four scenarios do not lead to gap release with a 72-hour-truncated simulation time (see Table 40 for a tabulate classification):

(1) Boil off Scenario with No SFP Leaks. As mentioned earlier, the SFP water level in this scenario would take more than 7 days to decrease to the top of the fuel rack. Because of the long time available for response, multiple opportunities are available to prevent damage to the SFP; therefore, the human error probability (HEP) (in this study the HEP is equivalent to mitigation failure probability) is negligible.

(2) Mitigated Scenario for Small Leaks. No fuel damage occurred when the makeup water was injected into the SFP at the time specified by the SFPS. The SFPS suggests that, as long as the spent fuel is covered with water, SFP failure would not occur. Therefore, the available time for operators to respond is longer than the SFPS injection time in some scenarios. For these scenarios, the HEPs were calculated.

(3) Unmitigated Scenario in Late Phases (i.e., OCPs 4 and 5). These scenarios have low decay heat. Even when relying only on natural air circulation, heat convection and radiation, and other natural means of heat transfer, overheating of the spent fuel can be prevented. For these scenarios, the HEPs were not calculated.

(4) Mitigated Moderate Leak Scenarios in OCP2, OCP3, OCP4, and OCP5. When SFP water is drained to the top of fuel rack, the radiation level is considered too high to deploy SFP mitigation strategies on the refuel floor (discussed in section 8.2.3). In OCP2, the SFP water takes almost 6 hours to drain to the top of the fuel rack but only about 2.5 hours in OCP3. The HEPs were calculated for OCPs 2 and 3. The HEPs for OCPs 4 and 5 were not calculated because SFP decay heat is insufficient to cause fuel damage, as noted above.

The SFPS shows that the SFP status is stable in the four scenario classes listed above following termination of the computer simulations at 72 hours after the initiating event. This result implies that, for the unmitigated scenarios, if spent fuel damage does not occur within the first 72 hours, spent fuel damage would not occur afterward because the decrease in decay heat and long time available for response.

Table 40 The SFPS Simulation Results.

	No Leak (90%)	Small Leak (5%)	Moderate Leak (5%)
OCP 1 (0.9%)			**~ 0.05%**
OCP 2 (2.4%)		~0.8%	
OCP 3 (5.0%)			
OCP 4 (25.7%)			
OCP 5 (66%)		~ 99.2%	

- OCP: Operating Cycle Phase
- Percentages above are percent of the time for corresponding condition.

Table 40 provides an overview for performing an HRA as follows:

- The green cells represent that either the HEP is negligible or mitigation does not affect the end consequence. For the SFP no leakage scenario, the SFPS calculated that SFP water would take more than 7 days to boil to the top of the fuel rack. Because of the long time available for mitigation, the HEP is negligible. For the scenario in which the earthquake occurs during OCPs 4 and 5, the SFP fuel decay heat is insufficient to cause a gap release event even without the provision of SFP makeup flow; therefore, mitigation does not affect the end consequence. These two scenario classes (i.e., no leakage and the occurrence of the earthquake during OCPs 4 and 5) are colored as green cells and total about 99.2 percent of the conditional probability. An HRA is not performed for the scenarios in the green cells.

- The OCP 1 moderate leakage scenario (i.e., the red cell with a ~0,05% conditional probability in Table 40) would result in a gap release regardless of whether mitigation has taken place because the current NEI guidance for complying with 10 CFR 50.54(hh)(2) is insufficient (providing at least 500 gpm of injection flow or 200 gpm of spray flow[40]). The flow rates are provided by two flow paths using fire hoses. Significantly increasing the mitigation flow rate requires setting up additional fire hoses to provide additional flow paths. Because the procedures do not provide instructions on when additional flow paths should be established, this study concludes that no additional flow path other than the two procedure-instructed flow paths will be used for SFP mitigation. Therefore, gap release would occur in the OCP1 moderate leak scenarios. This is not because the mitigation flow cannot be deployed in time, but is because the flow rate is insufficient for the assumed OCP 1 decay heat load as determined by SFPS section 6.3.2.[41]

- The yellow colored cells represent conditions where gap release can be prevented if the minimum NEI recommended SFP makeup flow (i.e., 500 gpm of injection or 200 gpm of spray) is deployed in time. This HRA focuses on these scenarios for which mitigation would prevent gap release.

8.1.2 Key Factors Affecting Available Time for Mitigation

The SFPS divides the reference plant operation cycle into five OCPs. OCPs 1 and 2 occur during refueling in which the SFP and reactor cavity are hydraulically connected. Because the reactor cavity and SFP are located within the same reactor building and they are hydraulically connected, a reactor problem would affect the refueling floor work environment in which the effective mitigative actions to prevent SFP fuel damage are performed. OCPs 3, 4, and 5 occur during at-power operations in which the SFP and reactor cavity are hydraulically disconnected.

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[40] NEI 06-12, "B.5.b Phase 2 and 3 Submittal Guideline," issued December 2006 (ADAMS Accession Nos. ML070090060 and ML070080351) recommends minimum of 500 gpm of injection and 200 gpm of spray for implementation of the requirements in 10 CFR 50.54(hh)(2).

[41] In comparison with OCP 1, moderate leakage, and mitigated scenarios, the OCP 2 scenario has the same makeup type (i.e., injection), makeup flow rate, and makeup deployment time. However, gap release did not occur in the OCP 2 scenario because the hottest 88 assemblies for OCP 1 at approximately 4 days have a decay heat of 1,927 kilowatts (kW) or 65 percent of the whole SFP (2,951 kW), whereas the hottest assemblies for OCP 2 at 13 days have a decay heat of 1,143 kW or 32 percent of the whole SFP (3,567 kW).

This HRA assumes different rates of spent fuel decay heat for each OCP, which in turn affects the required mitigation flow and, to some degree, the available time for mitigation necessary to prevent SFP damage.

The SFPS groups the SFP damage caused by the earthquake into three classes: (1) no leakage, (2) small leakage, and (3) moderate leakage with a corresponding conditional probability of 90 percent, 5 percent, and 5 percent, respectively. The small leakage scenario is represented by 40 small tears in the stainless steel liner at the backup bar locations. The small cracks create an initial leakage rate of about 250 gpm. The leakage flow rate depends on the SFP water level. As the SFP water level decreases, the leakage rate reduces. The moderate leakage is represented by a long crack with a combination of the stainless steel SFP liner tear and a through-wall concrete crack at the bottom of the SFP wall. Section 4.1.5 of this report discusses the SFPS damage states in detail. The moderate leak creates an initial leakage rate of about 1,900 gpm.

The HRA assumes that the SFP leak rate affects the available time necessary for mitigation because, when the SFP fuel is not covered by water, the radiation level at the locations in which mitigative equipment is stored and mitigative actions are performed is assumed to be too high for performance of the mitigative actions in this study. Thus, the SFP leak rate directly affects the SFP fuel uncovery time. Table 41 shows the time to SFP fuel uncovery in the various scenarios.

Table 41 Approximate Time of Fuel Uncovery

Time	No Leak	Small Leak	Moderate Leak
OCPs 1 and 2	> 7 days	40 hours	6 hours
OCPs 3, 4, and 5	> 7 days	19 hours	2.5 hours

Figure 98 shows the approximate dose rate contours in the refueling area at the time of defueling when the SFP water level is at the top of the fuel rack. The radiation at the mitigation equipment storage location ranges from 3–30 rem per hour and the radiation level at the locations of the spray nozzles for SFP makeup is in the range of 10 to 300 rem per hour. Working at this radiation level could cause emergency responders who perform mitigation actions to receive doses greater than those in EPA's PAGs (EPA, 1992). This radiation map is the basis for specifying that the SFP makeup must be deployed before the SFP water level reaches the top of the fuel rack in order to credit mitigation success.

In addition to radiation, high temperature on the refueling floor is another factor that affects mitigation success. In this study, 140 °F (60 °C) is used as the temperature threshold. The refueling floor reaches 140 °F before the SFP water level is drained to the top of fuel rack only in the OCP 1 and 2 small leak scenarios. In these scenarios, the reactor head is open. Boiling in the reactor cavity significantly increases the temperature on the refueling floor. Figure 99 shows the time history of the refueling floor temperature of the OCP 1 small leak scenarios. The temperature reaches 140 °F in about 13.5 hours. Figure 100 shows the time history of the refueling floor temperature of the OCP 2 small leak scenarios. The temperature reaches 140 °F in about 26 hours. Because of the long available response time and steep temperature increase at the time of 140 °F reached, changing the temperature threshold to a higher temperature does not affect the HRA results.

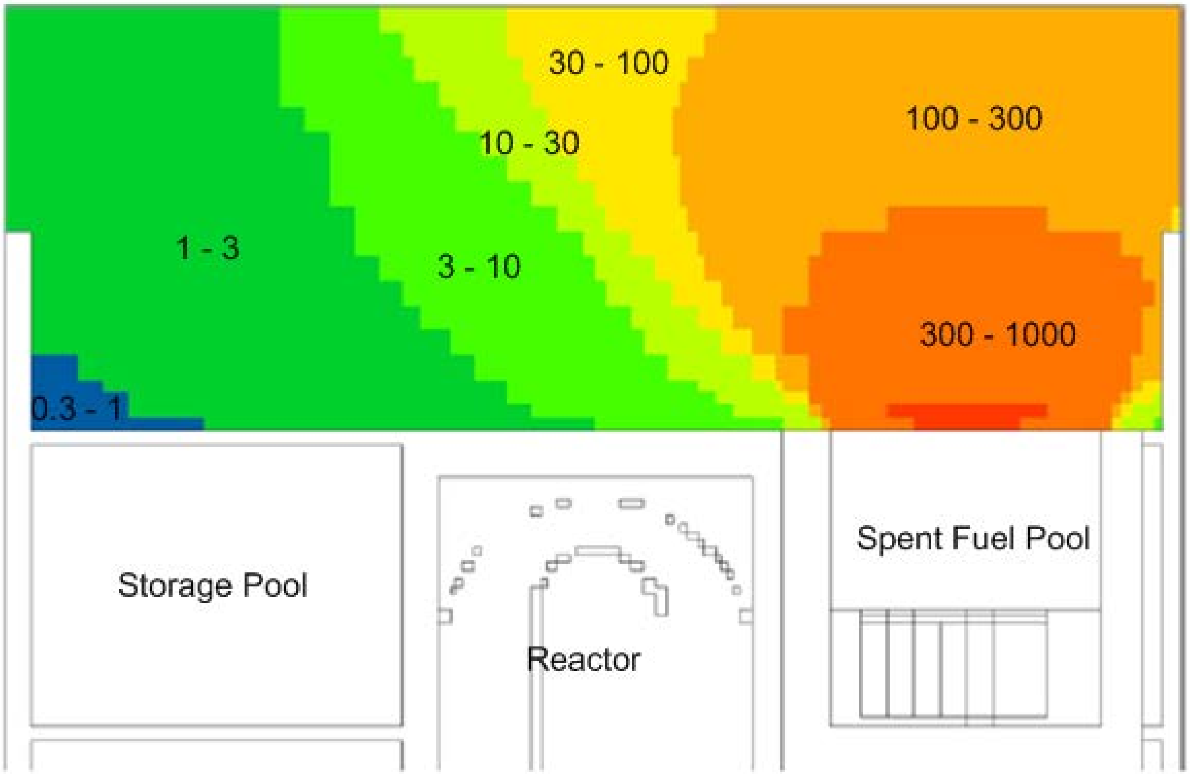


Figure 98 Approximate dose rate of elevation contours, water at the top of fuel hardware, around the time of defueling (rem per hour).

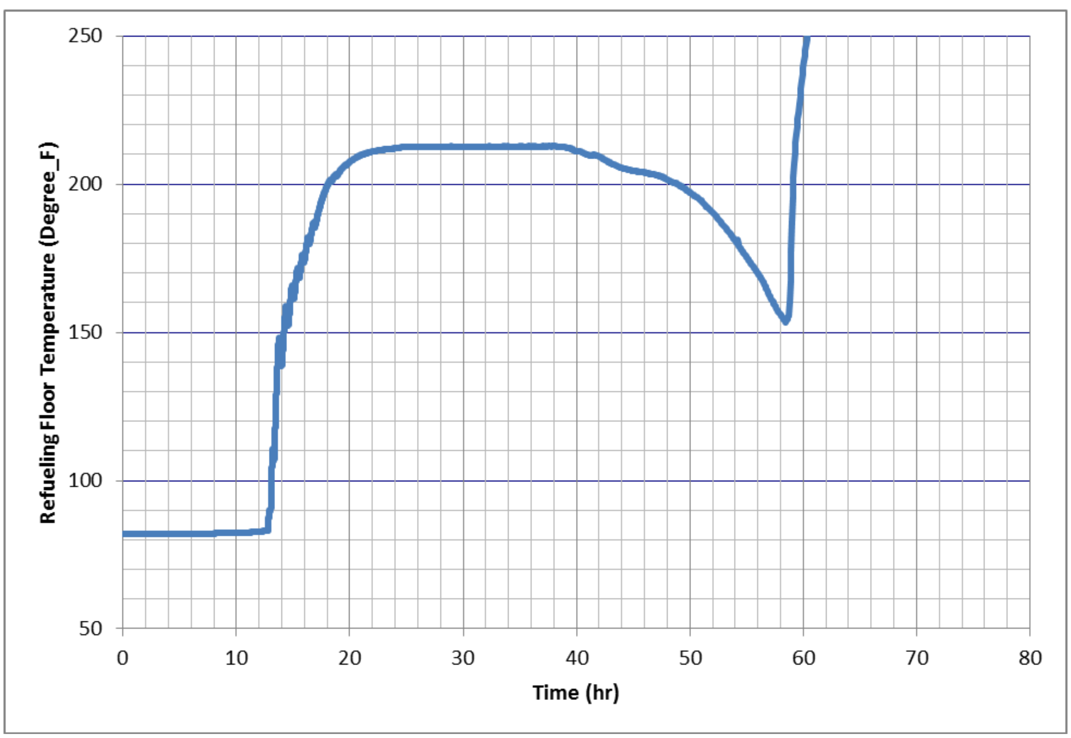


Figure 99 The refueling floor temperature of OCP1 small leak scenarios.

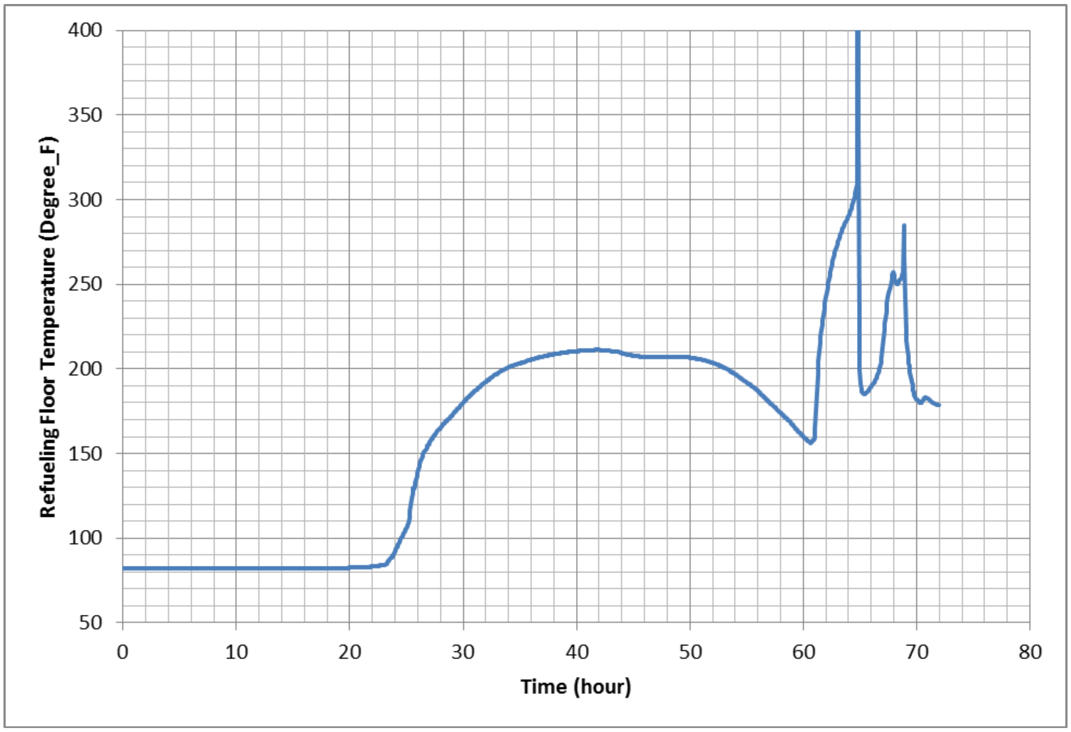


Figure 100 The refueling floor temperature of OCP 2 small leak scenarios.

In summary, successful deployment of the mitigation strategy has to be done before the earliest of either the SFP water reaching the top of the fuel rack or the reactor building atmosphere reaching 140 °F. Table 42 shows these available times for the scenarios of interest.

Table 42 The available time*

	Small Leak (hr)	Moderate Leak (hr)
OCP 1	13.5**	6***
OCP 2	26**	6***
OCP 3	19***	2.5***

*Assume Unit 3 reactor is not damaged
**Due to refueling floor temperature reaching 140°F
***Due to SFP water level draining to the top of fuel rack

8.2 Staffing, Mitigation Equipment, Strategies, and Procedures

8.2.1 Staffing, Procedures, Training, and Response Time

Staffing

This HRA assumes that sufficient plant staff is available for Unit 3 SFP mitigation. In the situation that the hypothetical earthquake causes damage to multiple SSCs, additional events (e.g., fire), and personnel injury, the assumption may not be applicable to some scenarios.

The reference plant uses a combined main control room for its two reactor units. Consistent with NEI 12-01, the on-shift personnel are assumed to be limited to the minimum complement allowed by the site emergency plan. This represents a staffing level during backshift, weekend or holiday. The staffing level of the reference plant, Units 2 and 3, during the backshift, weekend, and holiday includes the following:

- Main Control Room
  - One Shift Manager (Licensed Senior Reactor Operator (SRO)). The shift manager oversees the control room activities and assesses the emergency action level.
  - One Shift Technical Advisor (Licensed SRO). The shift technical advisor performs independent plant status assessment.
  - Two Control Room Supervisors (Licensed SROs). The control room supervisors implement procedures as a team with the reactor operators (ROs).
  - Two Licensed ROs. The ROs perform control board actions according to the control room supervisors' instructions and answer emergency phone calls.
  - Two Assistant (or Spare) Licensed ROs. The assistant operators perform the same functions as the ROs.
- On Site
  - One Field Supervisor (Licensed SRO). The field supervisor oversees onsite activities.
  - Nine Auxiliary Operators. The auxiliary operators will report to the main control room after the earthquake to obtain the master keys for the assigned tasks. Five of the nine auxiliary operators are on the fire brigade.
  - Additional Staff. Additional staffing comprises health physicists, chemical staff, maintenance personnel, and security staff onsite who can support mitigation (e.g., health physicists will provide refueling floor radiation information). However, these personnel are not expected to directly perform SFP mitigative actions.

The above summary describes a typical staffing level during backshift, weekend or holiday of the reference plant instead of the minimum staffing requirement, or during a normal weekday or refueling. If the earthquake occurs during normal working hours or if either Unit 2 or 3 is in a refueling outage, the staffing level would be significantly higher.

To augment staffing, except calling for the off-site plant staff (e.g., to mobilize emergency response facilities), the reference plant can also call for the nearby Delta-Cardiff Volunteer Fire Company to assist in tasks such as SFP mitigation, fire mitigation, and treatment of injured personnel. The fire company could send engines, tankers, a ladder fire truck, an air unit, an ambulance and personnel to the reference plant site. Based upon the above assumptions, this analysis assumes that there is sufficient staff for Unit 3 SFP mitigation. No detailed analysis is performed on the staffing situation for all scenarios.

<u>Procedures and Operator Initial Responses</u>

In the hypothetical earthquake that causes a station blackout (SBO), the general response is that the control room supervisors work with the ROs to implement the emergency response procedures. In this case, the entry conditions of the following three procedures are met:

(1) SE-11, “Loss of Offsite Power”
(2) SE-5, “Earthquake”
(3) TSG-4.1, “Operational Contingency Guideline”

The control room supervisors work with ROs to implement the above three procedures in parallel. The immediate objectives are to ensure that the reactor is properly tripped and ensure sufficient electricity, equipment, and water to maintain reactor cooling. Because a high-priority task in an SBO scenario is the provision of emergency electric power, the control room supervisors would send two auxiliary operators to inspect the emergency diesel generators and would direct one assistant (or auxiliary) operator to implement SE-11 to connect the dedicated power supply from the Conowingo Hydroelectric Generating Station (Conowingo) to the reference plant. If the earthquake has not affected Conowingo, connecting its supply power to the reference plant would take about 1 hour during normal conditions. The other auxiliary operators will be tasked with performing a plant walkdown and SFP inspection in accordance with SE-5.

<u>Training</u>

Training related to the implementation of TSG-4.1 and 10 CFR 50.54(hh)(2) includes the following:

- annual training in emergency response organization mobilization and implementation of the TSG-4.1 and TSG-4.2, “Extreme Damage Mitigation Guidelines for Loss of Large Area of the Plant,” procedures and the related requirements in 10 CFR 50.54(hh)(2)

- biannual training on security threat responses

- initial training on procedures and equipment related to 10 CFR 50.54(hh)(2)

<u>Response Time</u>

NEI 06-12, Revision 2, “B.5.b Phase 2 & 3 Submittal Guidance,” states that plants should be able to deploy a flexible means of providing SFP makeup (i.e., either 500 gpm of injection or 200 gpm of spray per unit) within 2 hours from the time in which plant personnel diagnose that external SFP makeup is required. This HRA study uses the 2-hour deployment time as the action time for deploying mitigation. The total mitigation time is the sum of delay time, diagnosis time, and action time (discussed in Section 8.3.2.2).

The analysis in Volume 1 of NUREG/CR-7110, “State-of-the-Art Reactor Consequence Analyses Project,” estimates that, given the hypothetical earthquake event that causes SBO but with dc power, the technical support center (TSC) is assumed operational within 2.25 hours after the earthquake. The NEI 12-01 guideline assumes the following site accessibility: (1) no site access within the first 6 hours; (2) limited site access between 6 to 24 hours; and (3) improved site access after 24 hours. The assumptions apply to a large-scale external event that occurs that results in: (1) all on-site units affected; (2) extended loss of AC power, and (3) impeded access to the reactor buildings. The emergency response facilities most relevant to responding to the hypothetical earthquake are the operational support center (OSC), which is an onsite assembly area separate from the control room, and the TSC to which licensee operations support personnel report in an emergency. NUREG/CR-7110 does not provide an estimated time in which the OSC will be operational. Therefore, for the purposes of this study the TSC assumption of 2.25-hours is also used for the OSC when neither Unit 2 nor Unit 3 is in a refueling outage. The OSC provides additional man power to mitigate plant damage, but this additional staff is not considered in this HRA study.

8.2.2 Mitigation Equipment

This HRA study assumes that portable mitigation equipment is available but the installed equipment is not available for Unit 3 SFP mitigation. The portable equipment includes the two portable diesel pumps discussed in this section. The installed equipment includes the fire system and residual heat removal system. If the earthquake causes damage to multiple reactors and SFPs that consequently requires mitigation equipment, there may not be sufficient portable equipment for the Unit 3 SFP mitigation. For the purposes of this study, portable mitigation equipment was assumed to be available.

The reference plant relies on the following onsite equipment and systems for SFP makeup:

- <u>Fire System</u>: One motor-driven fire pump and one diesel-driven fire pump are necessary to pressurize the fire header. The diesel-driven fire pump is designed to operate for 6.4 hours at full load. Making up the diesel fuel requires the use of a temporary 120-volt ac power source to restore a fuel oil transfer pump to deliver fuel for the diesel-driven fire pump. In situations in which both fire pumps are lost and cannot be repaired within 1 hour, the reference plant will contact the York County 911 center for a fire engine to pressurize the onsite fire header. If the reference plant cannot obtain offsite support and if the situation allows, the reference plant can use one of the two portable diesel pumps to pressurize the fire header.

  The two fire pumps are housed in a seismic Class I tornado-resistant structure. Therefore, the diesel-driven fire pump is assumed functional after the earthquake. However, the underground fire pipes may be damaged by the earthquake. Depending on the damage, the fire system may still be available by isolating the damaged section or sections or by using a fire hose in place of the fire main. The fire system is the preferred water source for the most effective mitigation strategies necessary to prevent spent fuel gap release. If the fire system is not available, the Conowingo pond or torus storage tank is the alternative water source.

- Diesel-Driven Portable (DDP) Pump. The diesel-driven portable pump has the capability of delivering 600 gpm of water. A trailer stationed near the pump stores all piping, connectors, and spray nozzle. A dedicated pickup truck will be used to tow the pump and trailer to the specified location for operation. With a full tank, which is the normal condition, the pump can continue to run for more than 12 hours. The DDP pump has a 4" discharge connection. To deliver the flow rate of 500 gpm of injection or 250 gpm of spray the plant staff uses a wye adapter to connect the 4" discharge to two 2.5" hoses. The reference plant demonstrated that the combination flow rate met the 500 gpm of injection and 200 gpm of spray requirements.

- Diesel-Driven High-Capacity Portable (DDHCP) Pump. The diesel-driven high-capacity portable pump has the capability of delivering 1,300 gpm of water. The DDHCP pump has two 4" discharge connections. To deliver the NEI recommended flow rate, the plant staff uses a wye adapter to connect a 4" discharge to two 2.5" hoses. Four 2.5" discharging hoses would be needed to reach the pump maximum discharge capacity. The TSG-4.1 instructs the plant staff to connect two 2.5" hoses to a 4" discharge connection for SFP makeup. The reference plant demonstrated that the combination flow rate of using two hoses exceeds the 500 gpm of injection and 200 gpm of spray requirements. Like the diesel-driven portable pump, dedicated pickup trucks will be used to tow the portable pump to the designated location for operation.

The HRA team identified the three systems listed above during a site visit to the reference plant in July 2012. The HRA team was aware that the referece plant planned to purchase more equipment to address Order EA-12-49 mitigating strategies; however, this HRA study does not credit the additional equipment.

The reference plant stores much of its mitigation equipment at grade level. Section 2.4.3.5 of reference plant's' FSAR discusses the effect of a simultaneous failure of the upstream Holtwood dam on the site. The FSAR indicates that the upstream Holtwood dam failure would not increase the level of the Conowingo pond such that it would exceed the grade level at the site. Therefore, a simultaneous Holtwood dam failure is not assumed to affect the availability of mitigation equipment.

8.2.3 Mitigation Strategies

NEI 06-12 discusses implementation strategies for SFP makeup and spray. The mitigation strategy described in NEI-06-12 provides that it should be implemented within two hours after the decision of deploying the mitigation strategy. This NEI guidance defines the actions that should be taken in situations in which normal procedures or command and control structures are not available. The notes in the parentheses below include items not considered applicable to the accident scenarios for the SFPS. The assumptions in the guidance include the NEI-06-12:

- An immediate threat warning does not occur.

- Access to the control room is lost (not expected in SFPS scenarios).

- Equipment or supplies normally located in the control room or in the building that houses the control room are lost (not expected in SFPS scenarios).

- Access to the building that contains the control room is lost (not expected in SFPS scenarios).

- All personnel normally in the control room are lost (not expected in SFPS scenarios).

- All ac and dc power required for operation of plant systems is lost (i.e., both class 1E and non-class 1E sources).

- Only minimum site staffing levels are available (i.e., weekend/backshift). Note: the minimum staff mentioned in the NEI guidance is not the minimum staff requirements. Instead, it refers to the normal staffing level during weekend or backshift. This assumption does not apply when either Unit 2 or Unit 3 is in refueling outage.

- Other onsite control rooms and personnel in separated building are unaffected. (Personnel injury is likely to occur given the hypothetical earthquake.)

- Operations personnel who are not normally located in the control building are available for implementation of extensive damage mitigation guidelines.

- Nonlicensed personnel, typically an auxiliary operator, can perform the mitigative actions.

- The level of training on implementing procedures and guidance is consistent with actions under severe accident management guidelines and is consistent with utility commitments made under B.5.b Phase 1.

- Before the event, the plant systems are in a normal configuration with the reactor at 100-percent power. (This SFP safety analysis includes refueling outages (i.e., OCPs 1 and 2).)

The above items that are noted in parentheses as not expected in the SFPS scenarios apply to TSG-4.2. TSG-4.2 may not apply to the SFPS scenarios. Instead, TSG-4.1 is the most applicable procedure for the SFPS scenarios. The sections below discuss the SFP mitigation strategies in accordance with TSG-4.1.

<u>Internal Makeup</u>

This strategy connects two fire hoses to the two existing fire system standpipes on the refueling floor to provide a minimum of 500-gpm total injection flow to the SFP. The fire system must be pressurized to implement this strategy. To implement this strategy, the operators need to remove the existing 1.5-in reducer from the two fire standpipes, connect two 2.5-in fire hoses to the two standpipes, and route the two fire hoses to the SFP. Operators can deliver makeup flow by fastening the hose to the SFP side for direct injection into the SFP. Operators can also deliver makeup flow by connecting the two fire hoses to the two spray nozzles to spray water into the SFP. This strategy will deliver a total spray flow of more than 200 gpm. All equipment mentioned is available on the refueling floor. This strategy assumes that the refueling floor is accessible for local makeup.

External Makeup and Spray

This strategy uses any of the two portable diesel pumps (Section 8.2.2) to inject or spray water into the SFP. This strategy requires the plant staff to (1) tow the portable diesel pump to the desired location at grade level, (2) lay two approximately 200-ft fire hoses that are connected by two sections from the refueling floor through a stairwell to the grade level (about 100 ft in elevation difference) to connect the hoses to the charging output of the portable pump, (3) connect the hose end on the refueling floor to an spray nozzle, and (4) connect the portable pump's suction to a fire hydrant. The hoses for connecting the pump discharge to the spray nozzles are stored on the refueling floor. Each spray nozzle can be adjusted for spray or to obtain full flow (i.e., injection).

The external makeup and spray mitigation strategy uses the fire water system as the default water source to the portable diesel pumps. Under situations in which the fire water system is not available, the reference plant's procedure SO-37L.1.a, "Diesel Driven Portable and Diesel Driven High Capacity Portable Pump Startup and Shutdown," identifies additional water sources, including the inner pond, discharge pond, and Conowingo pond. In the event of a seismically induced failure of the Conowingo pond, the loss-of-pond procedure provides cooling water management strategies. In addition, the reference plant Assignment Report No. 01001590 identifies the candidate water sources, including high-pressure service water, fire water, residual heat removal water, condensate transfer water, and cross connections to the opposing unit's spent fuel water supply. Detailed step-by-step instructions for using water from the alternative water sources are not available. However, this HRA study credits the use of the alternative water sources in the situations when the fire water is not available because of the similarity of using water from these sources to drafting fire water.

External Local Spray or Scrub

This strategy uses any of the following three procedures, individually or in combination, to provide spent fuel cooling or secondary containment spray to scrub potential radionuclides released from the SFP primary or secondary containment structures:

(1) Use the portable diesel pump to provide water to the two spray nozzles on the refueling floor to spray water into the SFP. This strategy requires operators to lay out fire hoses from the refueling floor to grade level, as described in the section above entitled, "External Makeup and Spray."

(2) Use the portable diesel pump to provide water to one or two of the two spray nozzles located on the turbine building roof to spray water to the secondary containment or the refueling floor through building breaches. This strategy requires operators to lay out hoses from the turbine building roof to grade level to connect the portable pump and the spray nozzle.

(3) Use a ladder truck to spray water into the SFP area through building breaches from the steel structure surrounding the SFP floor. This strategy requires the use of an offsite fire company's 100-ft ladder fire truck. Exelon Generation Company, LLC (owner of the reference plant), has a letter of agreement with nearby Delta-Cardiff Volunteer Fire Company. Upon dispatch and without additional complexity, the fire truck could arrive at the reference plant within 30 minutes (Assignment Report No. 01001590). The Delta-Cardiff Volunteer Fire Company possesses two fire trucks from Seagrave Fire

Apparatus, LLC, either one of which can perform the portable diesel pump function. Other nearby fire companies could support the reference plant mitigative efforts.

Items (2) and (3) above are useful primarily to mitigate the release of radioactivity off-site, but are not effective in preventing radioactive release from the SFP fuel rods. Because the mitigation success of this HRA study is to prevent radioactive release from the SFP fuel rods, items (2) and (3) are not credited in this HRA study. TSG-4.1 requires the plant to use the external local spray or scrub strategy after it has attempted the internal makeup and external makeup and spray strategies (as discussed earlier). For consistency with the NEI guidance, the reference plant uses the flowchart shown in Figure 101 as general guidance for deployment of SFP mitigation strategies.

Makeup with Residual Heat Removal Pump from the Torus

This strategy requires that electrical power is available for a residual heat removal pump to pump torus water into the SFP at a flow rate of 10,000 gpm. This flow rate is much larger than the maximum moderate leakage flow rate (i.e., approximately 1,900 gpm). In an SBO scenario, this strategy is not available because power is not available for the residual heat removal pumps. this HRA study assumes this strategy is not available.

Leakage Control

The reference plant has a list of stocked materials that could help to reduce the leakage flow rate, including steel plates, plywood, bag stopper, sealants, ropes, and rubber matting. Certain materials would require a crane for moving (e.g., 5/8-in by 4-ft by 4-ft steel plates. Based on its emphasis on the initiation of makeup strategies and the 72-hr scope of the analysis, the study did not consider repair options.

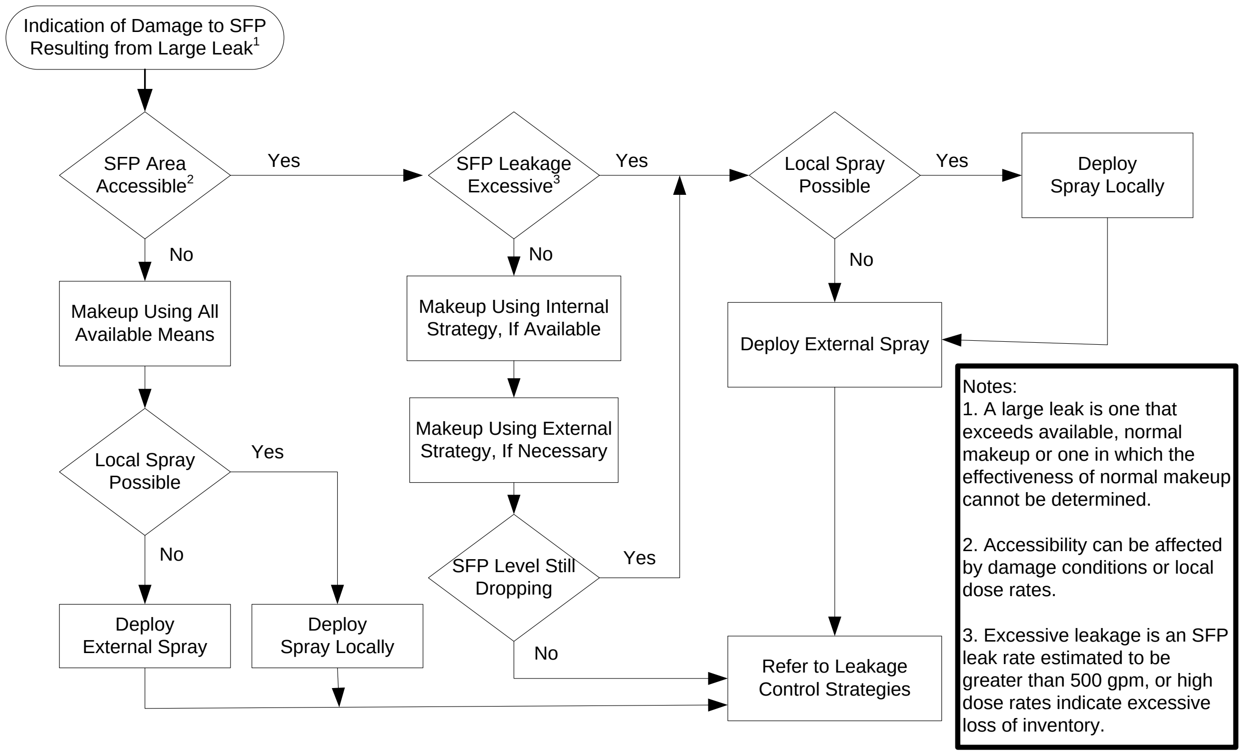


Figure 101 Generalized guidance for SFP makeup and spray decisions

8.3 Study Framework, Scope, and Approach

8.3.1 Study Framework and Scope

Preventing gap release of the Unit 3 SFP fuel is the success criterion defined in this HRA study. Because human performance is sensitive to the extent of earthquake damage to the plant, the study identifies a set of plant damage states and estimates an HEP for each damage state. Identification of the damage state is based on the status of a few key SSCs that include electric power availability, reactor status, and fire system availability. Figure 102 illustrates the framework and scope of this HRA study. The large rectangle with dashed lines shown in Figure 102 represents the scope of this HRA study. Each box within the dashed lines represents a probability. The partial enclosure of the damage state signifies that the scope of this HRA study only identifies the plant damage states; it does not estimate the probabilities of the damage states. Therefore, the gap release (and no gap release) probabilities could not be estimated, and the HEPs were computed to only provide initial general insights.

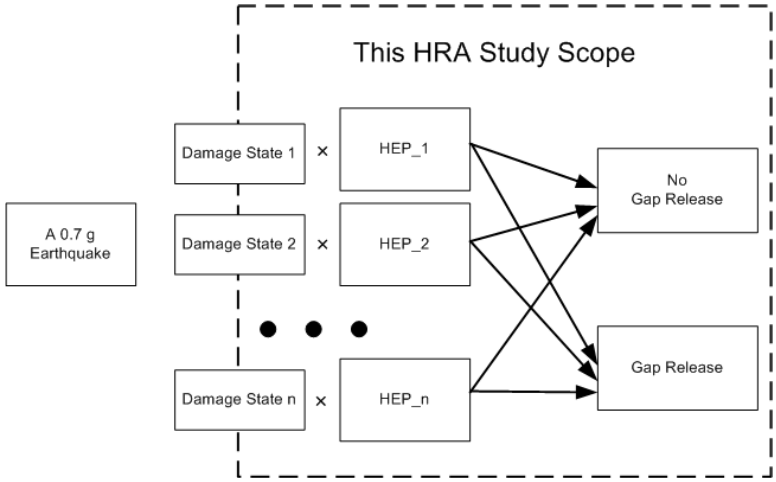


Figure 102 The study framework

8.3.2 Approach to Human Error Probability Estimates

8.3.2.1 A Two-Phased Approach to HEP Estimates

This HRA study used a two-phase approach to estimate mitigation failure probabilities or human error probabilities (HEPs). Phase 1 estimates HEPs for mitigating the reference plant SFP leak. These estimates consider the status of the electric power, reactor and primary containment, and fire system of Unit 3, the Unit 3 OCP, and Unit 3 SFP leakage rate. These are the dominant factors that affect mitigation of a single SFP leak. In Phase 2, other damage to the site that affects mitigation is discussed. Phase 2 involves situations that combine reactor and SFP problems or multiple unit problems caused by the same earthquake. The four discrete steps listed below represent the HEP estimation process used in this analysis. The four sections that follow discuss these steps in detail.

(1) Identify the time required for deployment of SFP makeup.
(2) Identify the damage states and corresponding available time.
(3) Estimate the HEP of each damage state.
(4) Identify additional feasibility considerations.

8.3.2.2 Step 1—Identify the Time Required

In this study, the total time necessary for deployment of the SFP makeup is the sum of the following three time segments:

(1) Delay Time: In an earthquake-induced SBO scenario, the control room operator's primary focus is on reactor safety. Although the SFP trouble alarm is triggered soon after the earthquake, a time delay occurs for starting a diagnosis process to investigate an SFP problem. The cue for starting to investigate the SFP is the earthquake procedure SE-5. Step 9 of the procedure instructs the operators to check the SFP, SFP cooling system, and fuel floor blowout panels. Based on an interview with PBAPS staff, the delay time ranges between 30 minutes and 1 hour. This study uses 45 minutes for the SBO scenarios, 30 minutes for LOOP scenarios, and 60 minutes for SBO without dc power scenarios because, based on crew interviews, the control room supervisor would, at a minimum, simultaneously implement SE-11 and SE-5. When less electricity is

available for maintaining reactor safety margin, the operators would put more effort into restoring electricity (i.e., SE-11). As a result, less time is spent on SE-5, which consequently would delay implementation of Step 9 in SE-5 to send auxiliary operators to check the SFP status.

(2) Diagnosis Time: Diagnosis time is the time between when auxiliary operators are deployed to inspect the SFP and when they report SFP leakage back to the control room operators. Based on the leakage rate (both small leakage and moderate leakage) and leakage locations (i.e., the SFP bottom at the elevation of a few inches above the 195-ft floor), detecting SFP leakage is not a challenging task. Based on the same interview with PBAPS staff and a plant walkdown of the path that the auxiliary operators would normally take to inspect the SFP, the diagnosis time was determined to be 15 minutes.

(3) Action Time: The 2-hr implementation expectation in NEI 06-12 is used for deployment of the portable diesel pump to provide SFP makeup. The HRA uses the 2 hours as the action time at which the fire system is available because TSG-4.1 instructs the staff on how to use the fire system as the water source. When the fire system is not available, using water from the alternative water sources would require additional time. An additional 1 hour of action time is necessary when the fire system is not available.

Table 43 and Table 44 summarize the time estimates based on the above discussion. Table 43 shows the mitigation time estimates in the scenarios for which fire water is sufficient for mitigation. Table 44 adds 1 hour of action time to Table 43 to account for the effect of unavailable or insufficient fire water. In Table 43 and Table 44, note that the total time difference between the LOOP scenarios and SBO without dc scenarios is only 30 minutes. However, the conditional reactor core damage probability in these two scenarios would be significantly different. Further, the conditional reactor core damage probability in these two scenarios would have a significant difference. That difference directly affects the refueling floor accessibility and, in turn, the mitigation success probability. This HRA study assesses HEPs for the SBO and SBO without dc scenarios with and without reactor core damage separately.

Table 43 Estimates of the Time Required for the Operator to Deploy SFP Makeup If Fire Water Is Available

	Delay Time	Diagnosis Time	Action Time	Total Time Required
LOOP	30 minutes	15 minutes	2 hours	2 hours 45 minutes
SBO	45 minutes	15 minutes	2 hours	3 hours
SBO without dc	60 minutes	15 minutes	2 hours	3 hours 15 minutes

Table 44 Estimates of the Time Required for the Operator to Deploy SFP Makeup If Fire Water Is Not Available or If It Cannot Deliver Sufficient Flow

	Delay Time	Diagnosis Time	Action Time	Total Time Required
LOOP	30 minutes	15 minutes	3 hours	3 hours 45 minutes
SBO	45 minutes	15 minutes	3 hours	4 hours
SBO without dc	60 minutes	15 minutes	3 hours	4 hours 15 minutes

8.3.2.3 Step 2—Identify the Damage States and Available Time

The key factors that affect the likelihood of successful mitigation of the Unit 3 SFP include SFP leakage size; OCP; and the status of the electric power, reactor and primary containment, and fire system of Unit 3. These factors characterize the damage states (as shown in Table 47). SFP leakage size and whether the SFP and the reactor cavity are hydraulically connected (i.e., during refueling and nonrefueling) largely determine available time. As discussed earlier, the available time is determined by the shorter time of either the SFP water reaching the top of the fuel rack or the refueling floor reaching 140°F. Table 45 shows the time required and time available of the damage states of interest assuming the Unit 3 reactor is not damaged.

Table 45 Estimates of time required and time available for mitigation

		Small Leak		Moderate Leak	
		Time Required(hr)	Time Available(hr)	Time Required(hr)	Time Available(hr)
OCP 1	LOOP	2.75(3.75)	13.5	2.75(3.75)	6
	SBO	3.0(4.0)		3.0(4.0)	
	SBO w/o DC	3.25(4.25)		3.25(4.25)	
OCP 2	LOOP	2.75(3.75)	26	2.75(3.75)	6
	SBO	3.0(4.0)		3.0(4.0)	
	SBO w/o DC	3.25(4.25)		3.25(4.25)	
OCP 3	LOOP	2.75(3.75)	19	2.75(3.75)	2.5
	SBO	3.0(4.0)		3.0(4.0)	
	SBO w/o DC	3.25(4.25)		3.25(4.25)	

*The numbers outside the parentheses are the time required when the fire system is available. The numbers inside the parentheses are the time required when the fire system is not available.

**These values assume that the Unit 3 reactor is not damaged and the staff uses the portable diesel driven pumps for SFP mitigation

8.3.2.4 Step 3—Estimate Basic HEPs of a Single Unit Event

This step estimates the basic HEPs for each damage state based on the following assumptions and practices:

- The required mitigative equipment stored outside of the reference plant, Unit 3, and water sources are available. Step 4 considers equipment and water unavailability and other factors.

- The plant staff is available for performing the mitigation activities.

- The earthquake damaged much of the nonsafety piping and equipment but the workplace is accessible (with additional difficulties from normal situations).

- The purpose of including some situations in the HRA (e.g., core damage within the specified available time) is to explicitly identify the key factors that affect human performance. Estimating the likelihood of the occurrence of these situations is outside the scope of this HRA study. Estimating the likelihood of each situation would require a PRA.

The main considerations necessary for assessing HEPs are based on the time margin and workload that affect staffing availability. Electric power availability strongly affects workload. The power availability is classified into: (1) LOOP only, (2) SBO, and (3) SBO without dc. The three classes of power availability impose significant differences in operator workload that, in turn, affect personnel availability to perform all required tasks. The flow diagram in Figure 103 shows the HEP estimation procedure, which is based on NUREG-6883 “The SPAR-H Human Reliability Analysis Method” issued in 2005, supplemented with the NRC staff’s expert judgment.

SPAR-H’s low power and shut down diagnosis worksheets classify time margin effects into five classes as shown in Table 46.

Table 46 Time margin effects on human error probability in the SPAR-H HRA method for cognitive activities in low power /shutdown operations

Class	HEP or HEP Multiplier	Note
Insufficient time	HEP = 1.0	Less than 2/3 of normally time required
Barely adequate time	HEP multiplier = 10	~2/3 of normally time required
Nominal time	HEP multiplier = 1	About the normally time required
Extra time	HEP multiplier = 0.1	Equal to or greater than 5 times of normal time required
Expansive time	HEP multiplier = 0.01	Equal to or greater than 50 times of normal time required

The SPAR-H’s action worksheets use slightly different time scales to adjust the HEP. The adjusting factor 1 in Figure 103 represent time margin effects on HEP based on SPAR-H’s classification. The adjusting factor 2 in Figure 103 represents the performance shaping factors of “complexity” and “ergonomics/human machine interface” of SPAR-H. Table 47 shows the HEP calculation results. Note that OCP1 moderate leak scenarios are likely to have gap release because the NEI recommended minimum mitigation flow rates are insufficient to prevent gap release. This is not reflected in Table 47 because this is not considered as a human error in a typical HRA application.

Table 47 shows that fire system availability in general does not have significant effects on human error probability. Table 48 summarizes the qualitative results of the HRA with respect to the likelihood of gap release in various plant states with the assumption of no reactor core damage. Two plant states have an HEP of 1.0: moderate leak in OCP1 and moderate leak in OCP3. In OCP1 moderate leak scenarios, the high likelihood is because the NEI recommended minimum mitigation flow rates are insufficient to prevent gap release. The high likelihood is not shown in Table 47 because the failure is considered as a design issue rather than a human error from a conventional HRA perspective. In OCP3 moderate leak scenarios, the high likelihood is because of the short time available for response (i.e., about 2.5 hours).

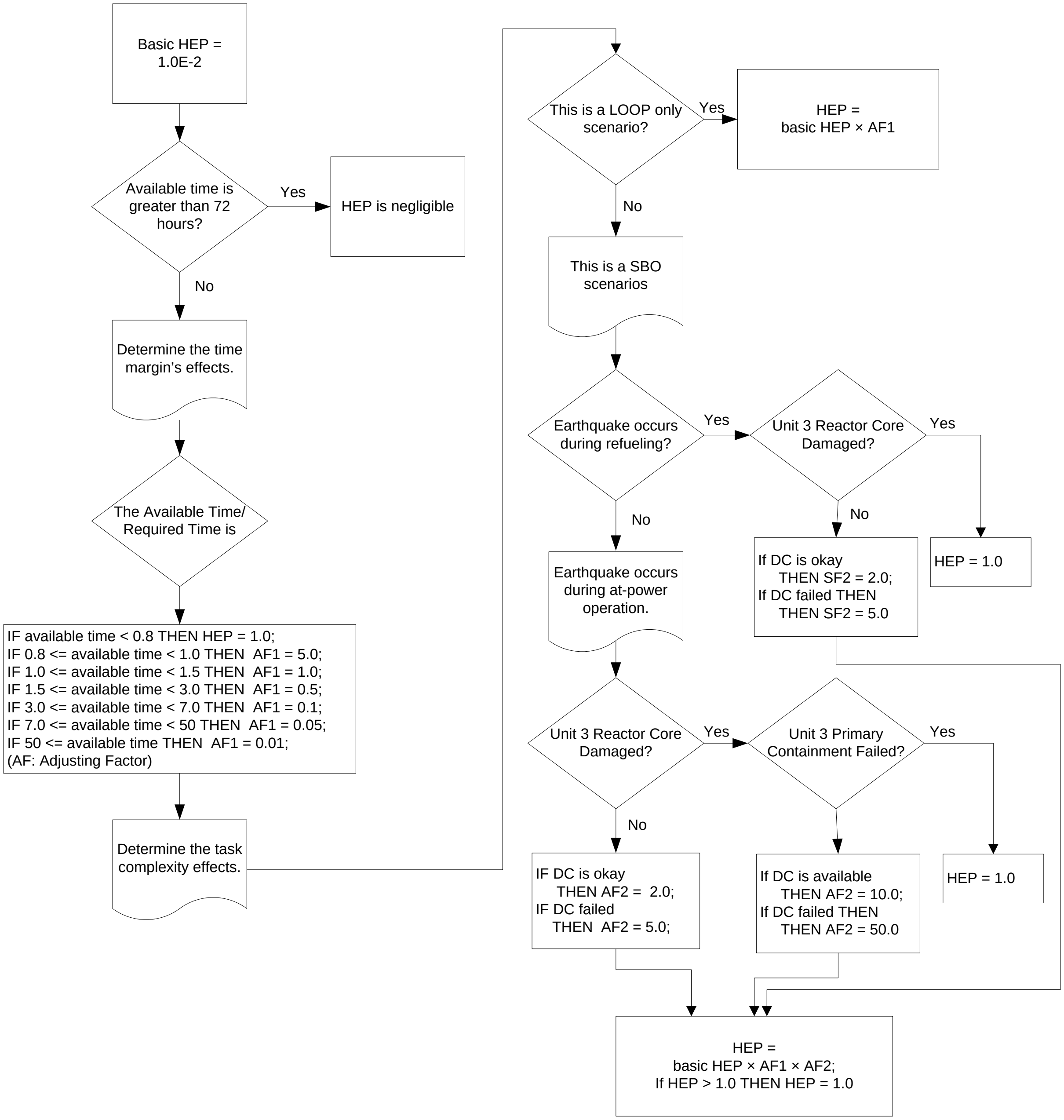


Figure 103 Flow chart for estimating HEPs for a single reactor unit event

Table 47 Human error probability estimates of a single unit event

			Small Leak	Moderate Leak
OCP 1	LOOP*		0.001 (0.001)	0.003 (0.003)
	SBO	No CD	0.006 (0.006)	0.002 (0.002)
		CD	1 (1)	1 (1)
	SBO w/o DC	No CD	0.015 (0.015)	0.005 (0.005)
		CD	1 (1)	1 (1)
OCP 2	LOOP*	No CD	0.003 (0.003)	0.0005 (0.001)
	SBO	No CD	0.006 (0.006)	0.001 (0.002)
		CD	1 (1)	1 (1)
	SBO w/o DC	No CD	0.015 (0.015)	0.0025 (0.005)
		CD	1 (1)	1 (1)
OCP 3	LOOP*	No CD	0.01 (0.1)	0.001 (0.001)
	SBO	No CD	0.2 (1)	0.002 (0.002)
		CD; CTM intact	1 (1)	0.05 (0.05)
		CD, CTM breach	1 (1)	1 (1)
	SBO w/o DC	No CD	0.5 (1)	0.005 (0.005)
		CD; CTM intact	1 (1)	0.05 (0.05)
		CD, CTM breach	1 (1)	1 (1)

*Assume no reactor core damage (CD)

**The numbers outside the parentheses are the HEPs when the fire system is available. The numbers inside the parentheses are the HEPs when the fire system is not available

Table 48 The likelihood of gap release*

	Small Leak (5%)**	Moderate Leak (5%)**
OCP 1 (0.9%)**	Low[1]	High[2]
OCP 2 (2.4%)**	Low[1]	Low[3]
OCP 3 (5.0%)**	Low[1]	High[4]

*Assumes only one SFP damaged without concurrent reactor core damage
**The probabilities are conditional probabilities given that the studied earthquake occurs
[1]The available time for response is long. The SFP fuel is submerged if SFP makeup is deployed in time.
[2]The NEI recommended minimum mitigation flow rate is insufficient to prevent gap release.
[3]The NEI recommended minimum mitigation flow rate is sufficient to prevent gap release.
[4]The available time for response is short so that the SFP makeup will likely not be deployed in time to prevent gap release.

8.3.2.5 Step 4—Additional Feasibility Considerations

This final step (i.e., Step 4) identifies situations that occur outside of the reference plant, Unit 3, that would have adverse effects on Unit 3 SFP mitigation. These effects are not considered in Step 3. These additional considerations include the following:

- Equipment demand cannot be met: When the earthquake causes extensive damage to the reactors and SFPs of Unit 2 and Unit 3 and the normal reactor and SFP cool down mechanisms are not available, the two portable pumps may not be available for Unit 3 SFP makeup given the multiple demands. The DDHCP pump has two 4” discharge connections, and the DDP pump has one 4” discharge connection. In combination, the two portable diesel pumps can deliver three times the NEI recommended minimum mitigation flow rate. The operators have to decide how to use the limited equipment for multiple problems for the reference plant’s two reactors and two SFPs. The decision will strongly depend on the situation.

- Damage to the mitigation equipment (e.g., the DDHCP pump and DDP pump) and support equipment (e.g., pump accessories and the designated truck to tow the pumps) would reduce the available equipment or delay mitigation.

- Simultaneous large or multiple fire events that demand more plant staff personnel than those available.

- Structural damage causes plant personnel injury that could result in less than adequate personnel available for SFP mitigation.

- Unit 3 Refueling floor is inaccessible for reasons such as Unit 3 reactor damage causing high radiation in the access path.

The cells in Table 48 with low gap release likelihood can be split into two groups:

- Greater than 13 hours for all small leak scenarios

- About 6 hours for OCP 2 moderate leak scenarios

In either situation, sufficiency of plant response personnel is likely not an issue because of the long available time of the small leakage scenarios and in refueling outage of the OCP 2 scenarios. Though not accomplished through a full scope PRA, this HRA attempted to account for the complexity of handling multiple reactors and SFP damage events. As such, an adjusting factor of 50 (based on the SPAR-H's performance shaping factors of "high complexity" and "low experience/training") was applied to Table 47. The results are summarized in Figure 104 and Table 49.

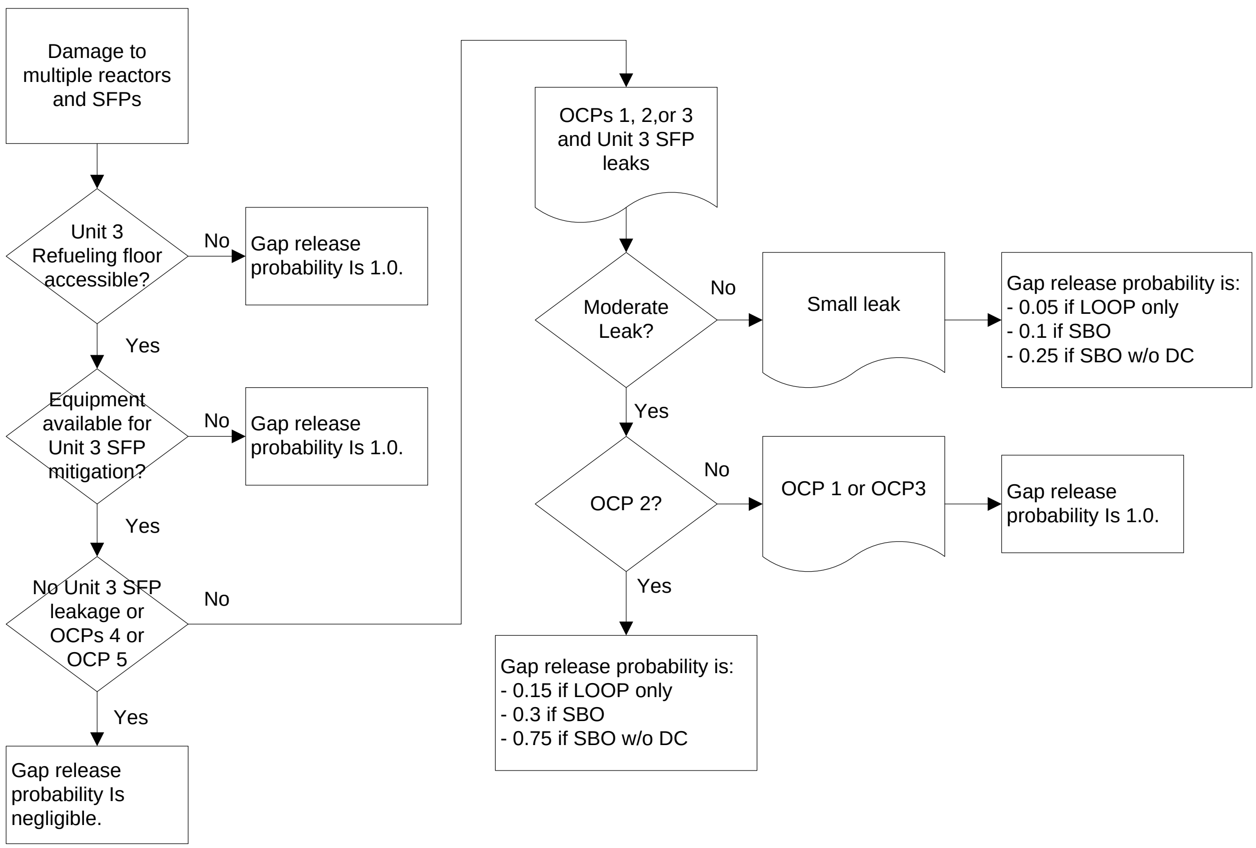


Figure 104 The gap release probability assessments given damage to multiple reactors and SFPs.

Table 49 shows three levels of likelihood of having radioactive release from the Unit 3 SFP fuel rods. Three colored coded regions are discussed below:

- Green Cells
  Two sub groups in the green coded cells: (1) the "no leak" scenarios have long available time (greater than 7 days) for response. The mitigation failure probability is determined to be negligible; and (2) The OCP4 and OCP5 have low decay heat. Even without mitigation, radioactive release is not expected.
- Yellow Cells:
  For the small leak scenarios, the available time ranges from more than 13 hours to more than 1 day. Give the long time available, time is not a critical factor affecting mitigation success. The mitigation failure probability is estimated to range from one failure out of twenty to one failure out of four. For the OCP2 moderate leak scenarios, the time

available is 6 hours. This increases the mitigation failure probability compared to the small leak scenarios. The mitigation failure probabilities for OCP 2 moderate leak scenario range from one failure out of twenty to three failures out of four.

- Red Cells:
  Two red cells are in Table 49. The OCP1 moderate leak scenario is red because the 500 gpm of injection or 200 gpm spray is insufficient to prevent fuel overheating. The OCP3 moderate leak scenario has only a short time available (2.5 hours), and mitigation is not expected to be deployed in time.

Table 49 Scenario Specific Human Error Probability Estimates*.

<table>
<tr><th></th><th>No Leak
(90%)</th><th>Small Leak
(5%)</th><th>Moderate Leak
(5%)</th></tr>
<tr><td>OCP 1 (0.9%)</td><td rowspan="4">Negligible</td><td rowspan="3">- 0.05 if LOOP only
- 0.1 if SBO
- 0.25 if SBO w/o DC</td><td>1.0**</td></tr>
<tr><td>OCP 2 (2.4%)</td><td>- 0.15 if LOOP only
- 0.3 if SBO
- 0.75 if SBO w/o DC</td></tr>
<tr><td>OCP 3 (5.0%)</td><td>1.0***</td></tr>
<tr><td>OCP 4 and
OCP 5 (91.7%)</td><td colspan="2">Inconsequential</td></tr>
</table>

- OCP: Operating Cycle Phase
- Percentages above are the percent of the time for the corresponding condition.
* Assume mitigating equipment is available for Unit 3 SFP, and Unit 3 reactor status does not deny access to the Unit 3 refueling floor.
**The NEI recommended minimum mitigation flow rate is not sufficient to prevent gap release. The procedure (i.e., TSG-4.1) does not instruct operators to establish an additional SFP makeup flow path to significantly increase the SFP makeup flow rate to be greater than the minimum flow rate recommended by NEI. The HEP is set to 1.0 to indicate that gap release would occur.
***Primarily due to short time available for response (i.e., ~ 2.5 hours). OCPs 1 and 2 (i.e., during refueling) have the reactor cavity and SFP hydraulically connected, which provides more time than OCP3.

8.4 Discussion and Summary

This SFP HRA study identifies a set of plant damage states and calculates the corresponding HEPs; however, it does not calculate the conditional probabilities of the damage states. The following information summarizes the human performance insights:

- The HEPs of the SFP no leakage scenarios are negligible because of the long time available for response. The scenarios in OCPs 4 and 5 would not lead to gap release of the SFP fuel because of the low spent fuel decay heat. These two groups of scenarios share 99.2 percent of the probability (i.e., 0.992). In other words, given the 0.5–1.0g earthquake, the SFPS estimates minimum 99.2-percent conditional probability that a gap release would not occur.

- 500 gpm of injection is not sufficient to prevent gap release in the OCP1 moderate leak scenarios, as determined in the SFPS. The SFPS did not perform sensitivity calculations to determine the NEI recommended flow rates (either injection or spray) to prevent gap release in this case. Therefore, this HRA study assumes that the plant staff would need to connect more than the procedurarlized two hoses to the portable diesel pump and use more than two spray nozzles to provide sufficient cooling. Because TSG-4.1 only provides instructions on establishing two hoses and two spray nozzles, the lack of procedures and insufficient equipment (i.e., hoses and spray nozzles) are assumed to cause the mitigation to fail in the OCP1 moderate leak scenarios. Even though the reference plant flow rates are greater than the NEI recommended minimum flow rate, sensitivity calculations on the actual flow rate from the spray nozzle would be needed for a more detailed assessment.

- The available time for SFP mitigation is determined by the shorter time of either the SFP water draining to the top of the fuel or the refueling floor reaching 140°F. In the OCP 1 and 2 small leakage scenarios, the refueling floor reaches 140°F earlier than the time necessary for the SFP water to drain to the top of the fuel rack thus causing refueling floor temperature to become the limiting factor for determining HEPs

- The two spray nozzles (as illustrated in TSG-4.1) for SFP makeup are set up in high radiation areas. Delivering the same amount of flow from a low-dose area (e.g., near the wall next to the storage pool) would significantly increase the available time because using the time necessary for the SFP water level to reach the top of the fuel rack as a criterion is based on the radiation level at the locations of the spray nozzles, as specified in TSG-4.1. Moving the spray nozzles setup locations to a lower dose area would significantly increase the mitigation success probabilities for moderate leak scenarios for which the time necessary for SFP water to drain to the top of the fuel rack is the limiting factor.

- The fire system availability (from earthquake-induced fire piping rupture) affects OCP 3 moderate leakage scenarios but not small leak scenarios because the small leak scenarios have at least 13 hours for mitigation deployment. Instructions on how to quickly determine whether the fire system can deliver sufficient flow for mitigation may improve the probability of successful mitigation.

The success criterion of this HRA study is to prevent a radioactive release from the Unit 3 SFP fuel rods. The mitigation strategies that emphasize keeping the radioactivity released from the fuel rods on site are not within this HRA scope. Deploying these strategies could mitigate radioactive releases to the environment.

The HEP results shown in Table 49 are based on the assumptions that mitigation equipment is available, there is no combination of Unit 3 reactor core damage and primary containment failure that causes inaccessibility of the refueling floor, and there is sufficient staff to deploy for the Unit 3 SFP mitigation. If the earthquake damages multiple reactors and SFPs some of the above assumptions may not apply. An analysis of these issues would require the performance of a PRA and associated HRA.

9. CONSIDERATION OF UNCERTAINTY

This section catalogues a set of sensitivity analyses to better understand the potential effect of certain assumptions on the results of this study. The sensitivity analyses include those for analyzing additional plant states (e.g., 1x8 pattern in a high-density loading configuration) and for analyzing parameter/model uncertainties (e.g., hydrogen combustion ignition). The assumptions analyzed were chosen from the list of key assumptions compiled in Section 2, based on their perceived importance and project constraints.

9.1 Sensitivity to Hydrogen Combustion (MELCOR)

A sensitivity calculation was performed to examine the response of the SFP to the hydrogen combustion ignition criterion. This calculation involved reducing the hydrogen concentration from 10 percent to 7 percent given the inherent uncertainties in this parameter discussed before. The case that showed the strongest sensitivity to this parameter is the unmitigated high-density, moderate leak size scenario from OCP 2. The base case reactor building concentration of gases in Figure 105 shows that, by the time the hydrogen concentration exceeds the ignition criterion of 10 percent, the oxygen concentration is below the 5-percent limit and no hydrogen combustion is predicted. However, at about 18 hours, both the hydrogen and oxygen concentrations are above 7 percent, which can support a hydrogen combustion. Figure 106 shows the mole fraction of gases for this sensitivity case. At about 18 hours, the hydrogen combustion consumes the hydrogen in the building as evidenced by the rapid decrease in the hydrogen concentration and is accompanied by a sudden increase in the oxygen concentration as the failure of the reactor building causes the outside air to enter. Following the air ingress, the clad oxidation power significantly increases (compare the base case in Figure 107 with the sensitivity case in Figure 108). The higher oxidation power leads to higher clad temperatures[42] (Figure 109 and Figure 110) and additional release of fission products from the fuel and release to the environment (Figure 111 and Figure 112). The cesium release fraction of 50 percent for this sensitivity is much higher than the base case of 1.6 percent (see Table 27), and it is comparable to the release fraction of 49 percent for the uniform pattern (see Table 50).

[42] The failure of the fuel rods leads to formation of debris that continues to release fission products.

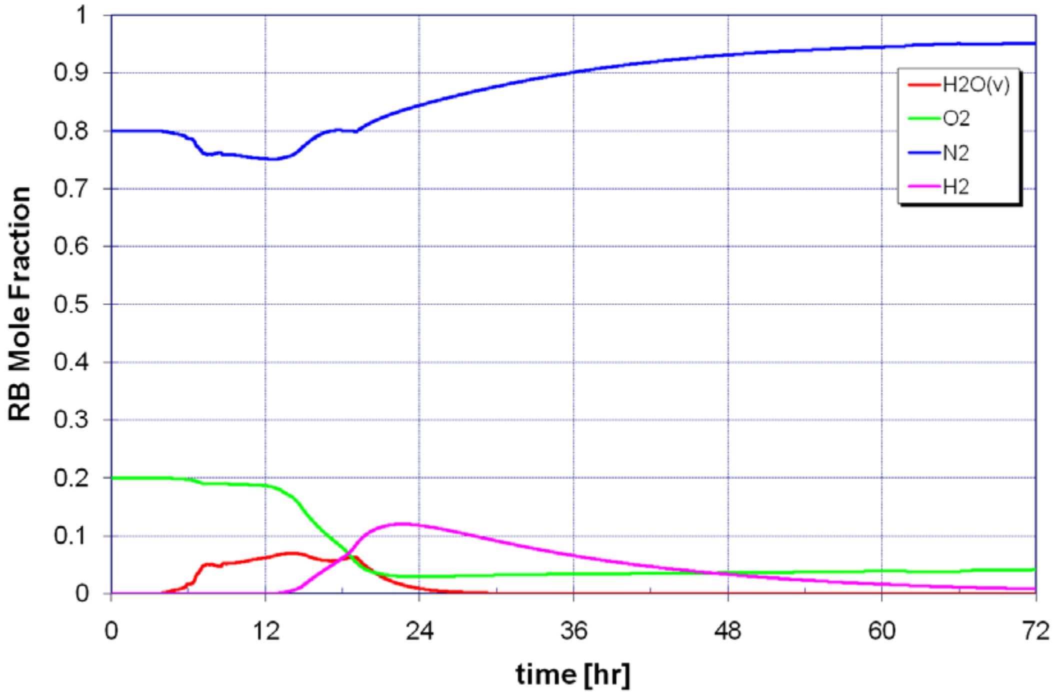


Figure 105 Reactor building mole fraction for unmitigated high-density moderate leak (OCP2)

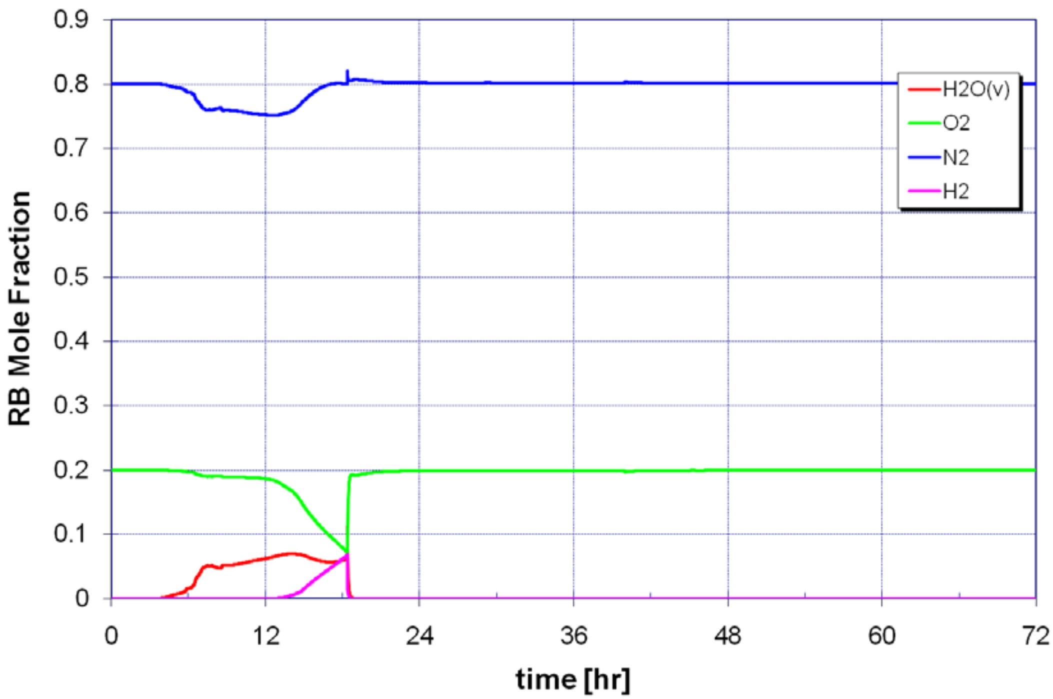


Figure 106 Reactor building mole fraction for unmitigated high-density moderate leak (OCP2-S)

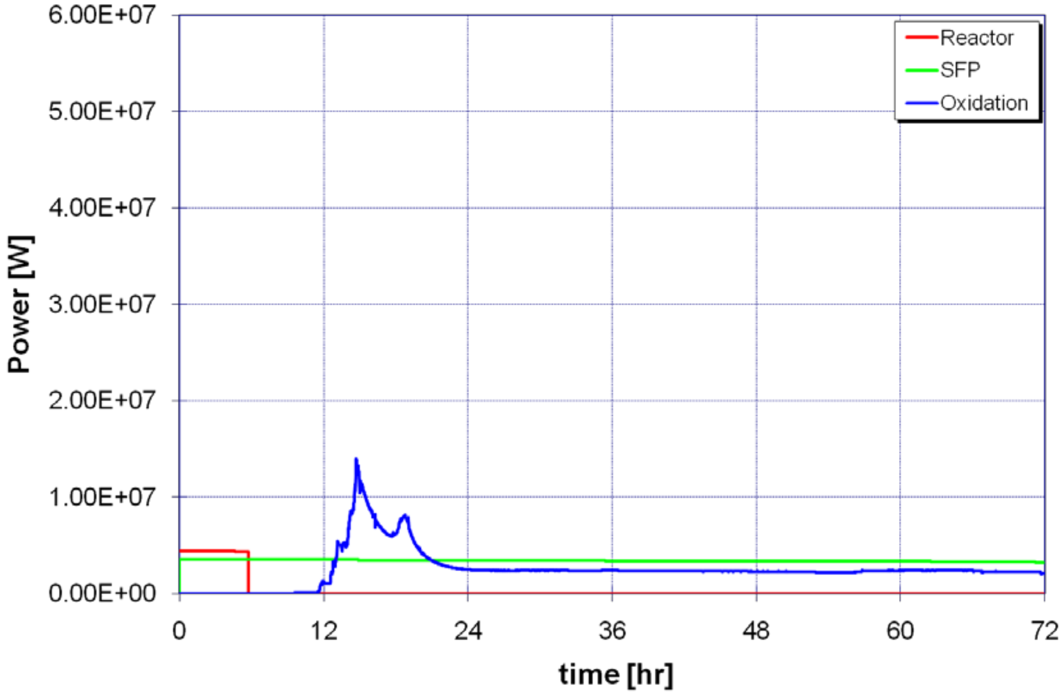


Figure 107 SFP power for unmitigated high-density moderate leak (OCP2)

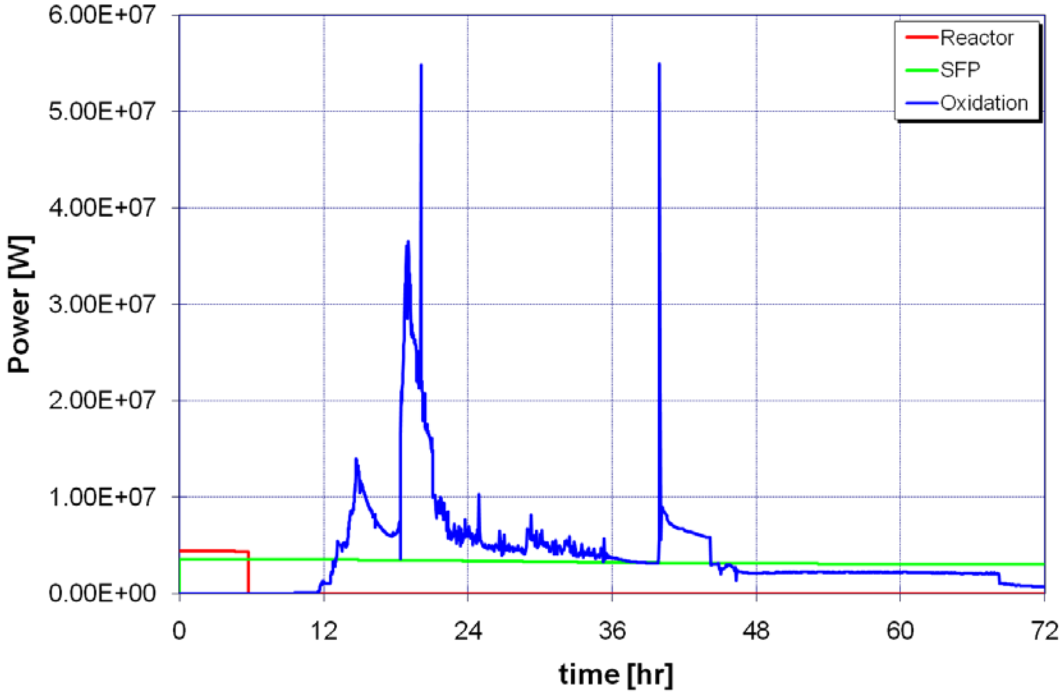


Figure 108 SFP power for unmitigated high-density moderate leak (OCP2-S)

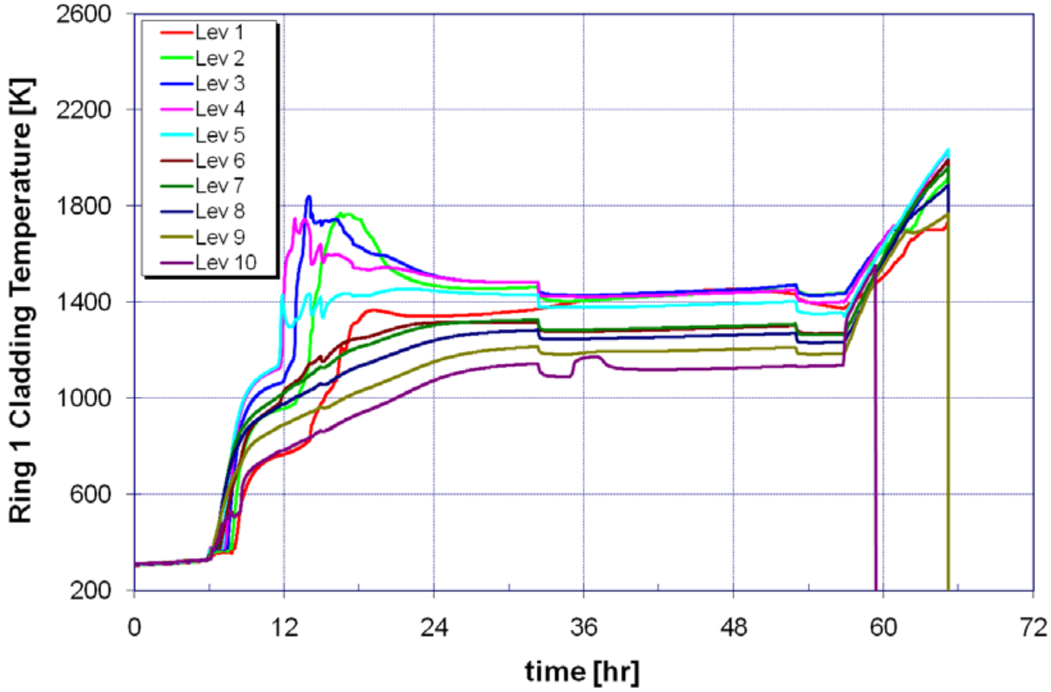


Figure 109 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP2)

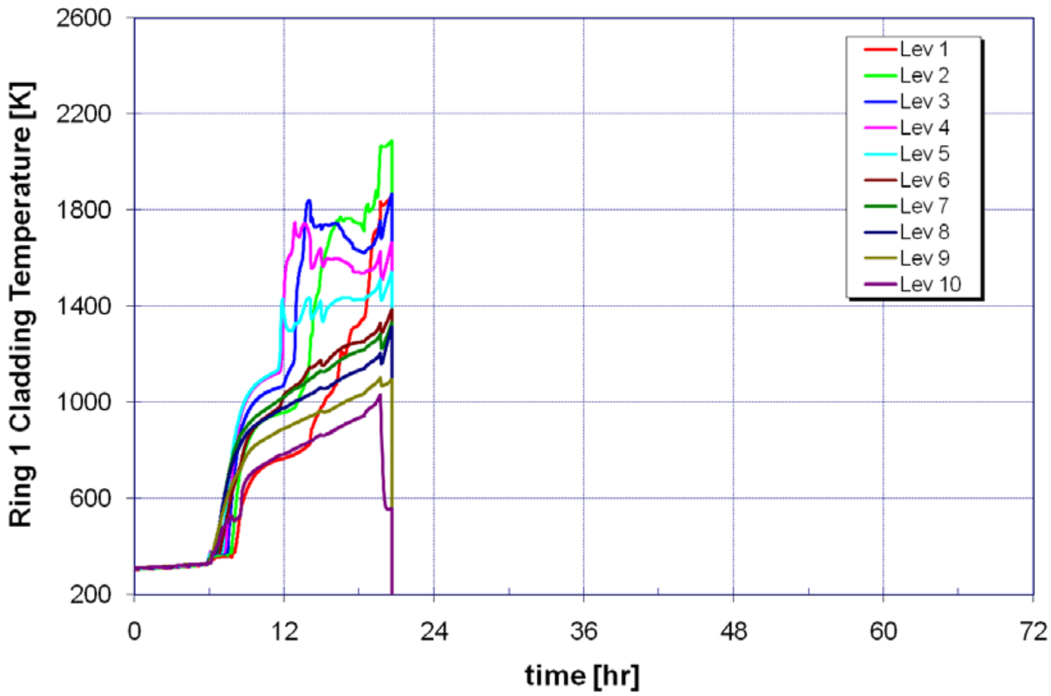


Figure 110 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP2-S)

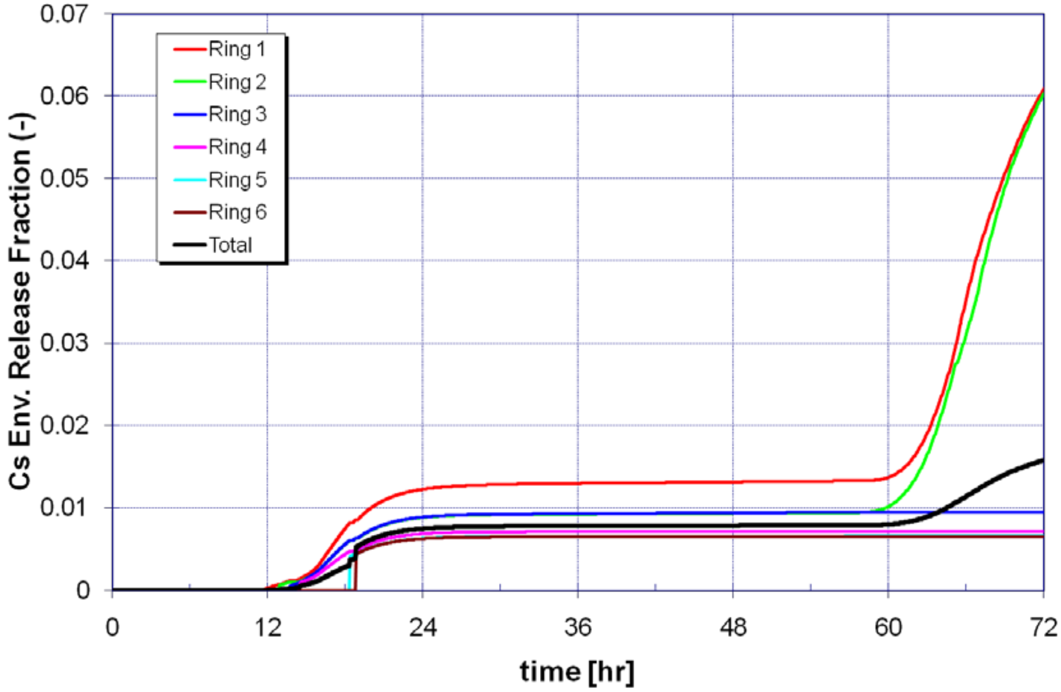


Figure 111 Cesium environmental release fraction for unmitigated high density moderate leak (OCP2)

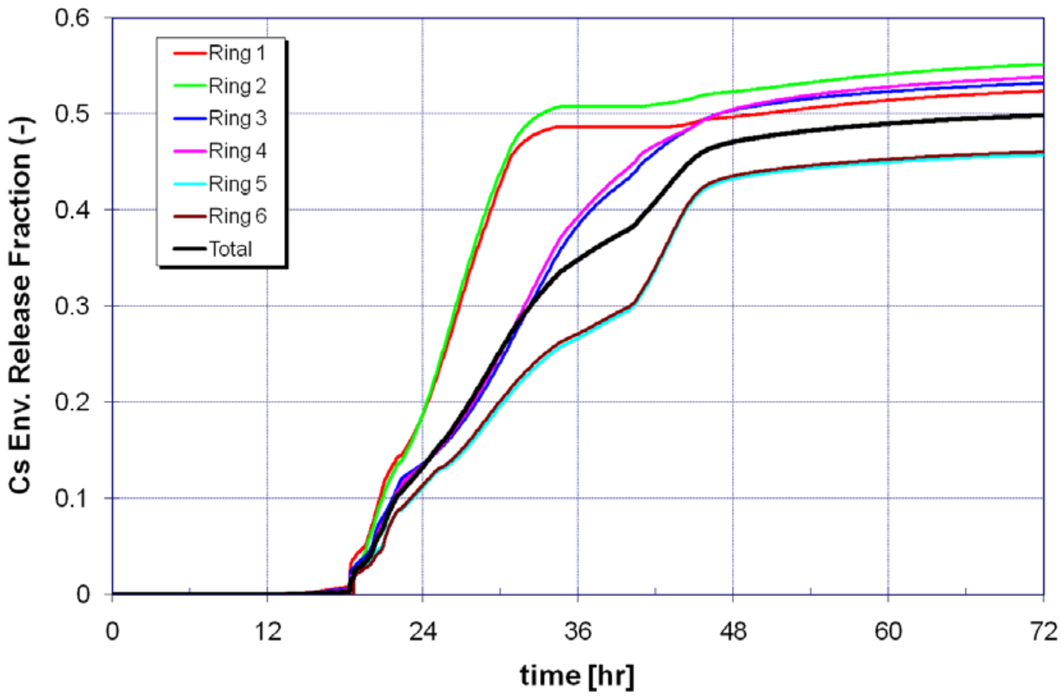


Figure 112 Cesium environmental release fraction for unmitigated high density moderate leak (OCP2-S)

9.2 Sensitivity to 1x8 Fuel Assembly Pattern (MELCOR)

This sensitivity involves a more favorable fuel pattern in which the hot assemblies are surrounded by eight cold assemblies. Figure 113 shows the assembly layout in a 1x8 pattern in which the 284 assemblies from the last offload are grouped into Rings 1, 3, and 5 (see Figure 46 for the 1x4 pattern). Rings 2, 4, and 6 contain all of the old fuel and have a total of 2,771 assemblies with their total decay heat distributed in each ring scaled by the number of assemblies.

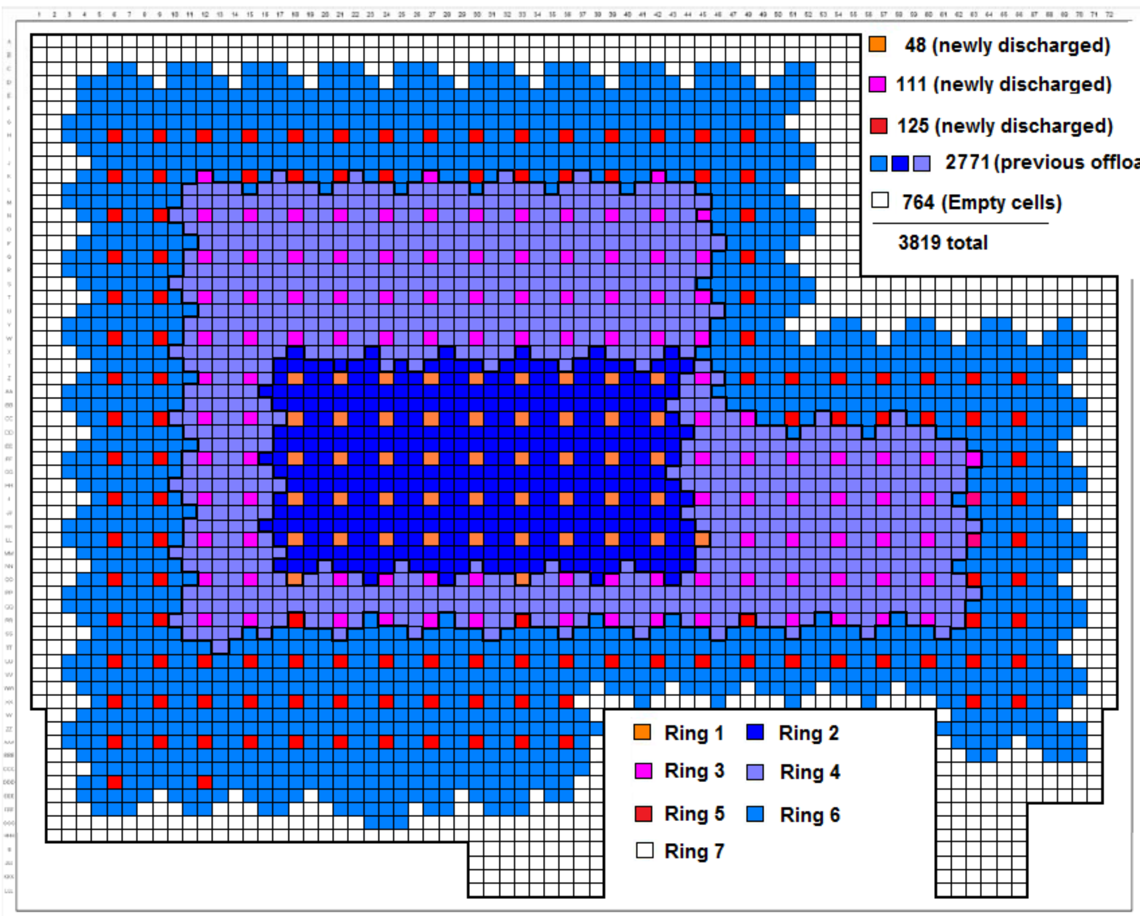


Figure 113 Layout of assemblies for OCP3 high density (1x8) model

A number of sensitivity calculations were performed for the high density, small leak scenarios in OCP2 and OCP3 (which had the highest release as a result of hydrogen combustion). Figure 114 and Figure 115 show the thermal response of the fuel in the 1x8 configuration for OCP3. Figure 114 shows that the highest power fuel assemblies in Ring 1 do not undergo a zirconium fire and the temperatures remain low enough to avoid gap release for the duration of the transient. The midplane fuel temperatures in the pool shown in Figure 115 have a more uniform heat up of the fuel assemblies than the comparable 1x4 pattern. There is more mass of the cold assemblies in the 1x8 pattern, which leads to lower heatup of the fuel. The fuel thermal response in the 1x8 pattern can be contrasted to the 1x4 pattern as shown in Figure 116 and Figure 117. For the 1x8 calculation, no release occurs from the fuel through 72 hours. In the 1x4 layout, a zirconium fire propagation began at 40 hours, which led to a 42-percent release of cesium inventory to the environment. For the OCP2 configuration results shown in Figure 118 and Figure 119, the decay heat is high enough to cause a zirconium fire in the hottest assemblies, even though the peak fuel temperatures in the 1x8 pattern are somewhat lower. The beneficial effect of the 1x8 pattern is also evidenced by the lower release fractions, as shown in Figure 120 and Figure 121.

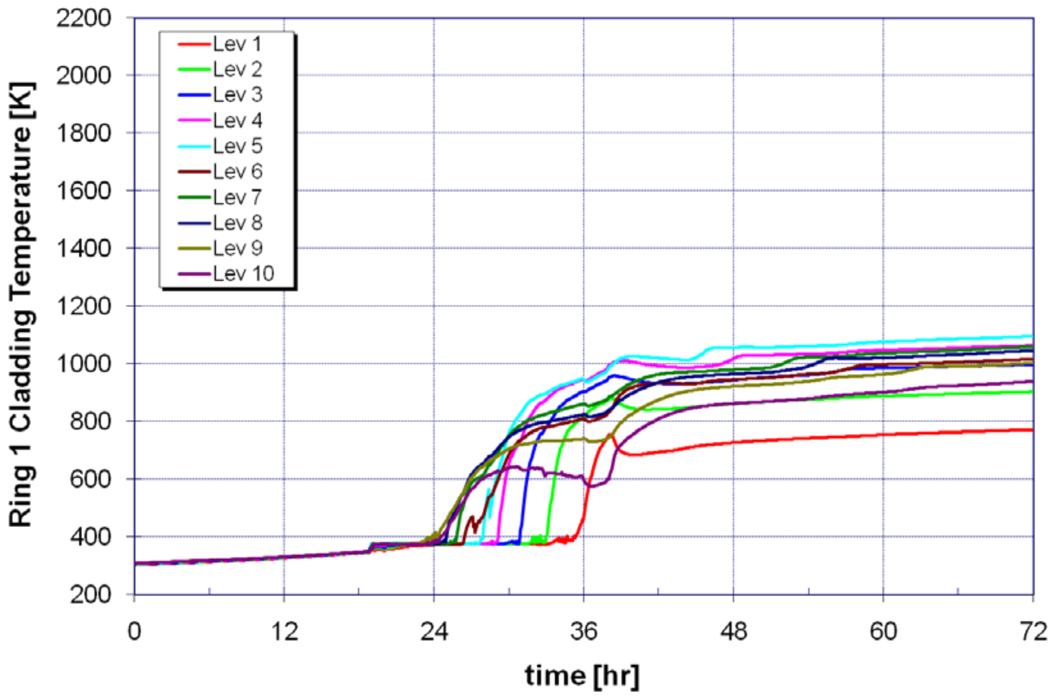


Figure 114 Ring 1 clad temperature for unmitigated high-density small leak (OCP3; 1x8)

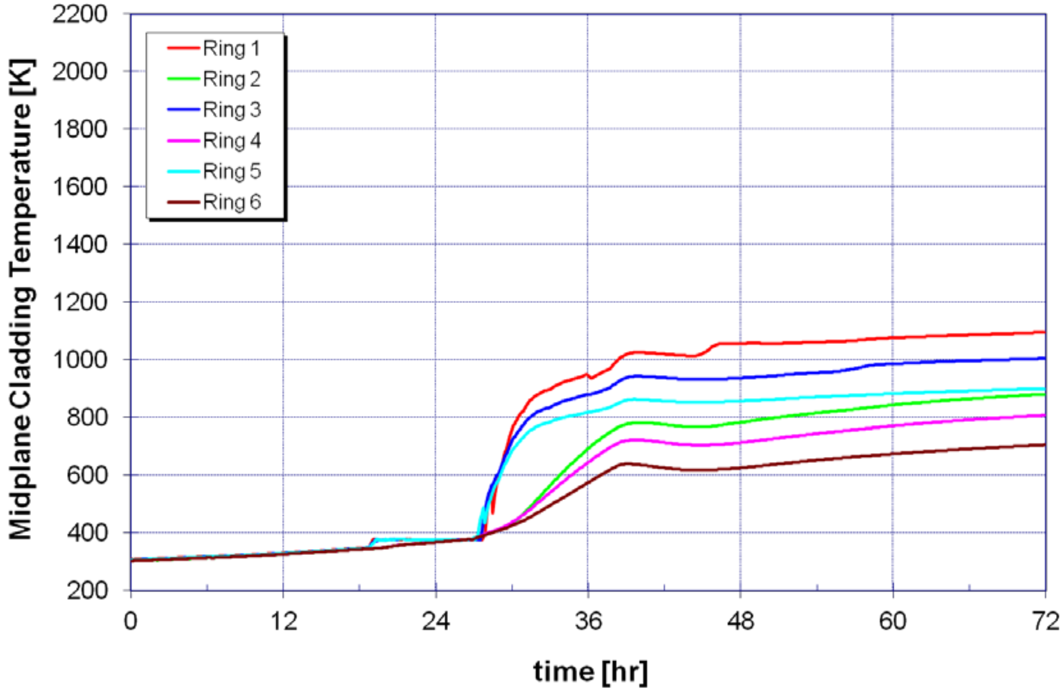


Figure 115 Midplane clad temperature for unmitigated high-density small leak (OCP3; 1x8)

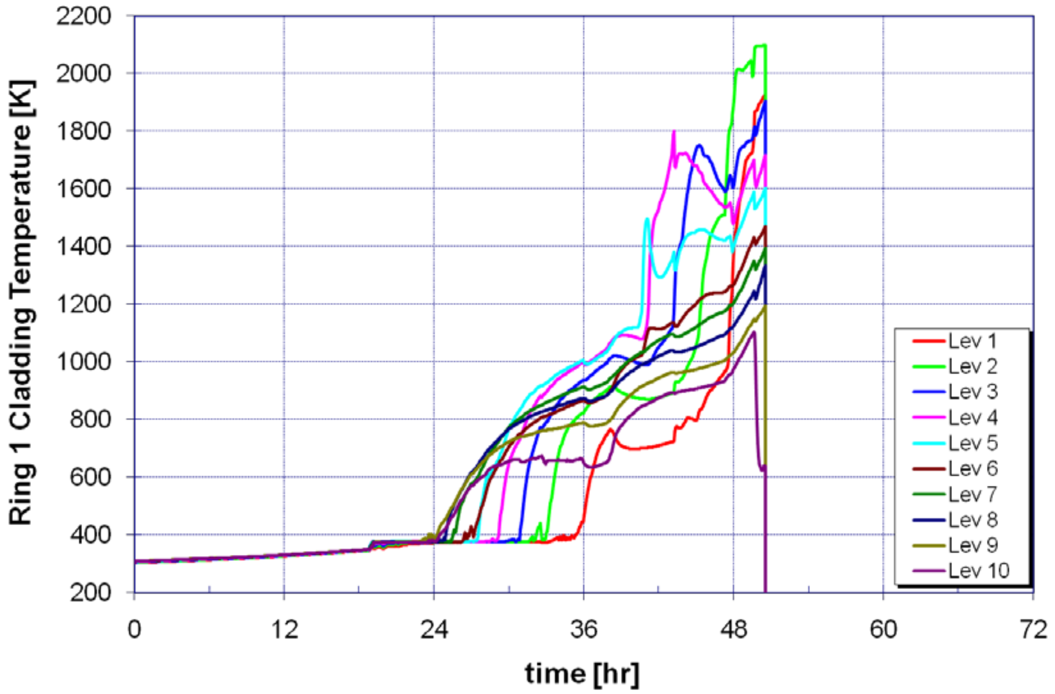


Figure 116 Ring 1 clad temperature for unmitigated high-density small leak (OCP3; 1x4)

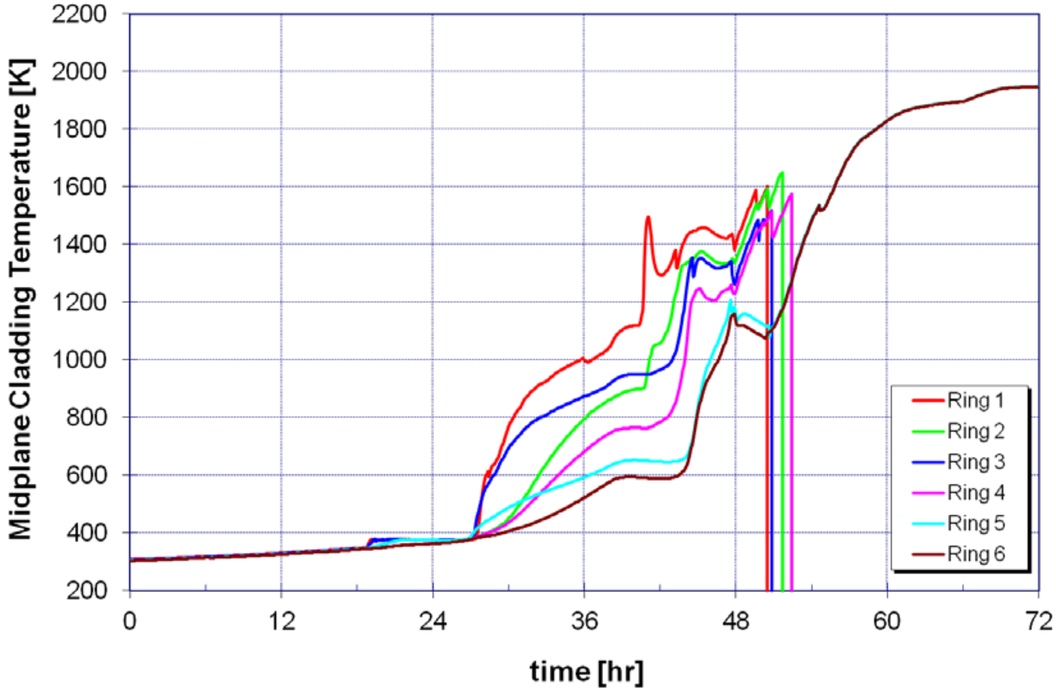


Figure 117 Midplane clad temperature for unmitigated high-density small leak (OCP3; 1x4)

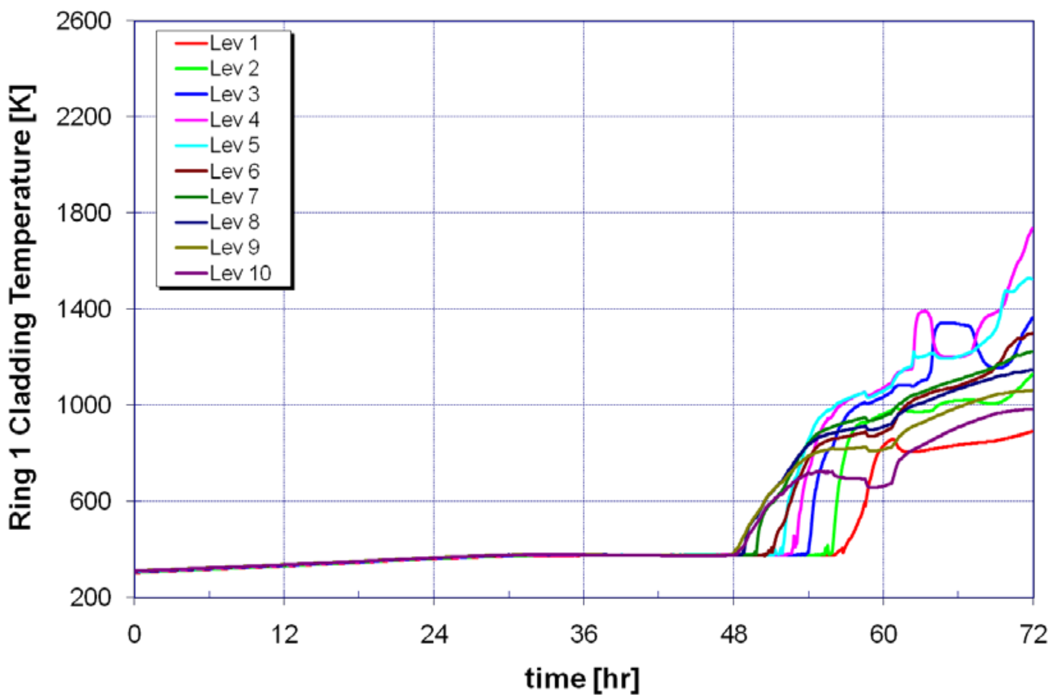


Figure 118 Ring 1 clad temperature for unmitigated high-density small leak (OCP2; 1x8)

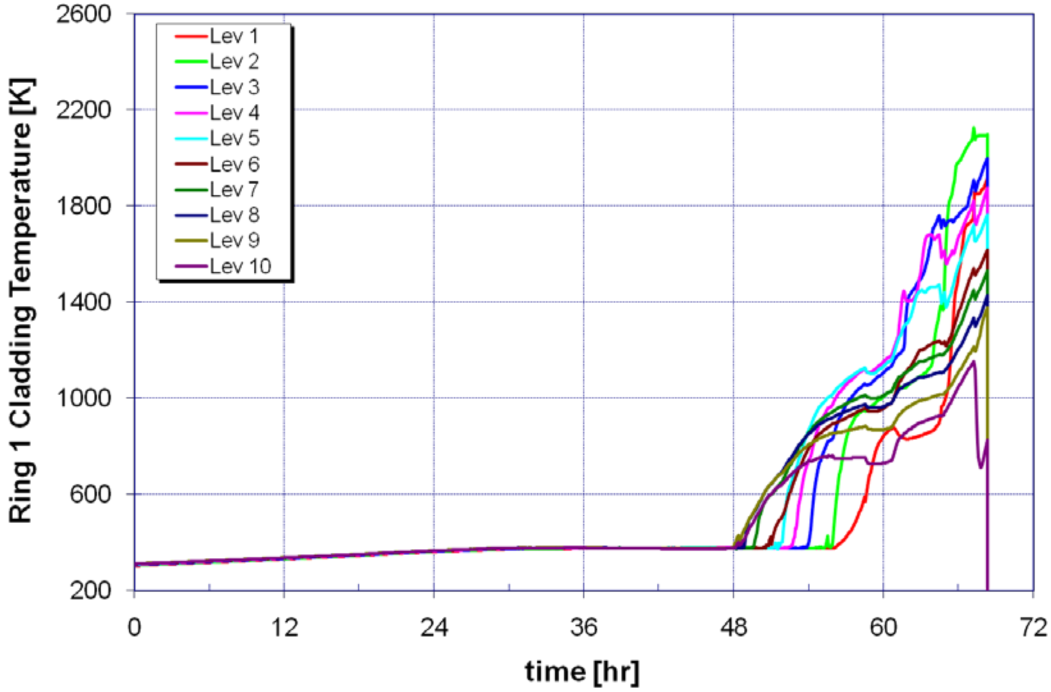


Figure 119 Ring 1 clad temperature for unmitigated high-density small leak (OCP2; 1x4)

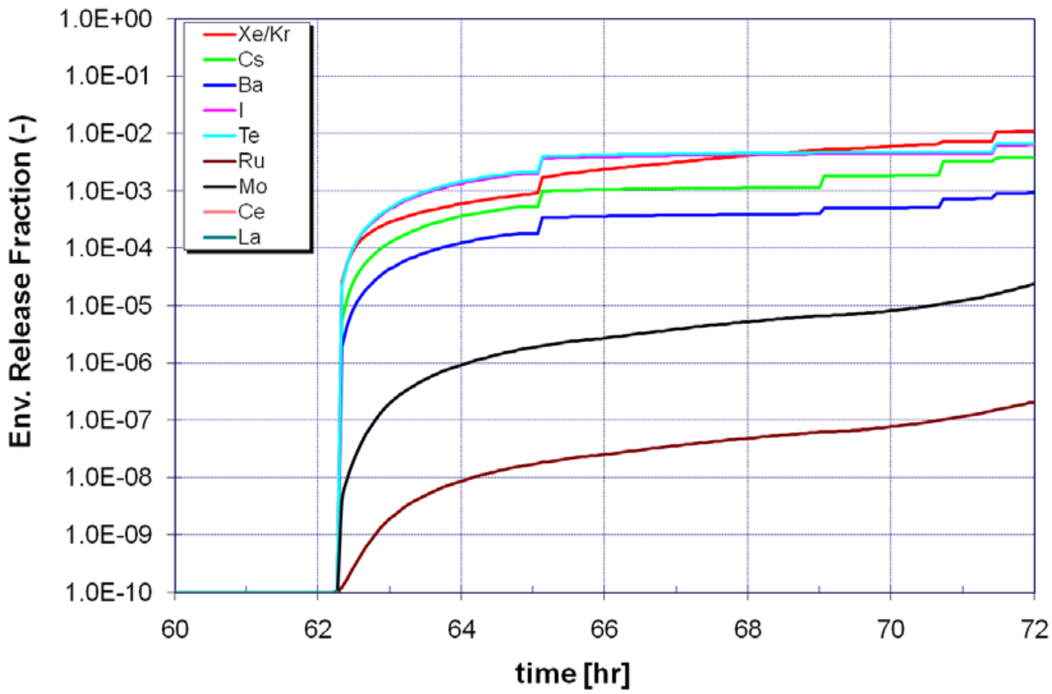


Figure 120 Environmental release fractions for unmitigated high-density small leak (OCP2; 1x8)

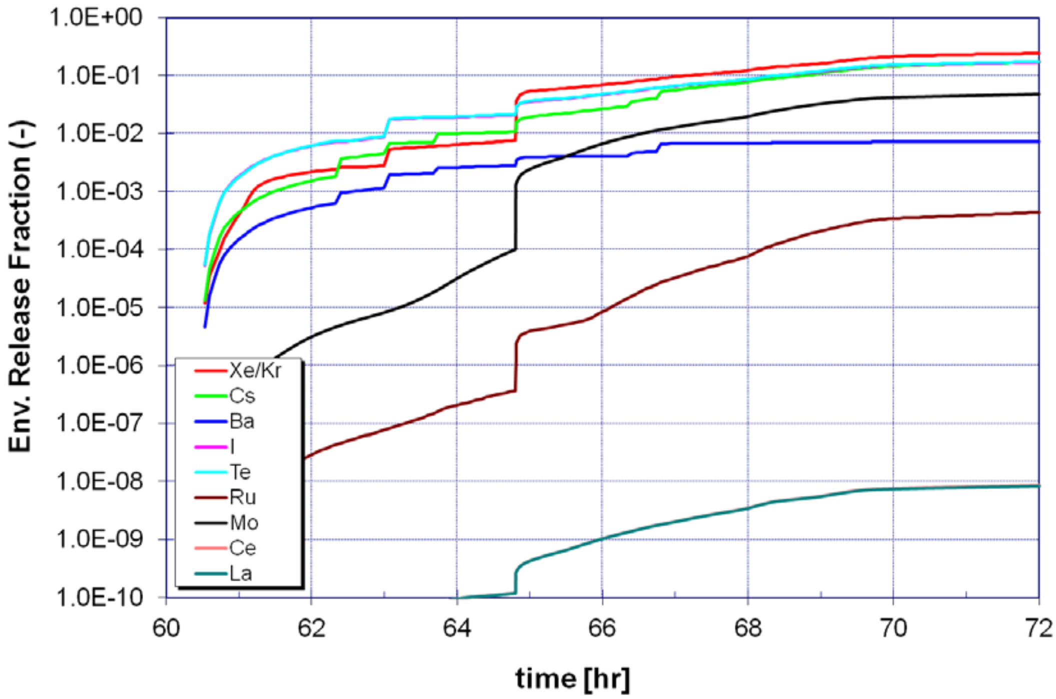


Figure 121 Environmental release fractions for unmitigated high-density small leak (OCP2; 1x4)

9.3 Sensitivity to a Contiguous (Uniform) Fuel Pattern during an Outage (MELCOR/MACCS2)

The reference plant studied has prearranged the SFP such that discharged assemblies can be placed directly into a 1x4 (actually 1x8 in the case of PBAPS) arrangement for the last two outages for both operating units. This approach is consistent with the requirements previously discussed in Section 5.1. However, those requirements do allow for the fuel to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. A requirement is 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 in which the fuel is unfavorably arranged during the outage to demonstrate the effect of this aspect on the results.

Figure 122 and Figure 123 show the layout of assemblies for the OCP1 and OCP2 uniform configuration. For the 1x4 pattern (see Figure 44), 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 pattern (Figure 122), the surface areas between Rings 1 and 2 and between Rings 3 and 4 were effectively reduced by about an order of magnitude, assuming that all of 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 was based on four panels per assembly. In the uniform pattern, the number of panels per assembly is estimated as 0.4 for Ring 1 ($4(88)^{1/2}/88$) and 0.3 ($4(196)^{1/2}/196$) for Ring 3. This is a stylized representation of a uniform configuration which limits the areas (and thus total heat transfer) between the hot rings and the rest of the assemblies in the pool.

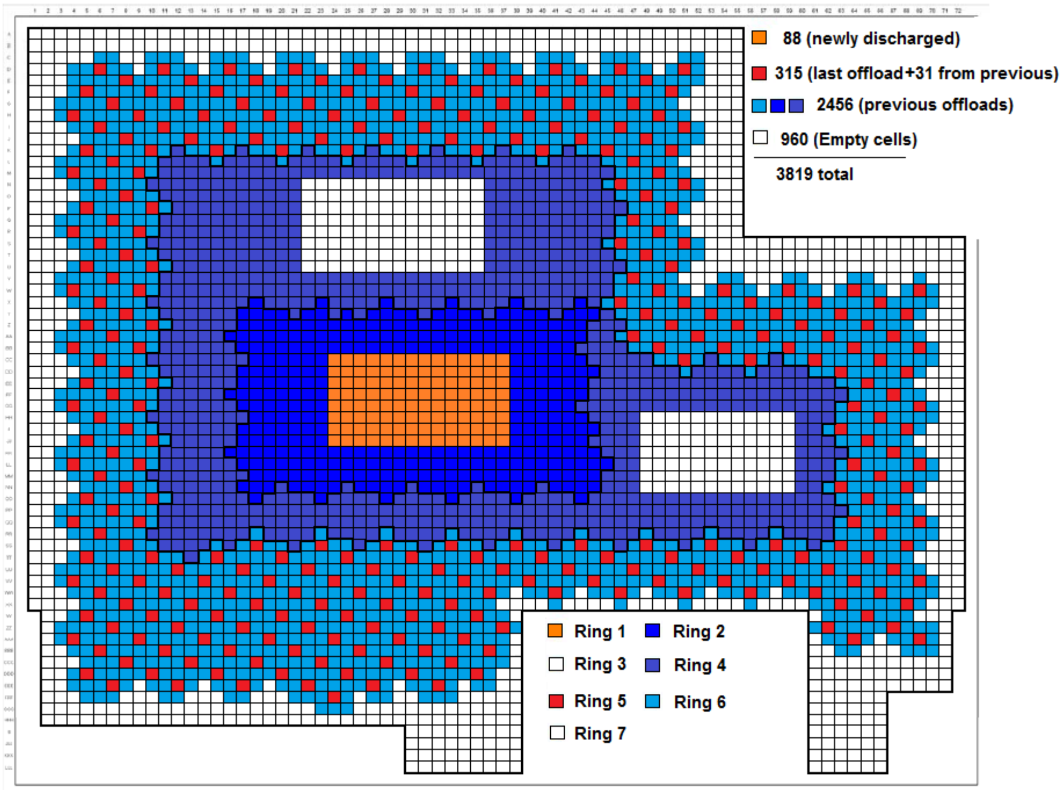


Figure 122 Layout of assemblies for OCP1 high-density (uniform) model

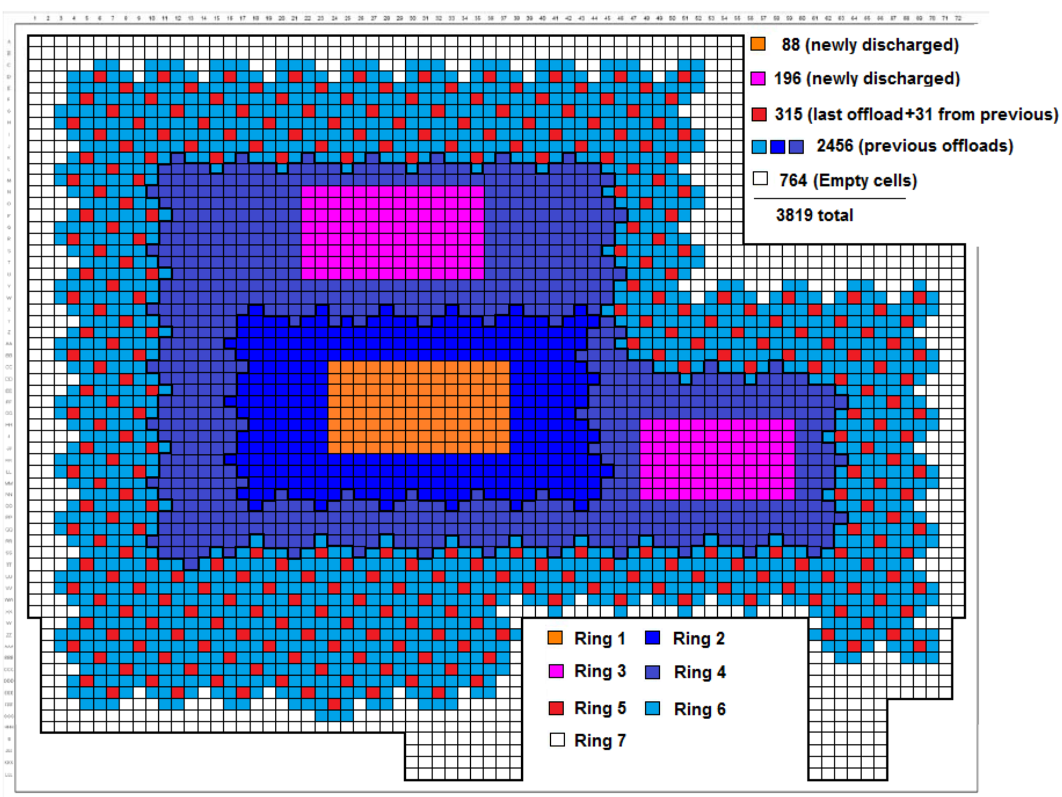


Figure 123 Layout of assemblies for OCP2 high-density (uniform) model

Unmitigated Moderate Leak (OCP1 Uniform) Scenario

Figure 124 and Figure 125 show the results of the calculation for the uniform OCP1. A comparison of the heatup with the 1x4 geometry (Figure 67) shows the higher temperatures in the uniform Ring 1 configuration because 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 heatup as the oxygen in the building is depleted, and formation of debris restricts airflow through the assemblies. Eventually, all of the fuel in Ring 1 collapsed and formed a debris bed. There is continuous release from Rings 1 and 2 and the overall cesium release to the environment is about twice of that in the 1x4 geometry (see Figure 72).

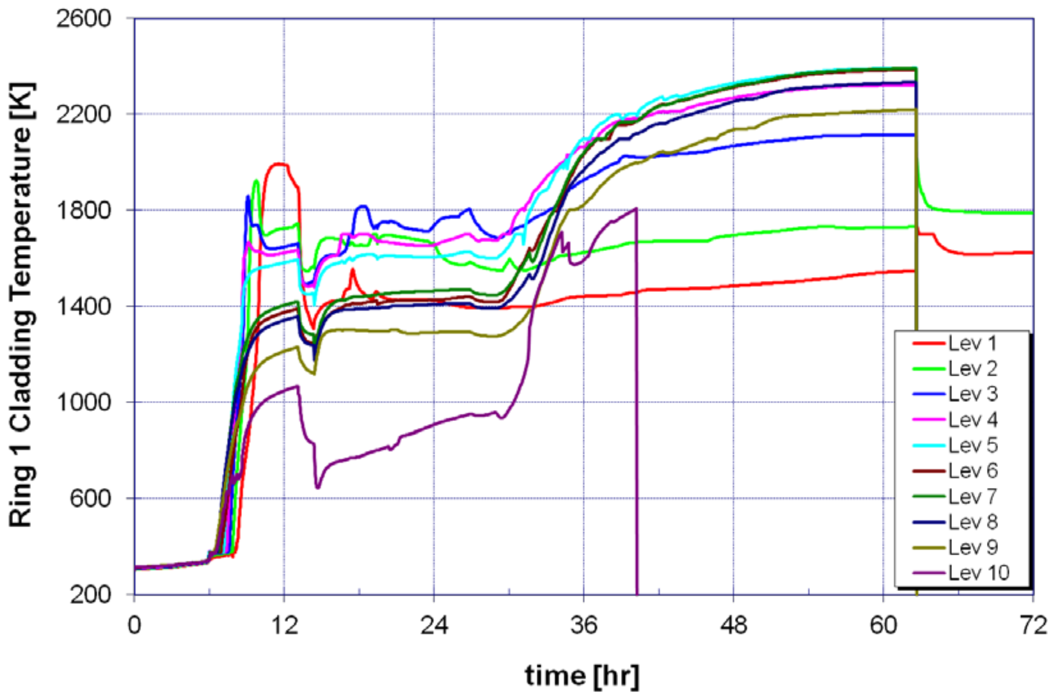


Figure 124 Ring 1 clad temperature for unmitigated uniform high-density moderate leak (OCP1)

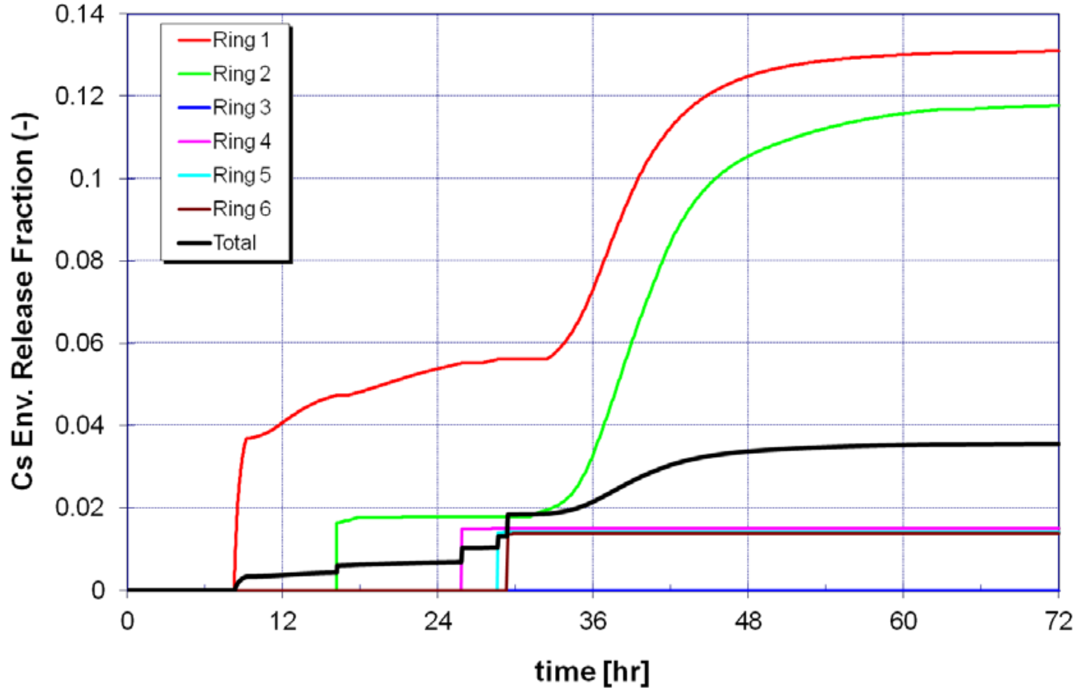


Figure 125 Cesium environmental release fraction for unmitigated uniform high-density moderate leak (OCP1)

<u>Mitigated Moderate Leak (OCP2 Uniform) Scenario</u>

For the mitigated case in the OCP2 uniform pattern that had the highest cesium release fraction (1.2 percent), a number of calculations were performed to determine the effectiveness of mitigation. The same scenario in the 1x4 pattern did not have any release. The overall behavior of fuel temperature is similar to the 1x4 pattern cases in OCP2 (not shown) and OCP 1 (Figure 77), but the fuel is experiencing a higher temperature that gradually decreases. For this base case (Figure 126), temperatures are high enough to cause a gap release and more gradual release of fission products from the fuel. Figure 127 illustrates the calculation for the 200-gpm spray instead of the 500-gpm makeup water, which actually shows a rapid heatup before the temperatures are stabilized.[43] A calculation was performed to test the effectiveness of a higher spray flow rate of 500 gpm and, as indicated in Figure 128, the fuel temperature is stabilized at much lower temperatures without release of fission products from the fuel. In all of the spray calculations performed in this study, the simple flow regime model was disabled because of a more stable and faster calculation, and the previous results from OCP3 had already demonstrated that both models predict comparable maximum clad temperatures.

[43] The initial higher temperature spike for the 200 gpm spray, as compared to lower temperatures for the 500 gpm injection case, results from a combination of the leakage versus makeup rate for this particular scenario. For the 500 gpm injection case, the lower portions of the assemblies are covered with water and high decay heat promotes steam cooling of the exposed portions of the fuel. A larger hole size would not have the benefit of steam cooling, and the spray is expected to perform better for a wide range of conditions. Even in this particular case, under quasi-steady conditions the fuel temperatures are generally lower for the spray case.

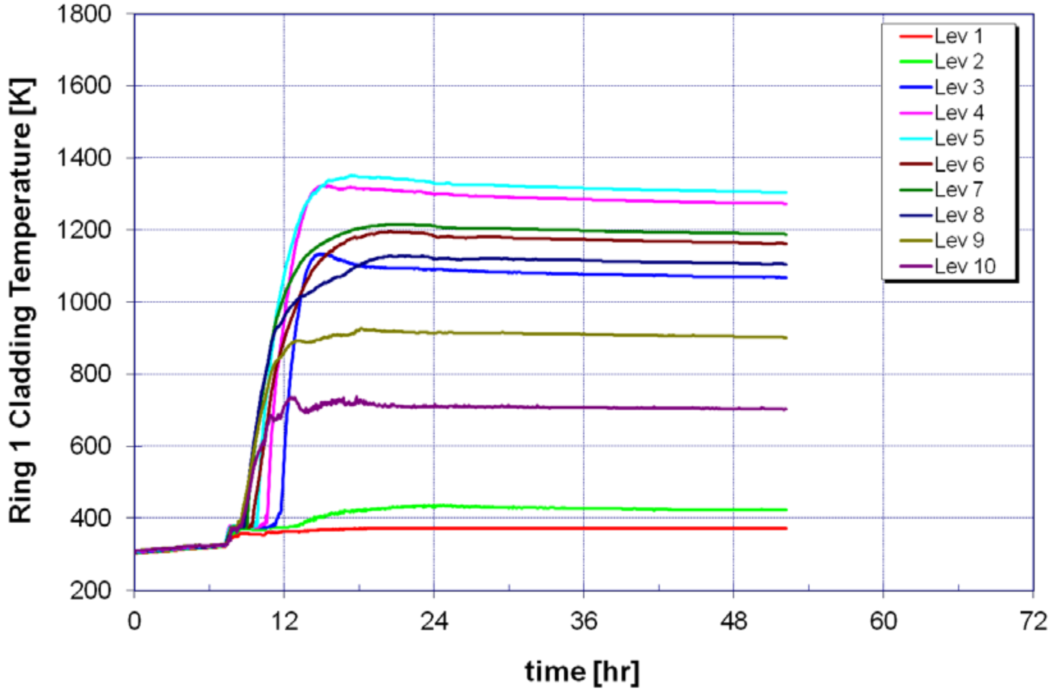


Figure 126 Ring 1 clad temperature for mitigated uniform high-density moderate leak (OCP2) with 500 gpm injection

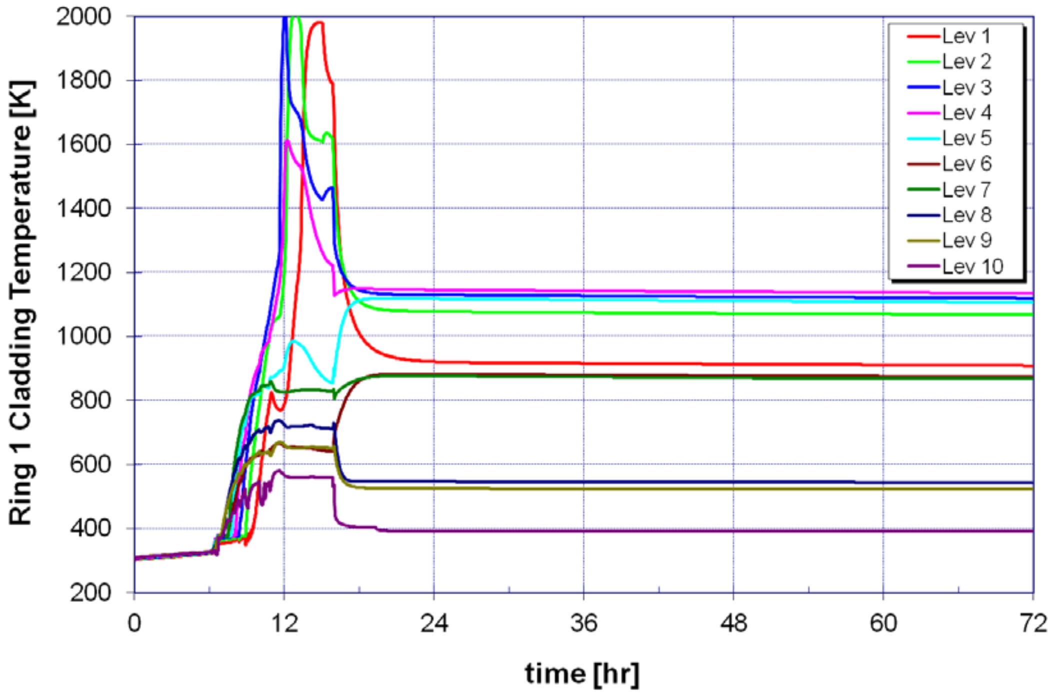


Figure 127 Ring 1 clad temperature for mitigated uniform high-density moderate leak (OCP2) with 200 gpm spray

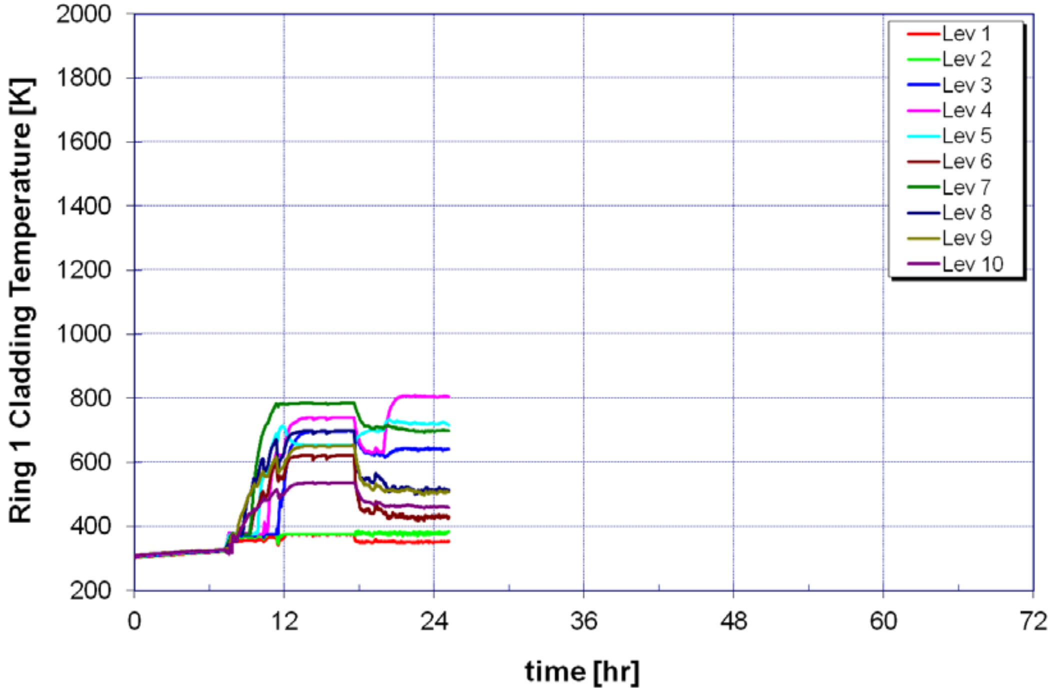


Figure 128 Ring 1 clad temperature for mitigated uniform high density moderate leak (OCP2) with 500 gpm spray

Table 50 Summary of Release Characteristics for High-Density, Uniform Pattern

High Density Case #	**Scenario Characteristics**					**Release Characteristics**			
	SFP Leakage?	50.54 (hh)(2) Equip-ment?	Fuel Uncovery (hr)	Gap Release (hr)	Hydro-gen Defla-gration (hr)	Cs release at 72 hours	Cs-137 (MCi) Released	I release at 72 hours	I-131 (MCi) Released
OCP1	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
OCP2	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	21.6	49.1%	22.71	68.4%	7.65

For the offsite consequence analysis, the sequences with recently discharged fuel in a uniform configuration were binned in a similar manner to the low-density and high-density (1x4) loading scenarios. Since the licensee must either preconfigure the SFP to allow direct placement of discharged fuel in or move their recently discharged fuel to a more favorable configuration after a certain amount of time, this sensitivity simply assumes that the high-density uniform case becomes identical to the high-density (1x4) case after OCP2 (i.e., that the actions to meet the requirements on fuel pattern discussed in Section 5.1 are taken at the end of OCP2). While the uniform case has different release categories, the situations that lead to release are largely the same as the low-density and high-density (1x4) base cases. The one exception is for OCP2 with a moderate leak and deployed 10 CFR 50.54(hh)(2) equipment, in which case a successful deployment of mitigation equipment is expected to prevent release for the high-density (1x4) and low-density scenario, but not for the sensitivity scenario of recently discharged fuel in a uniform configuration.

Table 51 Listing of Uniform Pattern Release Sequences

<table>
<tr><th colspan="8">High Density (uniform) Loading</th></tr>
<tr><th colspan="4">Unsuccessful mitigation</th><th colspan="4">Deployed 50.54(hh)(2)</th></tr>
<tr><th colspan="2">Sequence</th><th>Release Frequency (/yr)*</th><th>Release Category</th><th colspan="2">Sequence</th><th>Release Frequency (/yr)</th><th>Release Category</th></tr>
<tr><td rowspan="2">OCP1</td><td>small leak</td><td>6E-09**</td><td>RC12</td><td>OCP1</td><td>mod leak</td><td>6E-09</td><td>RC11</td></tr>
<tr><td>mod leak</td><td>6E-09</td><td>RC23</td><td>OCP2</td><td>mod leak</td><td>2E-08</td><td>RC23</td></tr>
<tr><td rowspan="2">OCP2</td><td>small leak</td><td>2E-08</td><td>RC23</td><td colspan="4" rowspan="4">No Release</td></tr>
<tr><td>mod leak</td><td>2E-08</td><td>RC33</td></tr>
<tr><td rowspan="2">OCP3</td><td>small leak</td><td>4E-08</td><td>RC33</td></tr>
<tr><td>mod leak</td><td>4E-08</td><td>RC11</td></tr>
<tr><td colspan="2">Total</td><td>1E-07</td><td></td><td colspan="2">Total</td><td>2E-08</td><td></td></tr>
</table>

* Release frequency = initiating event frequency * ac power fragility * OCP probability * liner fragility for the specified leak size (see Section 5.6.3 for conditional probabilities)
** Example calculation: $1.7x10^{-5}$/yr · 0.84 · 0.0086 · 0.05 = $6x10^{-9}$/yr

Table 52 reports the consequence results for the sensitivity scenario of recently discharged fuel in a uniform configuration. It is similar to Table 33 for the base scenarios.

Table 52 Uniform Pattern Consequence Results

SFP Fuel Loading	High Density (uniform)	
Seismic Hazard Frequency[1] (/yr) (PGA of 0.5 to 1.0g)	1.7E-05	
50.54(hh)(2) Mitigation Credited	Yes	No
Conditional[2] Probability of Release	0.14%	0.69%
Hydrogen Combustion Event	“Not Predicted”	“Possible”
Conditional[3] Consequences (Release Frequency-Averaged[4])		
Cumulative Cs-137 Release at 72 hours (MCi)	0.5	11
	Measures Related to Individual Health and Safety	
Individual Early Fatality Risk	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles	7.3E-04[(7)]	6.9E-04
	Measures Related to Cost Benefit Analysis	
Collective Dose (Person-Sv)	1.4E+05	4.9E+05
Land Interdiction[6] (mi^2)	1.1E+03	1.3E+04
Long-term Displaced Individuals[6]	6.2E+05	5.6E+06
Consequences per year (Release Frequency-Weighted[4])		
Release Frequency (/yr)	2.3E-08	1.2E-07
	Measures Related to Individual Health and Safety	
Individual Early Fatality Risk (/yr)	0	0
Individual Latent Cancer Fatality Risk[5] Within 10 Miles (/yr)	1.7E-11	8.1E-11
	Measures Related to Cost Benefit Analysis	
Collective Dose (Person-Sv/yr)	3.1E-03	5.7E-02
Land Interdiction[6] (mi^2/yr)	2.5E-05	1.5E-03
Long-term Displaced Individuals[6] (Persons/yr)	1.4E-02	6.7E-01

[1] Seismic hazard model from USGS (Peterson et al., 2008)
[2] Given specified seismic-event occurs
[3] Given atmospheric release occurs
[4] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions (as applicable); additionally, “release frequency-weighted” results are multiplied by the release frequency.
[5] LNT and population-weighted
[6] 1st year post-accident; calculation uses a dose limit of 500 mrem per year, according to Pennsylvania Code, Title 25 § 219.51
[7] Slightly higher conditional risk with mitigation is due to a more prolonged release (allowing changes in wind direction to affect additional portions of the 10 mile area) and effective protective actions that limit individual risk (regardless of release magnitude); the difference is small compared to the reduction in release frequency.

The insights of the high density 1x4 scenario are also applicable here to the uniform pattern: There is very small likelihood of release. When there is a release, no offsite early fatalities attributable to acute radiation exposure are predicted. On average, significant land contamination is predicted when there is a release with unsuccessfully deployed mitigation. A significant numbers of latent cancer fatalities are also estimated; however, this is a small fraction of cancer fatalities from all causes, because protective actions are expected to keep doses below limits for habitation and ingestion. Overall, individual latent cancer fatality risk is very low, mainly because of the very small likelihood of release and protective actions.

Health effects that would be induced by low dose radiation are uncertain, and insights from a dose truncation for the uniform pattern scenario are similar to those for the high density 1x4 scenario. As can be seen in Table 53, dose truncation significantly lowers the estimated number of total latent cancer fatalities because the uncertain effects of small individual doses are excluded.

Table 53 Dose Truncation Comparison for Uniform Pattern

Dose-Response	High Density (1x4)	
50.54(hh)(2) Mitigation Credited	Yes	No
Conditional[1] Individual Latent Cancer Fatality Risk Within 10 Miles (Release Frequency-Averaged[2])		
Linear, No Threshold	7.3E-04[(3)]	6.9E-04
620 mrem/yr truncation	3.2E-07	1.1E-06
5rem/yr or 10rem lifetime truncation	2.3E-07	1.1E-06
Individual Latent Cancer Fatality Risk Within 10 Miles (/yr) (Release Frequency-Weighted[2])		
Linear, No Threshold	1.7E-11	8.1E-11
620 mrem/yr truncation	7.3E-15	1.3E-13
5 rem/yr or 10 rem lifetime truncation	5.3E-15	1.3E-13

[1] Conditional on a release occurring
[2] Results from a release are averaged over potential variations in leak size, time since reactor shutdown, population distribution, and weather conditions; additionally, "release frequency-weighted" results are multiplied by the release frequency.
[3] Slightly higher conditional risk with mitigation is due to a more prolonged release (allowing changes in wind direction to affect additional portions of the 10 mile area) and effective protective actions that limit individual risk (regardless of release magnitude); the difference is small compared to the reduction in release frequency.

Similar to the high density (1x4) scenario without deployed 50.54(hh)(2) equipment, the uniform scenario is sometimes predicted to have significant releases when there is a hydrogen combustion. Once again, this is because hydrogen combustion leads to much more zirconium oxidation from the influx of air, as well as a much smaller building decontamination factor. A comparison of this sensitivity analysis of a uniform fuel pattern and those of the base case (i.e. the high density 1x4 pattern and the low density configuration) are quantified in the Table 54 and Table 55.

Table 54 Consequence Comparison – High Density (1x4 and Uniform) Loading Without Successful 50.54(hh)(2) Mitigation

Benefit of High Density (1x4) vs. High Density (uniform) Fuel Loading (Scenario Specific, Weather-Averaged, Release Frequency-Averaged, Unsuccessful Deployment of 50.54(hh)(2))		
Type of Consequence	Consequences** (/yr)	Conditional* Consequences
	Reduction Factor (dimensionless)	
Release Frequency	1.0	-
Individual Latent Cancer Fatality Risk*** for 0-10 Miles	1.6	1.6
Collective Dose (Person-Sv)	1.4	1.4
Land Interdiction (mi^2)	1.4	1.4
Displaced Individuals (Persons)	1.4	1.4

* Conditional on a release occurring (frequency of 1E-7 per year, or lower)
** Release Frequency-Weighted
*** Linear-No Threshold, Population-Weighted

As can be seen in Table 54, without mitigation in the high-density configurations, consequences of the uniform pattern are discernibly higher than the 1x4 pattern. While other contributors could be partially responsible for this difference, this is largely because the accident progression analysis predicts a uniform pattern to sometimes have more detrimental hydrogen combustion events than the 1x4 pattern.

Table 55 compares consequences of high and low density with a uniform pattern for the high density loading, without mitigation. This is similar to Table 37 which uses a 1x4 pattern for the high density loading; however, Table 55 has larger differences because of the larger consequences predicted from a uniform pattern.

Successfully deployed mitigation in the high density configuration lowers the release frequency and most conditional consequences for both uniform and 1x4 patterns. For both patterns, hydrogen combustions are not predicted with MELCOR when 50.54(hh)(2) mitigation is successfully deployed, and therefore the relatively large releases are also not predicted. However, deployed mitigation is not quite as effective in the uniform pattern as it is for the 1x4 pattern. Additionally, deployed mitigation is predicted to be unsuccessful at preventing an additional release in the uniform pattern scenario as compared to the 1x4 pattern. The differences in the release frequencies and conditional consequences can be seen by comparing Table 52 and Table 33.

Table 55 Consequence Comparison – High (Uniform) Density / Low Density Loading Without Successful 50.54(hh)(2) Mitigation

Benefit of High (Uniform) Density / Low Density Loading (Scenario Specific, Weather-Averaged, Release Frequency-Averaged, Unsuccessful Deployment of 50.54(hh)(2))		
Type of Consequence	Consequences** (/yr)	Conditional* Consequences
	Reduction Factor (dimensionless)	
Release Frequency	1.0	-
Individual Latent Cancer Fatality Risk*** for 0-10 Miles	3.4	3.4
Collective Dose (Person-Sv)	18	18
Land Interdiction (mi^2)	78	78
Displaced Individuals (Persons)	70	70

* Conditional on a release occurring (frequency of 1E-7 per year, or lower)
** Release Frequency-Weighted
*** Linear-No Threshold, Population-Weighted

9.4 Sensitivity to Multiunit or Concurrent Accident Events (MELCOR)

These sensitivity calculations are intended to show the importance of the reactor building in the progression of accident in the SFP and the source term with a concurrent reactor accident. In the base calculation for a high-density, moderate leak scenario in OCP3 (see Figure 129), the fuel heats up and a zirconium fire is initiated. The reactor building refueling bay remains intact during the rapid draindown of the pool and there is very low hydrogen generation in the SFP. As the accident progresses, the atmosphere of the reactor building heats up as air is circulated through the assemblies. The oxidation of the SFP fuel depletes the oxygen in the reactor building and limits any long-term air oxidation and the associated exothermic power. Consequently, the long-term fuel heatup is limited primarily by decay heat, and the source term is relatively small (1.7-percent cesium release to the environment). The sensitivity calculations assume failure of the reactor building as a result of the hydrogen combustion caused by leakage from the containment (as evidenced from the SOARCA analysis and the Fukushima accident). The failure of the reactor building is based on the results of the PBAPS short-term SBO calculations for SOARCA (with and without reactor core isolation cooling (RCIC) blackstart). The reactor building failure times are at 8.5 hours (without RCIC blackstart) and 16.9 hours (with RCIC blackstart). It is further assumed that the failure of the reactor building and formation of debris in the pool results in a reduction of flow area at the exit of the assemblies (50 percent of nominal flow area) and increased flow losses.

Figure 130 and Figure 131 show the thermal response of the SFP with early (8.5 hours) and late (16.9 hours) failure of the reactor building. With early failure of the reactor building (before significant fuel heatup), the circulation of the cool air limits the fuel heatup and there is no release from the fuel. With the reactor building intact (Figure 129), the reactor building atmosphere keeps heating up, which limits the convective cooling of the assemblies. With late failure of the reactor building, the fuel becomes hot enough that a sudden increase in the flow of oxygen through the assemblies ignites and rapidly leads to significant air oxidation (zirconium fire). The fuel heats up leading to degradation and finally relocation (see Figure 131). This leads to 60-pecent cesium release to the environment. Finally in OCP4 (see Figure 132 and Figure 133), the decay heat and peak fuel temperatures are lower. The reactor building failure has no impact on the accident progression because the accident is not oxygen-limited. In fact, the reactor building failure (and thus lower temperature of circulating air) leads to lower fuel temperatures.

Table 56 compares the source term for low-density OCP1 for unmitigated small and medium leaks. In both cases, the loss of the reactor building, and thus the effectiveness of natural decontamination, leads to higher release by a factor of 2 to 4 depending on the radionuclide class.

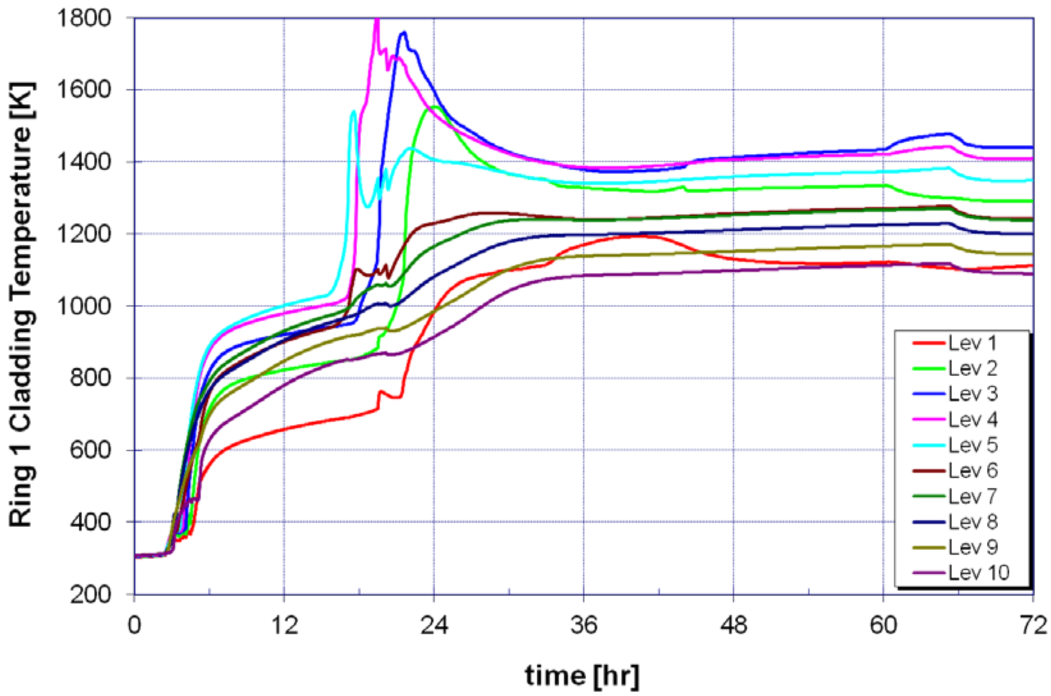


Figure 129 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP3)

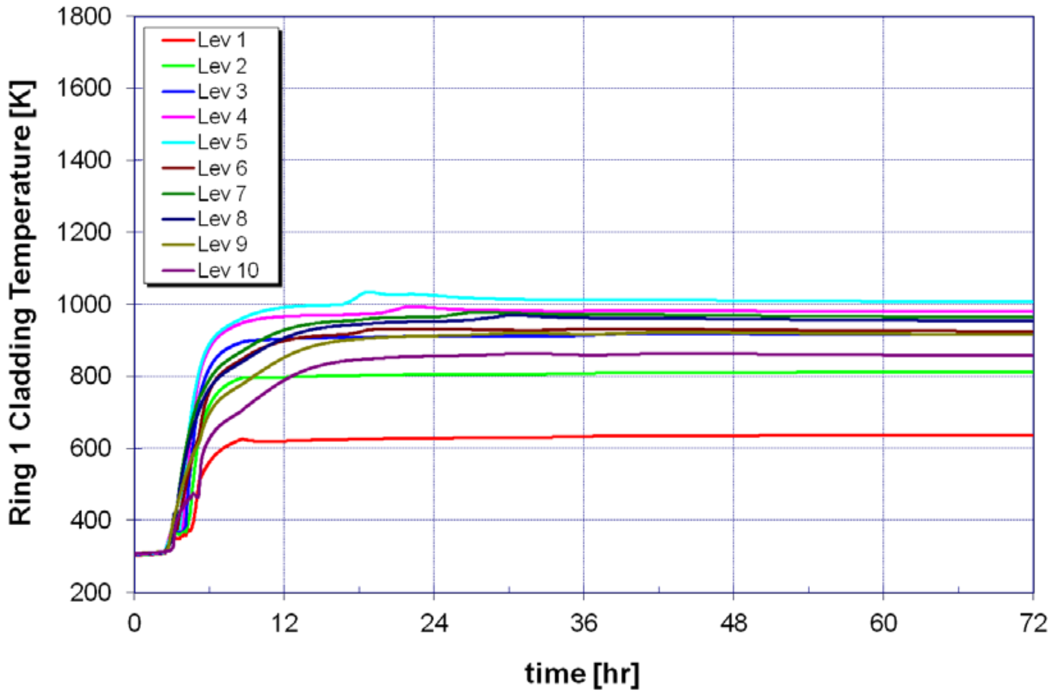


Figure 130 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP3; early reactor building failure)

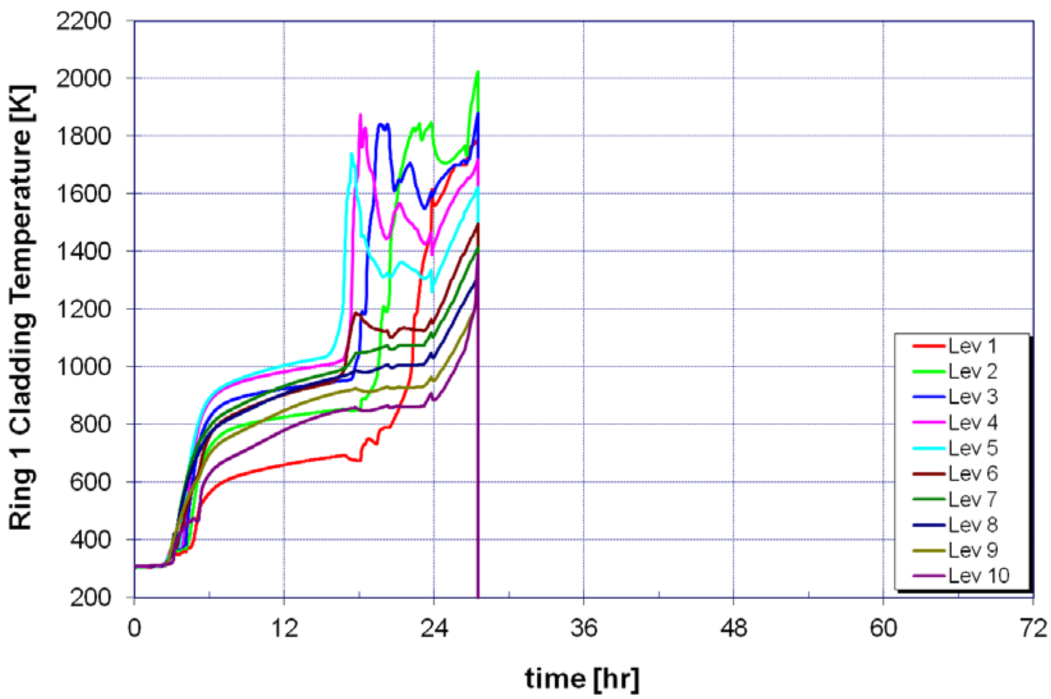


Figure 131 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP3; late reactor building failure)

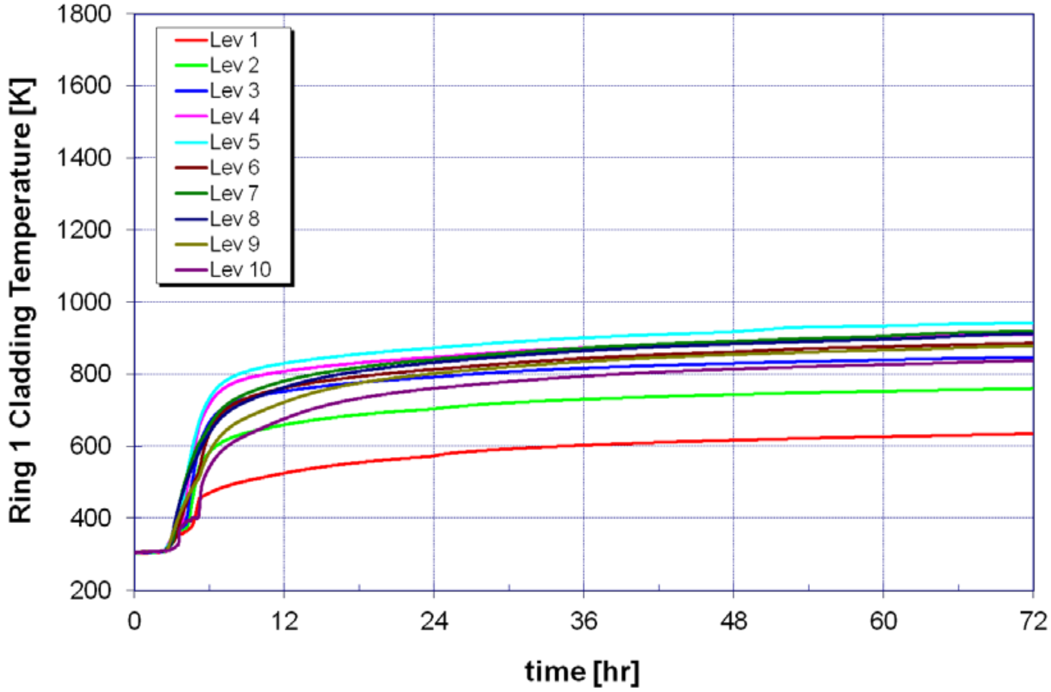


Figure 132 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP4)

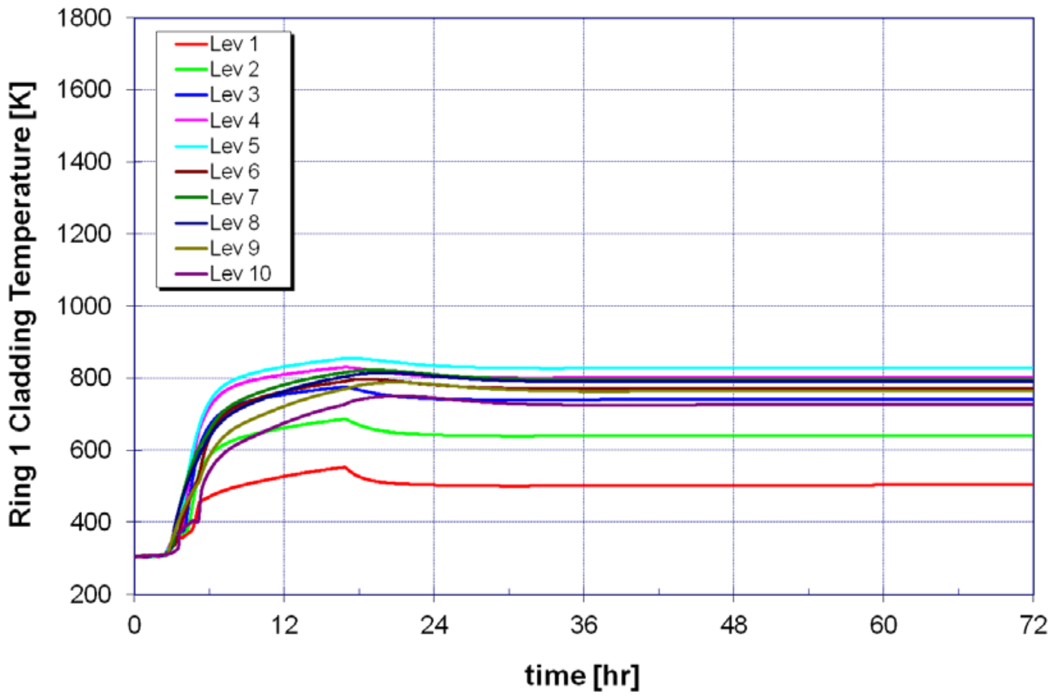


Figure 133 Ring 1 clad temperature for unmitigated high-density moderate leak (OCP4; late reactor building failure)

Table 56 Comparison of Low-Density OCP1 Release Fractions for a Concurrent Reactor Accident

Environmental release fraction	Small Leak		Moderate Leak	
	Base case	Late reactor building failure	Base case	Late reactor building failure
Xe/Kr	1.39E-01	1.85E-01	8.54E-02	8.87E-02
Cs	3.13E-02	1.18E-01	4.58E-03	1.49E-02
Ba	4.39E-03	9.66E-03	1.08E-03	4.45E-03
I	4.55E-02	1.41E-01	1.66E-02	5.60E-02
Te	4.54E-02	1.40E-01	1.68E-02	5.76E-02
Ru	2.17E-05	9.84E-05	2.09E-05	4.93E-05
Mo	8.86E-03	3.51E-02	2.60E-03	6.13E-03
Ce	1.49E-09	6.08E-09	4.94E-10	1.01E-09
La	1.34E-09	5.67E-09	4.37E-10	8.96E-10

9.5 Sensitivity to Molten Core-Concrete Interaction (MELCOR/MACCS2)

Accident Progression Analysis (MELCOR)

This sensitivity is a variation of the previous sensitivity calculation with late reactor building failure caused by a concurrent reactor accident. Even without MCCI, the SFP concrete floor starts to heat up and by the end of 3 days, a portion of the concrete experiences temperatures in excess of its ablation temperature (assumed to be 1500 K). Figure 134 shows the contours of temperature in the SFP floor. In the present sensitivity calculation, MCCI is assumed to be initiated in a control volume that becomes active once the floor liner melts and the debris contacts the concrete. Figure 135 shows the environmental release fraction of cesium. Without MCCI, the releases from the fuel are dominated by diffusion from the fuel matrix grain boundaries as modeled in the CORSOR-Booth model in MELCOR. The MCCI releases are modeled by the VANESA model in MELCOR which takes into account sparging of the concrete decomposition of gases and the presence of metal in the melt.[44] The release fraction of cesium is identical in both calculations (see Figure 16) until MCCI starts in Rings 1 and 2 at about 35 hours. MCCI results in a sudden increase in cesium release (and other fission products) at 35 hours and then again at 40 hours (start of MCCI in Rings 3 and 4) as soon as zirconium is added to the melt interacting with the concrete. In general, the release fractions with MCCI are higher, and for cerium and lanthanum groups, the MCCI releases are orders of magnitude higher.

[44] There are some limitations in representing MCCI in a SFP using MELCOR. The MELCOR MCCI model was developed to represent a pour of core debris from a failed reactor into a confined reactor cavity. In contrast, the relocation of fuel onto the SFP liner could be highly dispersed, especially in a favorable configuration. There could be regions of low-decay heat assemblies surrounding failed high-powered assemblies or open regions under the racks where only the high-powered assemblies relocated to the SFP liner. The MELCOR MCCI model immediately mixes all debris into a uniform debris bed with uniform temperature and decay heat power. Nevertheless, the MCCI sensitivity calculations illustrate the potential impact of MCCI physics on the radionuclide chemical form (and volatility) and the associated release of radionuclides to the environment. Certain radionuclide species can become more volatile in the presences of sparging ablation gases, which leads to the differences in Table 57.

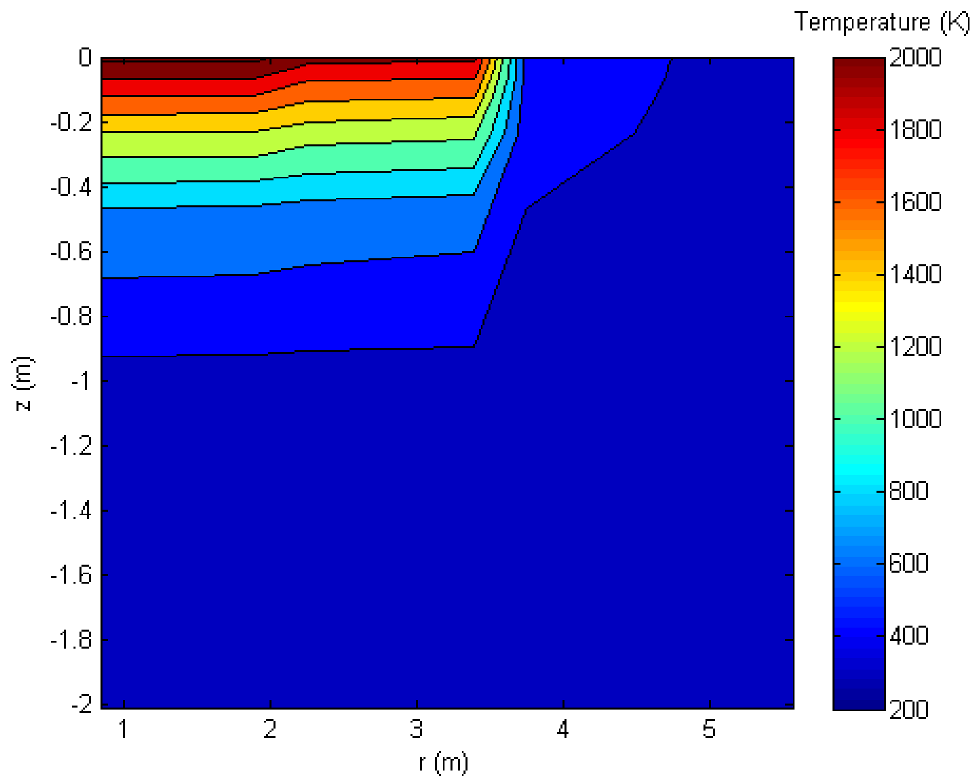


Figure 134 SFP concrete floor temperature for unmitigated high-density moderate leak (OCP3; late reactor building failure)

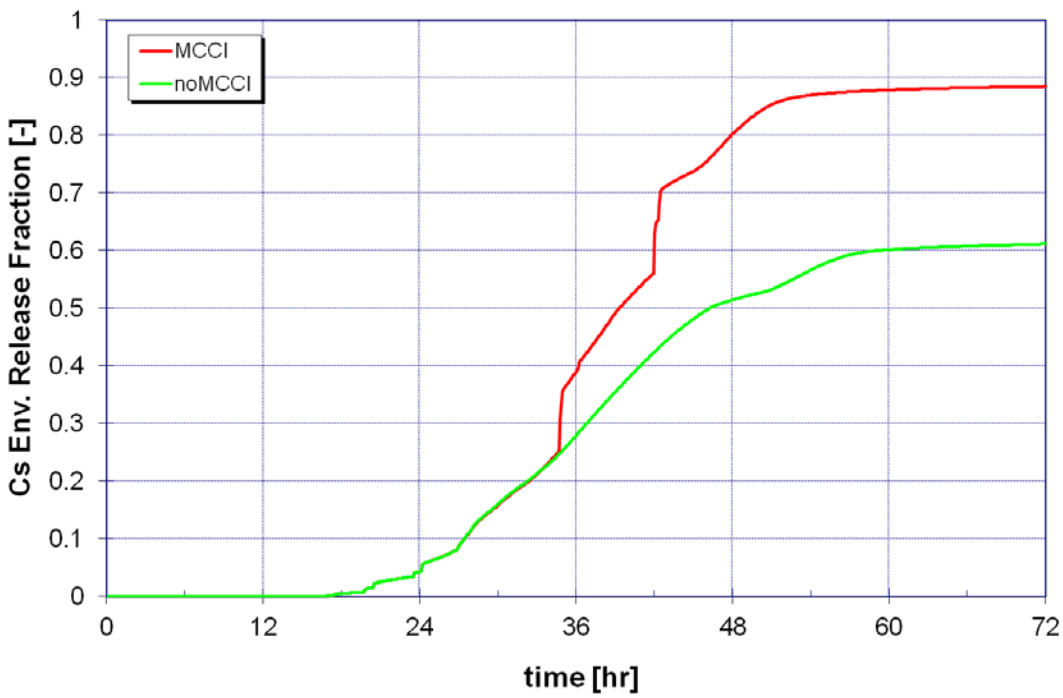


Figure 135 Cesium release fraction for unmitigated high-density moderate leak (OCP3; late reactor building failure) with and without MCCI

Table 57 Comparison of Release Fractions with and without MCCI.

Environmental release fraction	Without MCCI	With MCCI
Xe/Kr	0.92	0.92
Cs	0.61	0.88
Ba	0.01	0.07
I	0.83	0.91
Te	0.80	0.74
Ru	0.01	0.003
Mo	0.15	0.11
Ce	1.7E-07	0.007
La	1.6E-07	0.0002

Offsite Consequence Analysis (MACCS2)

The sequence used in the accident progression analysis was analyzed with MACCS2 to understand how MCCI affects offsite consequences. The sequence analyzed was the OCP3 moderate leak scenario, with hydrogen combustion in the refueling bay at 16.9 hours as predicted from the SOARCA short-term SBO with RCIC blackstart scenario.

The focus of this study was specifically on the SFP, and therefore, this sequence is not part of the main results. Rather, these are part of a different sensitivity investigating the effects of concurrent reactor events. Therefore both sequences with and without MCCI were calculated with MACCS2 in order to focus on MCCI. The individual consequence results of these sequences are not reported; however the effect of MCCI on the offsite results is shown in Table 58.

Table 58 Consequence Comparison – Molten Core Concrete Interaction

Molten Core Concrete Interaction Sensitivity (Weather-Averaged; OCP3 Moderate Leak sequence with reactor building failure at 16.9 hours)	
Type of Consequence	Conditional* Consequences
	Percent Increase
Individual Latent Cancer Fatality Risk** for 0-10 Miles	-17%
Collective Dose	9%
Land Interdiction (mi^2)	33%
Displaced Individuals (Persons)	17%

* Conditional on a release occurring (frequency of 1E-7 per year, or lower)
** Linear-No Threshold, Population-Weighted

No early fatalities are predicted for these sequences. These sequences have considerable release fractions, including increased contributions from some of the typically non-volatile chemical groups. However, these characteristics were not enough to reach the dose thresholds associated with early fatalities in OCP3, for which the last fuel offload has cooled for 37 days since shutdown.

Although the release is larger with MCCI, the individual LCF risk for 0-10 miles non-intuitively decreases. The reason for this is likely due to the significant level of protective actions and the type of radionuclides in the different source terms. The radionuclides in the MCCI sensitivity likely have different dose contributions relative to their LCF risk contribution, and therefore are more likely to cause protective actions despite having relatively lower risk factors, which in turn causes lower LCF risk.

To verify this phenomenon, the risk and dose contributions of different radionuclides to offsite consequence could be investigated. This is not done here; however, when farther distances are included (which are areas where protective actions and this phenomenon are less likely to occur), this reduction in the results no longer exists. This can be seen in the effect of MCCI on the collective dose in Table 58. Similarly, land interdiction and displaced individuals have higher consequences, as one may expect with relatively larger source terms.

9.6 Sensitivity to Radiative Heat Transfer (MELCOR)

For this sensitivity calculation, the surface area between Rings 2 and 4, Rings 4 and 6, and Rings 6 and 7 are modified by plus or minus 25 percent. Although Rings 2, 4 and 6 are empty, they still contain rack components and can impact the heatup in Rings 1, 3, and 5. Table 59 reports the results of the calculation for the unmitigated small leak (OCP2), low-density configuration. In general, the highest differences are observed for the reduction in the area (approximately 30-percent reduction in the release of volatiles), while increasing the surface area only has a modest effect on the release. The fuel heatup, radiation between rack components, initiation of air cooling, and interaction between different assemblies and racks are all complex phenomena that contribute to fission product release in a nonlinear fashion. Nevertheless, the releases for the low-density case are still small, and uncertainties in radiation modeling do not seem to significantly change the results

Table 59 Low-density OCP2 Release Fraction Sensitivity to Ring-Ring Radiation

	Base Case	-25% surface area	+25% surface area
Xe/Kr	4.41E-02	4.44E-02	4.40E-02
Cs	1.71E-02	1.23E-02	1.66E-02
Ba	5.19E-03	3.69E-03	5.06E-03
I	3.31E-02	2.50E-02	3.22E-02
Te	3.54E-02	2.53E-02	3.45E-02
Ru	9.27E-07	9.27E-07	8.75E-07
Mo	9.95E-05	9.93E-05	9.39E-05
Ce	3.54E-11	4.03E-11	3.16E-11
La	3.58E-11	4.06E-11	3.19E-11

9.7 Sensitivity to Land Contamination (MACCS2)

The measure of contaminated land area can vary significantly with the criterion used to measure or estimate the level of contamination. This study calculates the land that exceeds 500 mrem in the first year after the accident as an indicator for land contamination, based on the Pennsylvania 500 mrem annual dose limit for habitability. However, other protective action levels exist that can also be used as an indicator for measuring land contamination. These protective actions tend to be related to dose levels associated with either land interdiction or decontamination, but not necessarily.

Radioactivity levels are not typically used as a basis for protective actions. Instead, activity levels are usually measurements for estimating different dose levels, which are in turn used as the basis for protective actions. A range of typical activity levels for Cs-137 is included in the table below. These particular levels have been widely reported as the zoning criteria for the Chernobyl nuclear disaster.

The EPA intermediate phase PAG levels are 2 rem in the first year, and 500 mrem annually thereafter. Previous studies have typically used one criterion (4 rem in 5 years) to represent these PAG levels. How well this represents the actual EPA PAG levels was not analyzed here, although this criterion is included here.

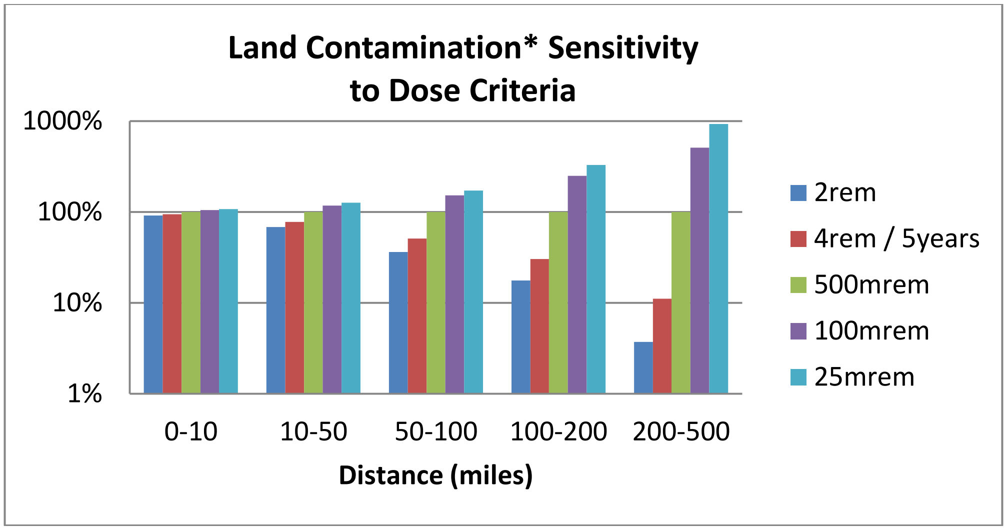
For simplicity, this sensitivity was based on the weather-average results of a single accident sequence. This is unlike the results of the production analyses, which are frequency-weighted averages of all the release sequences. The sequence chosen was the OCP3 small leak from a high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment, which has a cesium release fraction of 42% at 72 hours. Since the consequence results of individual sequences are not reported, the results of this sensitivity have been normalized.

Table 60 Consequence Comparison – Land Contamination Sensitivity

Total Land Area Sensitivity to Dose/Activity Criteria (Weather-Averaged; OCP3 Small Leak from high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment)		
Dose		
Protective Action Basis	Dose Level[1]	Land Area
EPA intermediate phase PAG[2]: 1st year	2 rem	21%
EPA intermediate phase PAGs[2] (as commonly represented in previous studies)	4 rem / 5 years	32%
Pennsylvania dose limit to the public	500 mrem	100%
ICRP recommendation[3]	100 mrem	361%
10CFR Part20 Subpart D		
10CFR Part20 Subpart E[4]	25 mrem	731%
Activity		
Protective Action Basis	Activity Level (Cs-137 Bq/m2)	Land Area
-	1.48E+06	73%
-	5.55E+05	181%
-	1.85E+05	346%
-	3.70E+04	557%

[1] Annual doses, unless otherwise noted
[2] EPA intermediate phase PAGs are: 2 rem in the first year, and 500mrem annually thereafter.
[3] ICRP recommends using the lower portion of a band that spans 1-20 mSv as a reference level for protective measures, and past experience demonstrates 1mSv is typical.
[4] 10CFR Part20 Subpart E also includes ALARA, which is not considered here.

As seen from the table above, different dose or activity levels can significantly change the amount of land area that exceeds a given limit. In addition to the total land area, a range of different distances were also analyzed in the graph below.



*Weather-averaged; OCP3 Small Leak from high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment

Figure 136 Land Contamination Sensitivity to Dose Criteria

At shorter distances, the change in the land area that exceeds a given dose limit is not significant, while at far distances, the change can be more than a factor of 10. The distance where the amount of land contamination becomes sensitive to different dose criteria is expected to depend on the initial concentration and the deposition rate. For this release magnitude, most of the plume exceeds all of the dose criteria at close distances. However, irrespective of the release magnitude, the affected area will increase and the concentrations will decrease as the plume spreads. Therefore, for all releases, land contamination is more dependent on the dose criteria at far distances.

9.8 Sensitivity to Time Truncation (MELCOR/MACCS2)

Project staff judged that a reasonable approach for the project is to consider radionuclide releases only if the fuel has become uncovered by 48 hours and to assume that any potential radiological release is stopped at 72 hours (Section 5.3). However, the use of a time truncation is uncertain, and is capable of significantly affecting the consequences. This assumption could be pessimistic since many resources are available at the State, regional, and national level that could be available to potentially truncate the accident more aggressively. On the other hand, this time truncation could be optimistic, as it assumes that an ongoing spent fuel pool release is capable of being truncated.

Given the uncertainty, this sensitivity considers the effects of both a more aggressive and a less aggressive time truncation. For simplicity, the sensitivity of longer time truncation was based on the weather-average results of a single accident sequence. This is unlike the results of the production analyses, which are frequency-weighted averages of all the release sequences. The sequence chosen was the OCP3 small leak from a high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment, which has a cesium release fraction of 42% at 72 hours. Since the consequence results of individual sequences are not reported, the results of this sensitivity are reported as the fractional increase in consequences over the original results. The sensitivity of a shorter time truncation discusses in which sequences releases would be averted.

A period of 96 hours was chosen to represent a less aggressive time truncation.   MELCOR and MACCS2 calculations were extended to 96 hours from the original 72 hours. The effect on the release fractions and the relative effect on offsite consequence can also be seen in the Figure 137 and Table 61 below.

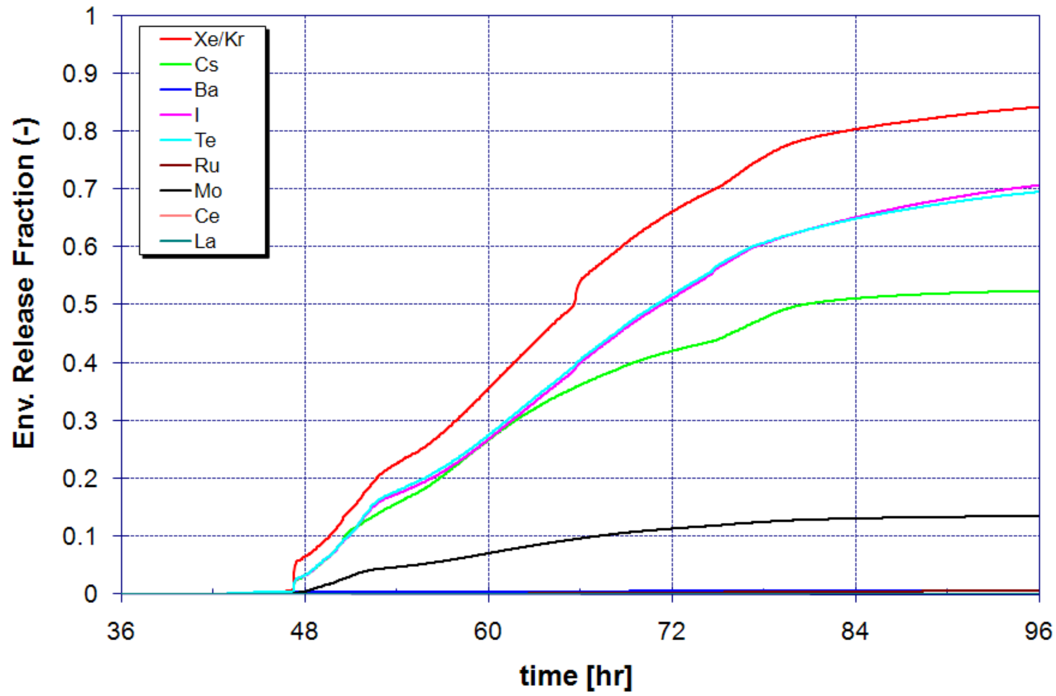


Figure 137 Atmospheric release fractions for unmitigated high density small leak (OCP3) with a 96 hour time truncation

Table 61 Consequence Comparison – Time Truncation Sensitivity

Time Truncation Sensitivity: 72 Hour vs. 96 Hour (Weather-Averaged; OCP3 Small Leak from high density SFP with unsuccessful deployment of 50.54(hh)(2) equipment)	
Type of Consequence	Conditional* Consequences
	Percent Increase
Individual Latent Cancer Fatality Risk** for 0-10 Miles	38%
Collective Dose (Person-Sv)	27%
Land Interdiction (mi^2)	28%
Displaced Individuals (Persons)	27%

* Conditional on a release occurring (frequency of 1E-7 per year, or lower)
** Linear-No Threshold, Population-Weighted

A shorter time truncation, however, can also significantly affect the results. For the sequences involving a small leak from a high density SFP during OCP1 and OCP2 with unsuccessful deployment of 50.54(hh)(2) equipment, fuel uncovery occurs around 40 and 43 hours (compared to a baseline time truncation of 48 hours). Using a time truncation less than 40 and 43 hours respectively, would avoid releases for these sequences. In other scenarios, fuel uncovery and release occur much sooner than the baseline time truncation.

These results highlight that some releases are expected to be prolonged and therefore a choice in a time truncation can affect offsite consequence predictions.

9.9 Sensitivity to Reactor Building Leakage (MELCOR)

Four sensitivity calculations were performed to examine the impact of the reactor building leakage on hydrogen combustion and accident progression. These covered the small leak scenarios in OCP2 and OCP3 without successful deployment of mitigation. Two larger leak sizes were considered, (1) an increase in the nominal leakage area by a factor of 10, and (2) an increase in the nominal leakage area corresponding to area of a blowout panel. In general, while an increase in area by a factor of 10 increases the leakage, any further increase in area has no effect since the building pressure adjusts to limit the leakage. The leakage area has no significant impact on accident progression, and since the hydrogen is produced over a relatively short time, the hydrogen mole fraction quickly reaches the 10% threshold for ignition. The Cs release fractions are not significantly altered. In OCP2, Cs release fraction is reduced by ~12% while it is increased by ~2% in OCP3 owing to slight variations in the course of the accident.

10. ASSESSMENT OF PREVIOUS STUDIES OF SAFETY CONSEQUENCES ASSOCIATED WITH LOADING, TRANSFER, AND LONG-TERM DRY STORAGE

10.1 Introduction

Staff has performed an assessment to 1) identify previous studies of safety consequences of spent fuel accidents in both wet and dry storage, 2) determine the extent to which those previous studies are comparable to results from the SFPS, and 3) to the extent practicable, update the results of the previous studies to facilitate a comparative assessment. The SFPS discusses off-site consequences of a spent fuel pool accident in Chapter 7, and provides limited discussion of several similar previous spent fuel pool studies. This Chapter provides a more detailed comparison between off-site consequences calculated for the SFPS and those calculated from previous studies. This Chapter also provides a comparative assessment of SFPS results against previous studies of the safety consequences associated with loading, transfer, and long-term storage in dry cask storage systems (DCSS). In these assessments, staff limited its focus to offsite consequences of accidental releases at commercial nuclear power plants. Specifically, these assessments compare the direct impacts due to offsite radiological exposure and the indirect (e.g., economic or land use) impacts of protective measures taken to avert offsite radiological exposure, of the various studies considered. Offsite impacts from routine operations, doses to workers from routine or accidental exposures, or non-safety related impacts such as costs of spent fuel management, were not considered. Furthermore, staff focused on studies associated with accidents, rather than studies of safety consequences associated with deliberate human actions such as sabotage or terrorism.

10.2 Previous Spent Fuel Pool Studies

There have been several previous studies of the consequences of spent fuel pool accidents. These include those in support of Generic Safety Issue 82 and of consequences from spent fuel pool accidents at shutdown nuclear power plants:

- "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82" (NUREG/CR-4982, 1987)
- "Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools," (NUREG/CR-5281, 1989)
- "Regulatory Analysis for the Resolution of Generic Issue 82 'Beyond Design Basis Accidents in Spent Fuel Pools'" (NUREG-1353, 1989)
- “A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants” (NUREG/CR-6451, 1997)
- “Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (NUREG-1738, 2001)

The studies conducted to evaluate beyond design basis accidents in spent fuel pools in the late 1980’s (NUREG/CR-4982, NUREG/CR-5281, and NUREG-1353) report a variety of impacts related to both radiological doses (e.g., collective doses), as well as the potential impacts associated with limiting radiological doses, such as costs and extent of land condemnation. NUREG-1353 reports collective doses within a 50 mile radius of 8 to 26 million person-rem per event based on MACCS calculations documented in NUREG/CR-5281. It also reports an interdiction area (“*area with such a high level of radiation that it is assumed that it cannot be decontaminated”*), based on CRAC2 calculations from NUREG/CR-4982, of 0 to 244 square miles (within a 50 mile radius) and offsite property damages of $3 to $26 billion in 1983 dollars (also within a 50 mile radius), based on MACCS calculations documented in NUREG/CR-5281. The more recent studies conducted to examine the risks from spent fuel pool accidents at shutdown nuclear power plants (NUREG/CR-6451 and NUREG-1738) also report a variety of both radiological and non-radiological impacts. NUREG-1738 reports radiological impacts of 0 to 200 early fatalities and potential latent cancer fatalities (out to 500 miles) in the hundreds of thousands. These studies used a variety of assumptions regarding pool inventory, release fraction, population density, and emergency response. The results of previous spent fuel pool studies are compared to the SFPS results for various consequence metrics in Table 62.

10.2.1 Quantitative Comparison of Spent Fuel Pool Analytical Results

The following table presents selected consequence results from previous studies of spent fuel pool accidents, and the SFPS.

Table 62 Comparison of consequence results from current and previous spent fuel pool analyses

Metric	NUREG/CR-4982, Table 4.7	NUREG/ CR-5281, Table 3.2	NUREG/ CR-6451, Tables 4.1/4.2	NUREG-1738[1] Tables 3.7-1/3.7-2	SFPS Results[2,3]
Early fatalities (0 to 500 miles)	Not reported	Not reported	0 to 101	0 to 200	0
Individual LCF risk within 10 miles (conditional)	Not reported	Not reported	Not reported	7.7e-4 to 8.2e-2	2.0e-4 to 4.4e-4
Collective dose within 50 miles in Person-Sv	11,000 to 26,000[6]	80,000 to 256,000	30,000 to 810,000	37,000 to 240,000	7,400 to 39,000
Collective dose within 500 miles in Person-Sv	710,000		40,000 to 3,400,000	450,000 to 600,000	27,000 to 350,000
Interdicted land (square miles)	Not reported	Not reported	Not reported	Not reported	170 to 9,400
Condemned land (square miles)	4 to 224[4,6,7]	Not reported	1 to 2,800	Not reported	<1 to 83

1. Results presented in Section 3.7 are taken from there; otherwise values are from Appendix 4, Note that the upper end of these values is generally driven by high ruthenium source terms with late evacuation.
2. 0 to 500 mi results are actually 0 to 1000 mi results, which is likely analogous to past study modeling assumptions; uniform pattern results are not included at this time; only LNT results are presented
3. The range of results is not bounding, as it does not represent ranges due to many uncertainties such as weather, operating cycle phase, or pool damage states. Direct consideration of these uncertainties would increase the range, as it likely would for the previous studies as well.
4. Note that the definition of interdicted land is not consistent with the definition used in the SFPS report. The text in NUREG/CR-4982 clarifies that what is reported is permanently uninhabitable land, which is analogous to condemned land.
5. These values use the annualized release frequency, combined with the conditional consequences, thereby over-estimating the average risk.
6. This range is for fire scenarios. For the non-fire scenario, the values were 4 person-rem and 0.0 sq. mi interdiction area.
7. Note that this metric does not change between cases 1A (50 miles) and 1C (500 miles), indicated that there is no additional condemned land beyond 50 miles in this analysis
8. The range at which this metric computed is not specified in NUREG/CR-5281

10.2.2 Comparison of SFPS Results to previous Spent Fuel Pool Studies

Comparison of SFPS results to past spent fuel pool studies is not straight-forward, because those studies reported a variety of consequence metrics and used a range of assumptions regarding pool inventory, release fraction, population density, and emergency response. These ranges present a variety of approaches to represent uncertainties from select input parameters, depending on the study. For instance, the range of NUREG-1738 results represents a range due to evacuation times, ruthenium release modeling, time since reactor shutdown, and two competing seismic hazard models. The range of the SFPS, on the other hand, represents a range due to uncertainty in deployment of mitigation equipment and variations of potential pool loading density. In addition, the SFPS results are expected to be sensitive to uncertainties in hydrogen combustion ignition criteria and the time truncation value (and these uncertainties are not reflected in the range of results), as well as uncertainties in weather, decay power, and pool damage states (which are not explicit in the range of results since average results are presented). It is also important to remember that past studies generally used generic assumptions intended to envelope the situation, as opposed to the focus on site-specificity with the SFPS. Nevertheless, these ranges of consequence metrics are often cited by external stakeholders, and thus comparison is informative.

A comparison of the release characteristics from previous spent fuel pool studies demonstrates that releases of cesium are generally less in the current study than in previous studies, and the time from accident initiation to release to the offsite environment is generally longer:

Table 63 Comparison of Source Terms from Current and Previous SFP analyses

Resolution of GI-82 (NUREG-1353, NUREG/CR-4982, NUREG/CR-5281):	NUREG-1738	SFPS (preliminary results):
• 10 to 100% Cs release (100% assumed for cases 1 and 2) • Release over 8 hours for a propagating SFP zirc fire (assumed) • 0.25 (BWR) or 1.0 (PWR) conditional probability if fuel becomes uncovered	• 75% Cs release (assumed from NUREG-1465) • Instantaneous draindown for large seismic • 2 to 14 hour heatup depending on fuel age (see Table 1A-1)	• Cs release = < 1% to 49% • Draindown to uncovery – 2.5 to 43 hours (when leak exists) • Start of release = 8 hours to > 72 hours

The lack of any early fatalities attributable to acute radiation exposure in this study is consistent with results of some past SFP studies, and much lower than others (e.g., up to 200 early fatalities from NUREG-1738). The range of latent fatalities predicted in this study is consistent with the lower end of the range reported in past SFP studies. The conditional individual latent cancer fatality risk from 0 to 10 miles for the scenarios studied in this report is several orders of magnitude below that reported in NUREG-1738, which was the only other study to report this metric. Even when the early evacuation scenario from NUREG-1738 is used for comparison (average individual risk is in the range of 2.6E-3 to 4.8E-3), the results from the current SFPS study are significantly lower. The collective dose values predicted in this study are consistent with the lower end of the range reported in past SFP studies. The SFPS reports temporarily interdicted land (uninhabitable land during the first year following the postulated accident), in order to remove uncertainty in longer-term effects and policies related to weathering and decontamination decisions. Reporting interdicted land makes the results incomparable to the past SFP studies which have presented condemned land. The SFPS does not report other aspects of offsite property damage.

10.3 Previous Dry Cask Storage Studies

The number of studies of the consequences from dry cask handling and storage accidents are more limited than those for spent fuel pools. Safety analysis reports for dry cask storage systems, submitted in support of applications or renewals for site-specific independent spent fuel storage installation (ISFSI) licenses or for DCSS certificates, include some information on offsite consequences of potential accidents (e.g., tornado missile impacts, earthquakes, floods). However, such accidents are generally shown by analysis not to result in a release, and the likelihood of more severe accidents is sufficiently low that the consequences need not be explicitly evaluated. Staff identified one previous NRC analysis on the offsite safety consequences of accidents from dry cask storage systems. The report, "A Pilot Probabilistic Risk Assessment of a Dry Storage System at a Nuclear Power Plant" (NUREG-1864, ML071340012), documents a pilot PRA for a specific dry cask system (Holtec International HI-STORM 100) at a specific boiling-water reactor (BWR) site. The study included an assessment of potential offsite consequences from the drop and failure of a cask. It provides estimates of the annual risk for one cask in terms of the individual probability of a latent cancer fatality within 16 km (10 miles) of the site, and also reports that there are no prompt fatalities. The assessment was performed using MACCS2 for a representative site and is described in detail in Appendix E to NUREG-1864. Site-specific data important to modeling a HI-STORM dry cask 30.5 meter (100-foot) drop accident scenario in the MACCS2 consequence calculation were collected and used. The important parameters/variables required to model the site are the population density/distribution and the site meteorology. The radionuclide inventory, source term (i.e., release fraction, release start time, and release duration), initial plume dimensions (related to the system geometry), and plume heat content were described. Other settings and models necessary for a MACCS2 calculation (e.g., food chain model) were taken from the NUREG-1150 study MACCS2 input file prepared for the Surry Power Station. The input file is documented in Appendix C to the MACCS2 code manual and is referred to there as Sample Problem-A.

10.3.1 Supplemental Analyses

In order to provide quantitative estimates of safety consequences for accidents during dry cask handling and storage that are directly comparable with the results of the SFPS, and to provide additional output metrics for comparison, staff performed limited MACCS2 supplemental dry cask storage analyses. These supplemental analyses used the source term characteristics from NUREG-1864 coupled with the site-specific characteristics reflected in the MACCS input decks used in the SFPS analyses. The analyses conducted in NUREG-1864 were conducted at a different geographic location than the site selected for the SFPS and evaluated impacts only in terms of selected human health metrics (the individual probability of a prompt fatality within 1.6 km (1 mile) and of latent cancer fatality within 10 miles, and the individual lifetime dose commitment). These metrics can be affected by site-specific characteristics such as meteorology and population distributions surrounding the site. To perform this analysis, staff modified the MACCS input files used for the analyses in the SFPS (described in detail in Chapter 7 of the SFPS) with the revisions discussed below.

Changes related to meteorology, site characteristics, and dosimetry:

No changes were made to the SFPS input deck related to meteorology, site demographic and economic characteristics, or dosimetry. Site data, including weather, population, and land values are therefore consistent with SFPS results. The dosimetry files used are consistent with FGR-13, whereas NUREG-1864 used the dose conversion factors used in NUREG-1150. This is a potential source of difference from the results reported in NUREG-1864.

Changes related to source term and release:

The radiological inventory was changed to be consistent with Table E.1 of NUREG-1864. In addition, a limited set of radionuclides present in the SFPS input deck that are expected to be in secular equilibrium with the nuclides listed in Table E.1 (Ba-137m, Pr-144, and Rh-106) were added with an activity equal to that of their parent radionuclide. However, a limited set of nuclides (Pm-147, Eu-154, Am-242m, Am-243, and Cm-243) reported in NUREG-1864 Table E.1 were not used in the SFPS MACCS2 input deck. Because the dosimetric data for these nuclides was not developed in the SFPS input deck, these radionuclides were not included in the modeled inventory for the supplemental analysis. Based on the much larger inventory of fission products such as Cs-137 and Sr-90, and of actinides such as Pu-241, the omission of these nuclides is not expected to significantly affect the results; however, this is a potential source of difference from the results reported in NUREG-1864. The number of chemical groups was changed to three to represent noble gases (NG), activation products (CRUD) and particulates (PART) to be consistent with the NUREG-1864 source term. Consistent with the NUREG-1864 source term, the only nuclide in the noble gas chemical group was Kr-85, and the only nuclide in the activation product chemical group was Co-60. For consistency with NUREG-1864, all other nuclides were assigned to the particulate group in view of the fact that releases from dry casks are likely to result from impacts at a sufficiently low temperature that radionuclides would be released by mechanical means rather than because of different volatilities. The inventory modeled in this supplemental analysis is provided below:

Table 64 Modeled Inventory for Supplemental Reanalysis

Nuclide	Bq	Chemical Group
Co-60	1.61E+14	CRUD
Kr-85	2.77E+15	NG
Sr-90	3.40E+16	PART
Y-90	3.40E+16	PART
Ru-106	2.92E+14	PART
Rh-106	2.92E+14	PART
Cs-134	5.13E+15	PART
Cs-137	5.54E+16	PART
Ba-137m	5.54E+16**	PART
Ce-144	5.08E+13	PART
Pr-144	5.08E+13**	PART
Pm-147	0* (*3.37E+15*)	PART
Eu-154	0* (*4.15E+15*)	PART
Pu-238	3.98E+15	PART
Pu-239	1.87E+14	PART
Pu-240	3.47E+14	PART

Pu-241	5.23E+16	PART
Am-241	1.20E+15	PART
Am-242m	0* (*1.97E+13*)	PART
Am-243	0* (*3.07E+13*)	PART
Cm-243	0* (*3.02E+13*)	PART
Cm-244	5.66E+15	PART

*These nuclides were not included in the supplemental analysis, as discussed above. The values from NUREG-1864 are provided in parentheses to allow comparison of source terms.
**These short-lived progeny were not in Table E.1 of NUREG-1864 but are included in the SFPS input deck. These were included in this table to represent the fact that these are likely to be in secular equilibrium with their parent radionuclides.

The particle size distribution assumed for NUREG-1864 was not identified in Appendix E. For purposes of this supplemental analysis, the particle size distribution for the particulate and activation product chemical group was assigned to be equal to the particle size distribution of the lanthanide chemical group in the SFPS, as lanthanides are presumed to be released due to mechanical measures rather than by volatility. Although this distribution is not the most appropriate for a dry cask storage scenario, using a cask-specific distribution would likely not change the conclusions of this Chapter. Specifically, the larger particle sizes expected to be associated with such a scenario would result in more deposition closer to the site, resulting in fewer exposed individuals within ten miles. The values used are given below:

Table 65 Particle Size Information

Particle Size Group	Particle Size Distribution	Dry Deposition Velocity (m/s)
1	3.2%	0.0011
2	15%	0.001
3	29%	0.0014
4	21%	0.0023
6	10%	0.0045
6	3.0%	0.0092
7	1.50%	0.0177
8	0.60%	0.0291
9	0.20%	0.0367
10	16%	0.0367

The release height and release fractions were varied to be consistent with NUREG-1864, Table E.1, as given in Table 66 below. To simulate the short duration release modeled in NUREG-1864, the number of plume segments was reduced to one with release starting at time zero, with a two minute (120 second) release duration. Reflecting the primarily mechanical rather than thermal nature of the release, the plume rise model was changed to a heat only option with a power of 18 kW to be consistent with NUREG-1864. However, parameters associated with building wake effects (e.g., building height, initial plume dimensions) were chosen to be consistent with SFPS values, as these values would be site specific. This represents another potential source of difference with NUREG-1864 values.

Changes related to emergency response and long-term protective actions

Consistent with the immediate release model and no evacuation assumption in NUREG-1864, the supplemental analysis eliminated all evacuating cohorts by changing the evacuation model to “No Evacuation”. However, sheltering and relocation parameters remained consistent with SFPS estimates. No changes were made to SFPS parameters for long-term protective actions such as decontamination levels and costs, as these were selected to be consistent with the SFPS site-specific values to allow for comparability. The application of SFPS emergency-phase sheltering, relocation, and long-term protective action parameters represent a source of difference between the results of NUREG-1864 and the supplemental analysis.

The results, and their comparability to the results provided in NUREG-1864, are provided in Table 66 and Table 67. Results are provided for a range of release fractions and release heights to facilitate comparison with the results reported in NUREG-1864. Staff considers the upper end of the release fraction for particulates in NUREG-1864 (0.12%) to represent a very conservative estimate of the potential respirable particulate release from a breached cask, as it assumes essentially complete fragmentation and entrainment of the high-burnup rim region and very limited filtration (10% released) within the cladding-fuel gap during entrainment flow. Reporting the full range of results, consistent with the results presented in Table E.1 of NUREG-1864, allows a more informed comparison of results including the effects of potential conservatisms in the analyses.

Results are reported for a variety of output metrics. These include both direct measures of health impacts (doses and probabilities of early and latent fatalities) as well as indirect measures such as the amount of land that is either temporarily interdicted or permanently condemned) or the numbers of temporarily or permanently displaced individuals.

Table 66 Parameters and Results from NUREG-1864, Table E.3

Release Fraction						
Noble Gases	Particles	CRUD	Release Height (m)	Ind. Risk of Prompt Fatality within 10 mi	Ind. Risk of LCF within 10 mi	Ind. Peak Dose at 1.2-1.6 km (Sv)
0.12	1.2E-03	1.5E-03	50	0	3.6E-04	1.85
0.12	1.2E-04	1.5E-04	50	0	5.2E-05	0.22
0.12	7.0E-06	1.5E-03	50	0	4.3E-06	2.6E-02
0.12	7.0E-07	1.5E-04	50	0	4.3E-07	2.7E-03
0.12	1.2E-03	1.5E-03	120	0	2.1E-04	0.14
0.12	7.0E-06	1.5E-03	120	0	2.6E-06	3.2E-03

Table 67 Supplemental Reanalysis with SFPS Input Deck

Release Fraction				Reanalysis with SFPS Input Deck						
NG	PART	CRUD	Release Height (m)	Prompt Fatality within 10 mi	Ind. Risk of LCF within 10 mi	Ind. Peak Dose at 1.2-1.6 km (Sv)	Collective Dose (0-50 mi) Person-Sv	Interdicted land in first year after accident (square miles)	Condemned land (square miles	Displaced Persons
0.12	1.2E-03	1.5E-03	50	0	7.1E-05	0.33	740	20	1.6E-03	5,800
0.12	1.2E-04	1.5E-04	50	0	8.9E-06	4.0E-02	86	1.4	0	150
0.12	7.E-06	1.5E-03	50	0	7.3E-07	6.6E-03	5.7	1.2E-02	0	1.8
0.12	7.E-07	1.5E-04	50	0	7.5E-08	6.8E-04	0.57	4.1E-06	0	-
0.12	1.2E-03	1.5E-03	120	0	5.1E-05	7.4E-02	780	24	3.9E-05	7,400
0.12	7.E-06	1.5E-03	120	0	7.0E-07	1.7E-03	6.2	3.2E-04	0	0.01

10.3.2 Quantitative Comparison of Dry Cask Storage and SFPS Analytical Results

The following table presents selected consequence results from the previous study of dry cask storage accidents, the supplemental dry cask storage study described above, and the SFPS.

Table 68 Comparison of consequence results from SFPS, NUREG-1864, and DCSS supplemental analyses

Metric	SFPS Results	NUREG-1864	DCSS Suppl. Analyses
Early fatalities (0 to 500 miles)	0	0	0
Individual LCF risk within 10 miles (conditional)	2.0e-4 to 4.4e-4	4.3e-7 to 3.6e-4	7.5e-8 to 7.1e-5
Collective dose within 50 miles in Person-Sv	7,400 to 39,000	Not reported	0.6 to 780
Collective dose within 500 miles in Person-Sv	27,000 to 350,000	Not reported	Not reported
Interdicted land (square miles)	170 to 9,400	Not reported	<<1 to 24
Condemned land (square miles)	<1 to 83	Not reported	<<1

10.3.3 Comparison of SFPS Results to Previous and Supplemental Cask Studies

Comparison of SFPS results to past dry cask studies is not straight-forward. This is because the type of information reported is different, the assumptions related to fuel and canister/cask damage are different, and the risks of dry cask handling, while low, are generally driven by design features that can vary significantly between different DCSS designs. For example, the NUREG-1864 study is based on a welded canister-in-overpack design, whereas the site selected for the SFPS study uses directly loaded bolted casks.

Nevertheless, meaningful comparisons can be made. An examination of the conditional individual latent cancer fatality probability metric demonstrates the effectiveness of emergency response and long-term protective actions at mitigating dose, consistent with the observations made in previous studies such as NUREG/CR-4982 and NUREG/CR-6451. The maximum consequences (in terms of latent cancer fatality probability) for both a pool accident and a dry cask accident, although involving substantially different amount of released material, are both limited to a range of 1E-4 to 1E-3 per event. The contrast to the much higher conditional consequence reported in NUREG-1738 (8.2E-2) is due to the assumption of a late evacuation coupled with a high source term in this study. The difference between impacts from pool and cask accidents is more clearly highlighted in measures related to the areal extent of contamination rather than in measures of peak individual risk. Inspection of Table 67 and Table 68 demonstrates that even in the case of very high release fractions from dry cask accidents, conditional results for metrics such as population dose or condemned or interdicted lands are several orders of magnitude lower than the low end of consequences of pool accidents. This comparison is significantly exaggerated if a less conservative estimate of the DCSS release fraction is used. The results suggest that a DCSS accident is unlikely to result in the need for extensive offsite protective action such as land interdiction or population displacement, in contrast to a pool accident that may require significant offsite protective action. Furthermore, for the risks (expressed as a frequency-weighted consequence) of a DCSS accident to be comparable to the risks of a pool accident, the frequency of a DCSS accident would have to be several orders of magnitude higher than that of a pool accident.

10.4 Summary of Assessment of Previous Studies

This assessment demonstrates that past SFP accident consequence estimates from large seismic events are similar to this study for most metrics. Comparison of this study to dry cask storage studies (NUREG-1864 and supplemental analyses from this Chapter), indicates that in some circumstances, the conditional individual LCF risk within 0 to 10 miles would be similar due primarily to the conservative upper bound estimate of the dry cask release as well as the expected effectiveness of protective actions in response to an SFP release. However, conditional results for metrics such as population dose or condemned or interdicted lands are several orders of magnitude lower for dry cask scenarios than the low end of consequences of pool accidents, due to the substantially smaller amount of released material.

11. REGULATORY ANALYSIS SCREENING SUMMARY

Based on past studies, the NRC has concluded that both spent fuel pools and dry casks provide adequate protection of public health and safety and the environment, and that the likelihood of an accident involving a radiological release from the spent fuel remains extremely small. While the staff believes that public health and safety is adequately protected for both spent fuel pool and dry cask storage, the Spent Fuel Pool Study (SFPS) provides one part of a technical analysis to confirm, using insights from Fukushima, that spent fuel pools continue to provide adequate protection. As indicated by its title, this study looks at the storage of spent nuclear fuel in spent fuel pools. The study also assesses whether any significant safety benefits (or detriments) would occur from expedited transfer of spent fuel to dry casks, and the potential costs associated with such expedited transfer.

The study establishes that both high and low density spent fuel pool arrangements at the reference plant provide reasonable assurance of adequate protection. The analysis in Appendix D, which is summarized here, assesses the benefits and costs of this action relative to the baseline of existing requirements, including current regulations and relevant orders.

11.1 <u>Decision Rationale</u>

11.1.1 Comparison to the Safety Goal Policy Statement

The Safety Goals for the Operation of Nuclear Power Plants: Policy Statement (51 FR 28044) (safety goal policy statement) was used to evaluate the impacts resulting from a severe spent fuel pool accident. The frequency of damage to the spent fuel pool is estimated to be approximately between 7.11×10^{-7} and 5.39×10^{-6} per year when considering all initiators that could challenge spent fuel pool cooling or integrity. This value, when compared to a target core damage frequency value of 1×10^{-4} per reactor-year in the Safety Goal Policy Statement, represents 0.71 to 5.39% percent of the overall frequency of core damage.

As described in Appendix D it is difficult to compare the estimated 7.11×10^{-7} to 5.39×10^{-6} per reactor-year release frequencies for the postulated spent fuel pool accident when considering all initiators to a target value of 1×10^{-5} per reactor year for a large early release frequency (LERF). The spent fuel pool source term is not similar to the core damage (or melt) source term. The consequences of a spent fuel pool accident are predicted to have no early fatalities and public health risk is dominated by latent cancer risks resulting from long-term exposures. Because the analyzed spent fuel accident is a slow progression with at least eight hours before an environmental release occurs and the resultant release is not expected to result in any offsite early fatalities, the analysis suggests that the spent fuel pool release does not fall within the definition of a large early release. Although this analyzed accident is different from a reactor accident, the spent fuel pool estimated release frequencies of 7.11×10^{-7} to 5.39×10^{-6} per reactor-year meet the 1×10^{-5} LERF guidelines.

Collective risk is based on the statistically expected number of early and latent cancer fatalities. The safety goal policy statement defines the early fatality area calculation as that within one mile from the site boundary. A ten-mile radius is defined for calculating latent cancer fatalities. The quantitative objective of the Policy Statement is for the risk to the population in the vicinity of a nuclear power plant from an accident at a nuclear power plant to not exceed 0.1 percent of the sum of cancer fatality risks resulting from all other causes. Based on recent data, the total fatality rate from cancer in the U.S. is 580,350 per 315,747,500 persons (http://www.census.gov/popclock/) or a risk of 1.84×10^{-3} per year, which results in a safety goal of 1.84×10^{-6} per year. Using the bounding frequency of damage to the spent fuel pool of 5.39×10^{-6} per year, which considers all initiators that could challenge spent fuel pool cooling or integrity, and the conditional individual latent cancer fatality risk within a 10-mile radius of 4.4×10^{-4} yields a latent cancer fatality risk of 2.37×10^{-9} per year. This calculated value of 2.37×10^{-9} latent cancer fatalities per reactor-year associated with a spent fuel pool accident represents a 0.13 percent fraction of the 1.84×10^{-6} per year societal risk goal.

Therefore, the risk and consequences of a spent fuel pool accident at the reference plant meet the Safety Goal Policy Statement public health objectives. They also meet the 1×10^{-5} per reactor-year LERF guideline. Therefore, the NRC concludes that a regulatory requirement for expedited transfer of spent fuel from the spent fuel pool to storage casks is not needed for the reference plant In order to meet the Safety Goals.

11.1.2 Cost-Benefit Analysis

The key findings of the analysis are as follows:

- Total Cost to the Reference Plant. The proposal to expeditiously move older spent fuel assemblies from pool storage to dry cask storage beginning in year 2014 to achieve and maintain a low-density loading in the pool within five years will result in an estimated present value cost of $46.81 million (using a 7-percent discount rate) and $42.10 million (using a 3-percent discount rate) over the next 26 years. The earlier upfront and incremental dry storage cask capital and loading costs dominated these incremental costs. The reference plant routine occupational health costs will result in an estimated present value cost of $27,000 (using a 7 percent discount rate) and $24,000 (using a 3-percent discount rate). Sensitivity analyses result in an estimated present value cost that ranged from $16.4 million to $46.9 million.

- Value of Benefits to the Reference Plant. The benefits for expeditious movement of spent fuel to dry cask storage will result in an estimated present value benefit of $493,000 (using a 7-percent discount rate) and $711,000 (using a 3-percent discount rate). These benefits result from the monetized value for averted public and occupational radiation exposure and averted onsite impacts and offsite property damage. Sensitivity analyses result in an estimated present value benefit that ranged from $0.5 million to $27.7 million.

- Costs to NRC. The NRC costs to require the expeditious movement of spent fuel to dry cask storage were conservatively ignored to calculate the maximum potential benefit. Even though the NRC is not expected to incur substantial implementation or annual costs for this alternative, these costs would further reduce the calculated net benefit for the proposed expeditious movement of older spent fuel assemblies from pool storage to dry cask storage for the reference plant.

There are uncertainties in estimating the frequency of events for natural phenomena that are postulated to challenge spent fuel pool cooling or integrity. There are also uncertainties in the calculation of event consequences in terms of the dispersion and disposition of radioactive material into the site environs. This is due in part to uncertainties regarding the degree to which topographical features and other phenomena are modeled at distances away from the reference plant. Estimating economic consequences also includes large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies and the loss of infrastructure on the general U.S. economy. An example of this is the supply chain disruptions that followed the 2011 Tohoku earthquake and subsequent tsunami in Japan or the 2004 Indian Ocean earthquake and tsunami in Thailand.

The NRC recognizes that there are also costs and risks associated with the handling and movement of spent fuel casks in the reactor building. These impacts, if included in this analysis, would further reduce the overall net benefit in relation to the regulatory baseline. These effects (e.g., the added risks of handling and moving casks) were conservatively ignored in order to calculate the maximum potential benefit by only comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and its implementation costs without consideration of cask movement risk.

The release of fission products to the environment resulting from other events that cause the loss of spent fuel pool cooling or integrity (i.e., missiles, heavy load drops, loss of cooling or make-up, inadvertent drainage or siphoning and pneumatic seal failures) are estimated to occur approximately once in 2.7 million years or 3.7×10^{-7} per reactor-year. Operator diagnosis and recovery are important factors considered in the development of the event frequencies for these events and portions of this evaluation are premised on licensees having taken appropriate actions to understand the potential consequences of spent fuel pool accident events and develop appropriate procedures and mitigating strategies to respond and mitigate the consequences.

In section 9.2 of the SFPS, a sensitivity analysis is provided in which a more favorable fuel pattern is considered in which eight cold assemblies surround each hot assembly (i.e., 1x8 fuel assembly pattern). Although only a few sensitivity analyses were performed using this configuration, the results looked promising for inhibiting spent fuel pool releases. The sensitivity calculations for the high-density 1x8 fuel pattern showed a shorter time to air coolability (i.e. no releases in OCP3). Even for the cases that led to the release of radioactive materials in OCP2, the release magnitude was much smaller than for the 1x4 fuel pattern, and comparable to the low density cases. The fuel thermal response has a slower heatup when compared to a fuel pattern in which four cold assemblies surround each hot assembly (i.e., 1x4 fuel assembly pattern) because there is more mass to absorb heat. Furthermore, the loading configuration may result in similar reductions in risk to the low-density storage option evaluated without the significant capital costs for implementation. Further evaluation of this alternative and possibly other loading configurations for all operating cycle phases is recommended as part of the regulatory analysis for expedited fuel movement as part of the program plan described in SECY-12-0095 to evaluate the transfer of spent fuel to dry cask storage.

Sensitivity analyses that extend the analyses beyond 50 miles at the reference plant show that the low-density spent fuel storage alternative was not cost-justified for any of the discounted sensitivity cases, and was only marginally justified if discounting was not applied. Therefore, the expedited transfer of spent fuel from pools to dry cask storage containers at the reference plant does not meet the cost-justified substantial safety enhancement criterion.

11.2 <u>Further Actions</u>

The NRC plans to use the insights from this study along with other analyses to inform a broader regulatory analysis, which will help decisionmakers determine whether operating or future nuclear power reactor licensees should be required to maintain a low-density configuration in their spent fuel pools.

The analysis for the reference plant and the longer-term generic regulatory analysis address the questions of what can go wrong; how likely is it; and what are the consequences. Although this approach is well established at the NRC and other government agencies, it is often difficult to explain following rare disasters such as the accident at Fukushima Dai-ichi, or in presenting the results of studies such as this one. It is not enough to look at only the estimates of the low probabilities for failing spent fuel pools or only at the worst-case consequences in the unlikely event of failures of spent fuel pool integrity and existing mitigating systems. One needs to look at the totality of information presented in this report, previous studies, operating experience, and assess both the potential advantages and disadvantages of regulatory actions regarding the movement of spent fuel from storage pools to dry cask storage containers.

12. SUMMARY AND CONCLUSIONS

12.1 Summary

This study sought to investigate the relative consequences between low and high-density loading situations for a selected site following a seismic event greater than the maximum earthquake reasonably expected to occur at the reference plant location. The NRC expects that the ground motion used in this study is more challenging for the spent fuel pool structure than that experienced at the Fukushima Daiichi nuclear power plant from the earthquake that occurred off the coast of Japan on March 11, 2011. That earthquake did not result in any spent fuel pool leaks. Chapter 1 discussed some of the considerations that are raised by stakeholders with respect to these differences. These are re-visited here to set the stage for presenting the study's findings.

- Expedited movement of fuel from the SFP 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)
OCP1	54	17	0.31
OCP2	59	22	0.37
OCP3	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 health effects, potential land contamination, and economic impacts

  This point is covered in the findings below.

- Removal of older fuel slightly 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)
OCP1	2,951	2,526	0.86
OCP2	3,567	3,143	0.88
OCP3	2,571	2,149	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.

<u>The results of the study are as follows:</u>

1. A beyond design basis event with a frequency of occurrence of 1 in 60,000 per year was used in this study, and more likely earthquakes are not expected to challenge the SFP structure.

2. Past studies have indicated that large seismic events could lead to the loss of structural integrity of the spent fuel pool liner. This study's results confirm that such a condition is unlikely. For the low probability seismic event described above, the study estimated a conditional probability of failure of 0.1. The specific conditions under which a failure might occur are site-specific.

3. NUREG-1353 (1989) predicted generic seismically-induced SFP liner failure likelihoods of 2×10^{-6} to 6×10^{-6} per year, generally associated with events greater than 0.5g peak ground acceleration. NUREG-1738 (2001) predicted generic seismically-induced SFP liner failure likelihoods of 2×10^{-7} to 2×10^{-6} per year, generally associated with events around 1.2 g. The current study looks at a seismic event in the range of 0.5 to 1 g, and estimates a site-specific SFP liner failure likelihood of 2×10^{-6} per year (based on the informed expectation that this seismic range has the greatest contribution to frequency-weighted consequences). Since the updated *initiating event* frequency estimate (based on the 2008 U.S. Geological Survey model) for the reference plant for events greater than 1 g is 6×10^{-6} per year, this portion of the seismic hazard (i.e., > 1 g) may contribute more significantly to the overall frequency-weighted consequences for the reference plant than previously anticipated, depending on the conditional structural SFP liner failure probability associated with these larger events. The effect of this scope limitation may be offset by potential conservatisms in the structural analysis described in Section 4 of this report.

4. In this study, no set of conditions short of a liner failure led to a radiological release in less than 3 days, which is consistent with past studies. In most cases, the available time to prevent a radiological release was much greater than 3 days.

5. In this study, without mitigative action, fuel is estimated to be air coolable for at least 72 hours for all but roughly 10% of the operating cycle[45]. Past studies estimated this time to be a greater fraction of the operating cycle, when hotter fuel was contiguously stored. In other words, use of the 1x4 pattern has a positive effect in promoting natural circulation air coolability and reducing the likelihood of a release should the SFP become completely drained. An even shorter time was predicted for the 1x8 pattern currently employed at PBAPS. While variability in SFP loading configurations was not a focus of this study, this report consistently shows the advantages associated with dispersed fuel loading patterns.

6. In the cases studied, which in general did not account for multiple or concurrent reactor and SFP accidents, the precise time to diagnose the need for SFP mitigation did not have an effect on the course of most scenarios.

   Nevertheless, the improved reliable and available SFP indication required by the NRC Order of March 12, 2012 (EA-12-051) is important to ensure that plant personnel can effectively prioritize emergency actions. The availability of such instrumentation may have changed the mitigation mode (makeup versus sprays) deployed to mitigate events that resulted in a release.

[45] The actual time is between 37 days (not air coolable) and 107 days (air coolable), with 60 days representing the demarcation point between these two Operating Cycle Phases. The citation of 60 days as a representative value is reasonable based on other separate effects analyses not documented in this report. The actual time to air coolability could be more or less, depending on specific conditions.

7. This study considered variations in both pool loading and the effective deployment (or lack thereof) of 10 CFR 50.54(hh)(2) mitigation capabilities (i.e., water makeup or spray using portable equipment). Of these, effective deployment of mitigation had the largest impact on preventing a release of radioactive material, reducing the release frequency by a factor of about twenty (from 1×10^{-7}/yr to 6×10^{-9}/yr).

    Note that ongoing regulatory actions under Order EA-12-049 dated March 12, 2012 (and related correction dated March 13, 2012) increase the capability of operating nuclear power plants to mitigate beyond-design-basis external events, such as the seismic event studied here.

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

    - In terms of the likelihood of release within 3 days, no difference was seen.
    - In terms of consequences, the low density cases resulted in a smaller release due to the smaller inventory of radioactive material and the lower potential for hydrogen combustion. For high-density loading, the rapid draindown cases in general had smaller releases mainly because the reactor building remained intact (hydrogen combustions not predicted). For slow draindown events, longer times are available for deployment of mitigation. Without successful deployment of mitigation, the releases could be up to two orders of magnitude larger (these cases are associated with hydrogen combustion events).

9. For all scenarios, no offsite early fatalities attributable to acute radiation exposure are predicted to occur. Due to radioactive decay, spent fuel pools tend to have significantly less shorter-lived radionuclides (e.g. I-131) than reactors. Partly because of this, the release is not predicted to be fast and large enough to significantly exceed offsite dose levels necessary to induce early fatalities. . When necessary, emergency response as treated in this study effectively prevents early fatalities from acute radiation exposure.

10. In both high and low density loading without successful deployment of mitigation, the individual latent cancer fatality risk within 10 miles for the studied scenarios is predicted to be on the order of 10^{-10} to 10^{-11} per year, based on the linear no threshold dose response model. While this risk is scenario-specific and related to a single spent fuel pool, it is several orders of magnitude lower than the 2e-6 per year individual latent cancer fatality risk corresponding to the quantitative health objective for latent cancer fatalities and therefore unlikely to contribute significantly to a risk that would challenge the Commission's safety goal policy (NRC 1986). In addition, there is uncertainty in the risk calculations because it is dominated by low doses. As a perspective on uncertainty, excluding the uncertain effects of low doses significantly reduced the quantified individual latent cancer fatality risk within 10 miles. Average individual latent cancer fatality risk is low because of low release frequencies and the expected protective actions.

11. Average individual latent cancer fatality risk is low and decreases slowly as a function of distance from the plant. For scenarios with large releases, significant collective doses are estimated; however, risk of cancer fatalities from these doses would be a small fraction of the risk of cancer fatalities from all causes. Additionally, these individual risks are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough to be considered habitable. In comparing pool configurations,

collective dose (and latent cancer fatalities) for the studied scenarios could be an order of magnitude higher for the high density loading situation as compared to the low density loading situation

12. The amount of land interdiction for the studied scenarios could be up to two orders of magnitude greater for certain high density loading situations as compared to the low density loading situations. Also, like releases in the low density loading situation, successfully deployed mitigation in the high density loading situation is predicted to reduce the amount of land interdiction to a similar extent. For both situations, the major difference is driven by hydrogen combustion events and associated large releases, which are only predicted to occur in scenarios with unsuccessful deployment of mitigation.

13. While the likelihood of release is very low, offsite protective measures in the form of population relocation and land interdication may be extensive. High-density loading releases without 10 CFR 50.54(hh)(2) mitigation measures are calculated to result in release frequency-weighted land interdiction values of 0.001 mi^2 per year and 0.5 displaced individuals per year which are arrived at by multiplying the estimated frequency and the estimated consequence. While the amount of land interdiction can be large, the fraction expected to be permanently interdicted is small if a release were to occur. For low-density loading or with successful deployment of 10 CFR 50.54(hh)(2) mitigation measures, considerably less land interdiction and displaced individuals are predicted.

14. A comparison of the risks of different fuel handling strategies, such as current practice and expedited transfer, depends on several factors including the relative, site-specific risks, and the time spent in each stage of spent fuel storage. Other risks, such as the risk from cask drop events damaging fuel in the cask or the SFP, may at least partially offset the benefit of lower spent fuel pool risk from low density loading.

15. The human reliability study shows that in most situations SFP mitigation can be deployed in time to prevent release given the assumptions that sufficient plant staff and equipment is available for SFP mitigation and the work area is accessible to perform mitigation. There are two exceptions where mitigation will be ineffective under the moderate leak scenarios: (1) the earthquake occurs at the beginning of a refueling outage when the spent fuel is too hot for the assumed mitigation; and (2) the earthquake occurs when spent fuel is relatively hot and the reactor and spent fuel pool are hydraulically disconnected resulting in insufficient time to deploy mitigation and natural cooling mechanisms cannot prevent fuel damage. This study identified that possible improvements in mitigation flow and nozzle placement in low-dose locations could improve mitigation success likelihood, but this would require further verification.

16. This study demonstrates that past SFP risk estimates from large seismic events are similar to this study for most consequence metrics (see Chapter 10). Comparison of this study to dry cask storage studies (NUREG-1864 and supplemental analyses from Chapter 10) indicates that in some circumstances, the conditional individual LCF risk within 10 miles would be similar due primarily to the conservative upper bound estimate of the dry cask release as well as the expected effectiveness of protective actions in response to an SFP release. However, conditional results for metrics such as temporary or permanently interdicted land or population dose are several orders of magnitude

lower for dry cask scenarios than the low end of consequences of pool accidents, due to the substantially smaller amount of released material.

17. Applying the NRC's regulatory analysis guidelines to analyse the results of the SFP Study indicates that requiring the low-density spent fuel pool storage alternative is not justified for the reference plant. This conclusion is subject to the analysis model, data, inputs and assumptions in Section D.3 of Appendix D. The risk due to beyond design basis accidents in the spent fuel pool analyzed in this study is sufficiently low that the added costs involved with expediting the movement of spent fuel from the pool to achieve the low-density fuel pool storage alternative are not warranted. Sensitivity analyses that extend the analyses beyond the primary area considered also show that the low-density spent fuel storage alternative was not cost justified for any of the discounted sensitivity cases.

12.2 Conclusions

In conclusion, past SFP risk studies have shown that storage of spent fuel in a high-density configuration is safe and risk is low. This study is consistent with earlier research conclusions that spent fuel pools are robust structures that are likely to withstand severe earthquakes without leaking. The study estimated that the likelihood of a radiological release from the spent fuel pool resulting from the selected severe seismic event analyzed in this study is on the order of one time in 10 million years or lower. The factors leading to this low likelihood, as discussed in Section 12.1, are summarized in Figure 138.

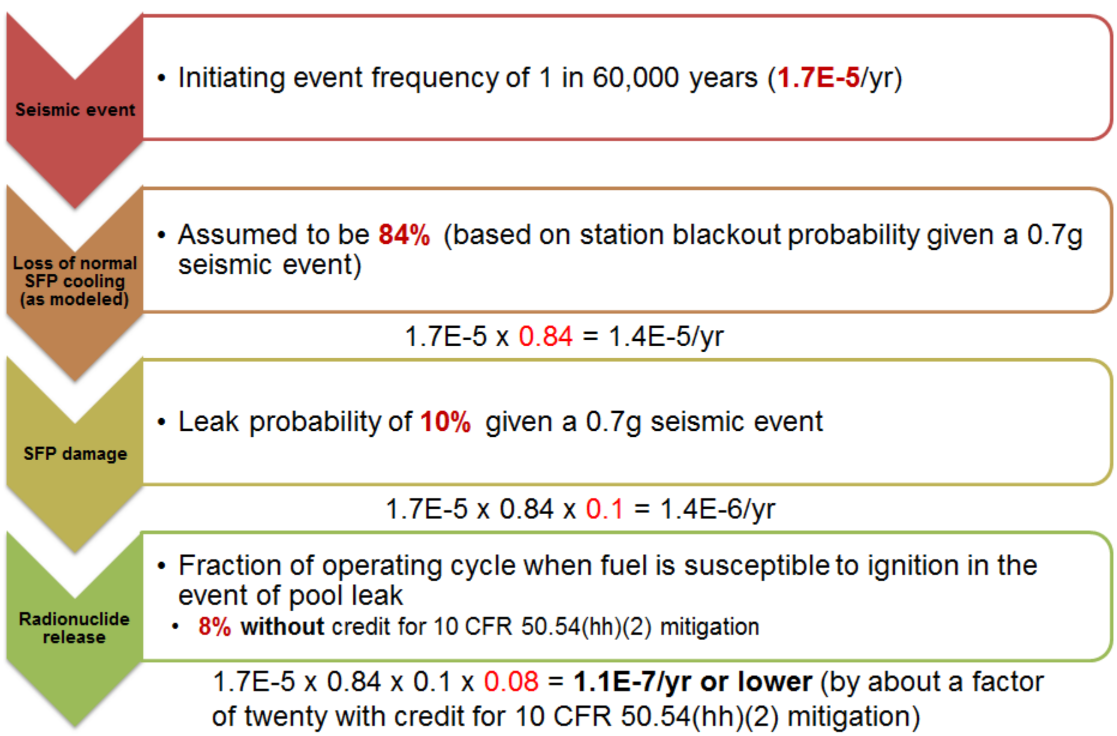


Figure 138: Affecting Likelihood of SFP Release from a Severe Seismic Event Event

For the hypothetical releases studied, no early fatalities attributable to acute radiation exposure were predicted and individual latent cancer fatality risks are projected to be low, but extensive protective actions may be needed. Comparisons of the calculated individual latent cancer fatality (LCF) risk within 10 miles to the NRC Safety Goal are provided in Figure 139 to give context that may help the reader to understand the contribution to cancer risks from the accident scenarios that were studied. The NRC Safety Goal for latent cancer fatality risk from nuclear power plant operation (i.e., 2×10^{-6} or two in one million per year) is set 1,000 times lower than the sum of cancer fatality risks resulting from all other causes (i.e., ~2×10^{-3} or two in one thousand per year). Comparing the study results to the NRC Safety Goal does involve important limitations. First, the safety goal is intended to encompass all accident scenarios on a nuclear power plant site, including both reactors and spent fuel. This study does not examine all scenarios that would need to be considered in a probabilistic risk assessment for a spent fuel pool, although seismic contributors are considered the most important contributors to spent fuel pool risk. Also, this study represents a mix of limited probabilistic considerations with a deterministic treatment of mitigating features. All analytical techniques, both deterministic and probabilistic, have inherent limitations of scope and method and also have uncertainty of varying degrees and types. As a result, comparison of the scenario-specific calculated individual LCF risk to the NRC Safety Goal is incomplete. However, it is intended to show how multiple spent fuel pool scenarios' risk results in the one in a trillion (10^{-12}) to one in 10 billion (10^{-10}) per year LCF range) are low. While the results of this study are scenario-specific and related to a single spent fuel pool, staff concludes that since these risks are several orders of magnitude smaller than the 2×10^{-6} (two in one million) individual LCF risk that corresponds to the safety goal for latent cancer fatalities, it is unlikely that the results here would contribute significantly to a risk that would challenge the Commission's safety goal policy (NRC, 1986).

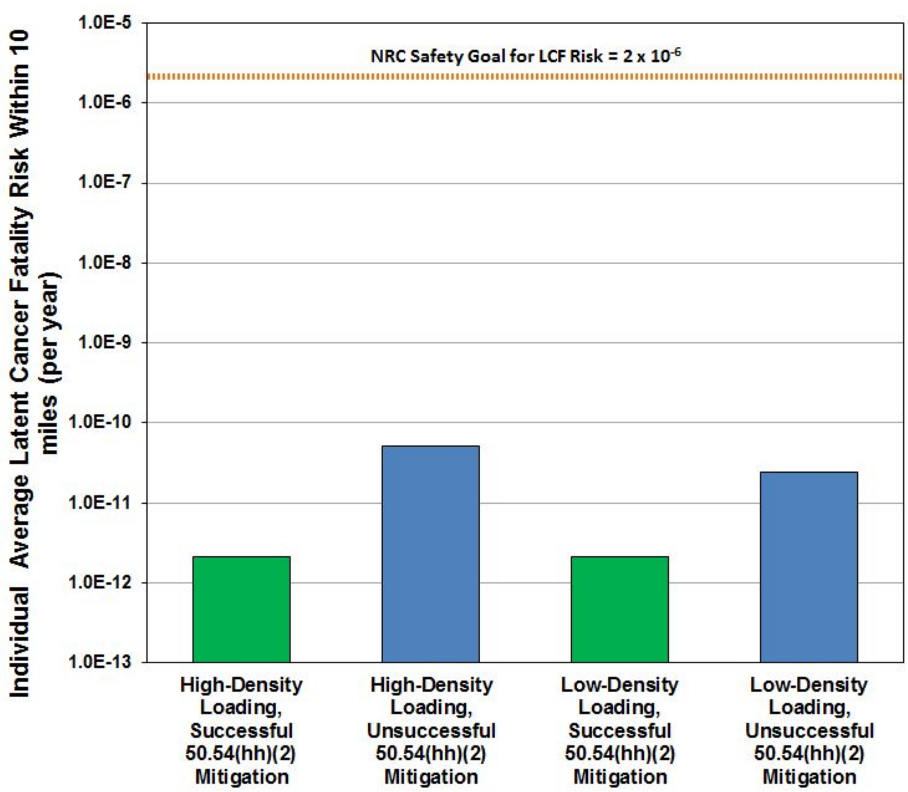


Figure 139: **Comparison of Population-Weighted Average Individual Latent Cancer Fatality Risk Results for this Study to the NRC Safety Goal (plotted on logarithmic scale)**

The study results demonstrated that in a high-density loading configuration, a more favorable fuel pattern or successful mitigation generally prevented or reduced the size of potential releases. Low-density loading reduced the size of potential releases but did not affect the likelihood of a release. When a release is predicted to occur, individual early and latent fatality risks for individuals within 10 miles do not vary significantly between the scenarios studied because protective actions, including relocation of the public and land interdiction, were modeled to be effective in limiting exposure. The beneficial effects in the reduction of offsite consequences between a high-density loading scenario and a low-density loading scenario are primarily associated with the reduction in the potential extent of land contamination and associated protective actions.

The results of the SFP Study show that the overall level of safety with respect to spent fuel storage in a spent fuel pool currently achieved at the reference plant is high and that the level of risk at the reference plant is very low. Applying the NRC's regulatory analysis guidelines to analyze the results of the SFP Study with respect to the quantitative benefits attributable to expedited transfer of spent fuel at the reference plant, and the risk reduction attributable to expedited transfer against the NRC's Safety Goals, the NRC concludes the incremental safety (including risk) reduction associated with expedited transfer of spent fuel at the reference plant is not warranted in light of the added costs involved with expediting the movement of spent fuel from the pool to achieve low-density fuel pool storage. Therefore, an NRC requirement mandating expedited transfer of spent fuel from pools to dry cask storage containers at the reference plant does not appear to be justified. The NRC plans to use the insights from this analysis to inform a broader regulatory analysis of the spent fuel pools at US nuclear reactors.

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APPENDIX A: DETAILED EMERGENCY RESPONSE MODELS

The detailed evacuation timing and speeds for each cohort developed using the information and approach described in Section 7.1.4 are described in this appendix. 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, which are the beginning, middle, and late phases. Average evacuation speeds were derived from the reference plant's ETE report. Speed adjustment factors are 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. The remainder of the evacuation, following period defined by DURMID, is the late phase.

A.1 Evacuation Model 1: WinMACCS response parameters for sequences where PAGs are not exceeded beyond the EPZ.

The following cohorts were established for this evacuation model:

0 to 10 Miles, Early Evacuees: This population begins to evacuate before receiving an evacuation order. Focus group work conducted to support NUREG/CR-6953, Volume 2 (NRC, 2008c) 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 percent of residents of EPZs have packed a "go-bag" and are ready to leave. Because the accident is initiated by a severe earthquake, it is assumed 30 percent of the public evacuate.

10 to 20 Miles, Shadow: These residents evacuate from areas that are not under an official evacuation order. 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, a uniform fraction of the population is assumed to evacuate within the 10- to 20-mile region. Because the accident is initiated by a severe earthquake, it is assumed 30 percent of the public evacuate. This cohort will begin evacuating as they hear of the evacuation orders and observe EPZ evacuees traveling through the area.

0 to 10 Miles, Public: This population group evacuates over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is

modeled as a single cohort, while the rest of the group is captured as different cohorts, such as the tail.

0 to 10 Miles, Special Facilities: This is a small but unique population group within the 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 Miles, Tail: The tail represents the last 10 percent of the EPZ population who typically take a longer time to begin to evacuate.

0 to 10 Miles, 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.

Nonevacuating Public: A portion of the public does not follow protective action orders. It is assumed that 0.5 percent of the general public within the EPZ refuse to evacuate.

Table 69 Evacuation Model 1: EPZ Evacuation

Population			Response Delays (hours)				Phase Duration (hr)		Evacuation Travel Speeds (mph)		
Cohort		Population Fraction	Siren (OALARM)	Delay to Shelter	Delay to Evacuation	Total (Depart time)	Early (DURBEG)	Middle (DURMID)	Early (ESPEED)	Middle (ESPEED)	Late (ESPEED)
1	0 to 10 miles Early Evacuees	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	0.006	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	20	15	20
6	0 to 10 miles Nonevacuating Public	0.005	1	-	-	-	-	-	-	-	-

For this sequence, hotspot relocation is 5 rem at 4 hours and normal relocation is 1 rem at 8 hours. The values were established specific for this evacuation model developed for sequences with relatively small releases.

A.2 Evacuation Model 2: WinMACCS response parameters for late release sequences where the PAG is exceeded beyond the EPZ.

Preliminary results suggest that emergency-phase doses of 1 rem may extend 30 to 40 miles from the plant for some of the larger postulated releases. The EPA PAG suggests 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 a 30-mile radius of the reference plant is approximately 1.4 million. The population within the 40-mile radius is approximately 3.4 million. Because of larger populations at longer distances, it is important to better understand the potential directions that the plume would travel. The reference plant's wind rose in the figure below suggests that the predominant wind direction is to the south and east, which is generally toward lower population areas. A secondary direction in terms of likelihood is to the northwest to north. This region is also low in population. Thus, if a release were to occur, it is more likely that a relatively small population would be affected than if the release occurred at a facility near a major city.

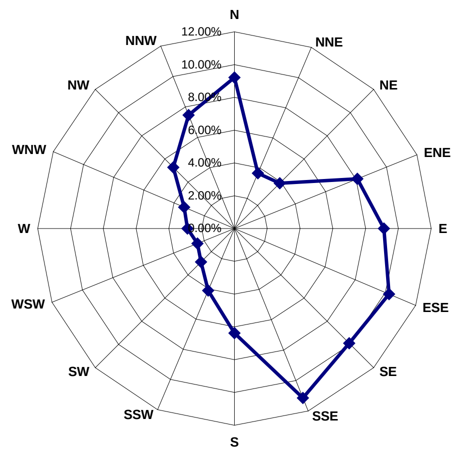

Figure 140 The Reference Plant's Wind Rose

It is assumed in this evacuation model that ORO's begin to order evacuations beyond the EPZ 24 hours after the start of the accident. This is based on preliminary results that indicates a large release beginning at 48 hours. For this sequence, the population within a 30-mile radius is evacuated after the EPZ has evacuated. The overall evacuation would be implemented as a staged evacuation, which is common for plume-related emergency response. In addition, a SIP is assumed to be ordered for the 30- to 40-mile radius area.

To develop an ETE and corresponding speeds for the areas beyond the EPZ, it was assumed 90 percent of the general public who reside between 10- and 30-miles from the plant can be evacuated 24 hours after ordered to evacuate. This is consistent with the lengthy travel times observed in hurricane evacuations of similar populations. The last 10 percent (evacuation tail) is estimated to take an additional 12 hours. Because of the lengthy time for this release to the atmosphere, this evacuation model effectively includes two separate evacuations, the first being within EPZ followed later by the 10- to 30-mile area.

The following cohorts were established for this evacuation model:

0 to 10 Miles, 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 Miles, Early Evacuees: This population begins to evacuate before receiving an evacuation order. Focus group work suggested that some residents are prepared and ready to evacuate at the first indication of an accident at the nuclear power plant (NRC, 2008c). Because the accident is initiated by a severe earthquake, it is assumed 30 percent of the public evacuate.

0 to 10 Miles, Public: This population group would evacuate over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is modeled as a single cohort, while the rest of the group is captured as different cohorts, such as the tail.

10 to 20 Miles, Shadow: These residents evacuate from areas that are not under an official evacuation order. 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. For simplicity, this cohort is assumed to be distributed uniformly over the 10- to 20-mile area. This cohort begins evacuating as they observe EPZ evacuees traveling through the area.

0 to 10 Miles, 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 Miles, Tail: The tail represents the last 10 percent of the EPZ population who typically take a longer time to begin to evacuate.

10 to 30 Miles, Public: This population group evacuates over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is modeled as a single cohort that enters the roadway network while EPZ evacuees are travelling through. The rest of the group is captured as different cohorts, such as the tail.

10 to 30 Miles, Special Facilities: Special vehicles needed to evacuate these facilities require additional time to mobilize and support the evacuation.

30 to 40 Miles, Shadow: A shadow evacuation may be expected in the area beyond the evacuation area. For this analysis, it is assumed that 20 percent of the general public from the 30- to 40-mile area evacuate. This cohort begins evacuating as they hear the order to evacuate for the 10- to 30-mile area, or observe evacuees traveling through the area.

10 to 30 Miles, Tail: The tail represents the last 10 percent of the population of this area who typically take a longer time to begin to evacuate.

30 to 40 Miles, Shelter in Place (SIP): For this evacuation model, it is assumed that 80 percent of the public remaining after the shadow evacuation complies with the SIP order.

Nonevacuating Public: A small portion of the public does not follow protective action orders. It is assumed that 0.5 percent of the general public within the 0 to 40 mile area refuse to evacuate. This group, however, is subject to relocation and a portion of this cohort is relocated according to the relocation parameters discussed above.

Table 70 Evacuation Model 2: Evacuation for PAGs exceeded beyond the EPZ (SFP release after 40 hours.)

Population			Response Delays (hours)			Phase Duration (hr)		Evacuation Travel Speeds (mph)		
Cohort		Population Fraction	Delay to Shelter*	Delay to Evacuation	Total (Depart time)	Early (DURBEG)	Middle (DURMID)	Early (ESPEED)	Middle (ESPEED)	Late (ESPEED)
1	0 to 10 miles Schools	.172	0.25	1.25	1.5	0.25	2	20	15	20
2	0 to 10 miles Early Evacuees	.2	0.5	0.5	1	1	2	20	10	20
3	0 to 10 miles General Public	.517	1	2	3	0.25	3	5	2	20
4	10 to 20 miles Shadow	.3	2	2	4	0.25	6	20	15	20
5	0 to 10 miles Special Facilities	.006	0	5	5	3	2	2	5	20
6	0 to 10 miles Evacuation Tail	.1	3	3	6	2	2	2	5	20
7	10 to 20 miles General Public	.552	24	4	28	2	18	2	1	20
	20 to 30 miles General Public	.852								
8	10 to 30 miles Special Facilities	.043	15	15	30	1	10	1	1	20
9	30 to 40 miles Shadow	.2	24	8	32	1	6	15	5	20
10	10 to 30 miles Evacuation Tail	.1	24	16	40	10	2	1	10	20
11	30 to 40 miles Shelter in Place	.795	NA	NA	NA	NA	NA	NA	NA	NA
12	0-40 miles Non-evacuating Public	.005	NA	NA	NA	NA	NA	NA	NA	NA

*Delay to shelter is from the start of the accident (i.e. OALARM set to zero)

For this sequence, hotspot relocation is 5 rem at 4 hours and normal relocation is 1 rem at 16 hours. The releases for these sequences do not begin until about 40 hours or thereafter, and hotspot relocation does not begin until 4 hours after the plume reaches the location. OROs would be able to assemble considerable resources to monitor radiological conditions and could be expected to relocate people relatively rapidly should it be necessary.

A.3 Evacuation Model 3: WinMACCS response parameters for early release sequences where the PAG is exceeded beyond the EPZ.

This evacuation is similar to Evacuation Model 2. Preliminary results suggest that certain sequences that have large releases that begin between 8 and 18 hours and are capable of emergency-phase doses that exceed the PAGs beyond the EPZ. It is expected that dose projections would indicate protective actions beyond the EPZ are necessary. It is assumed that evacuation of the area beyond the EPZ would begin at 10 hours after the start of the accident.

Because the evacuation of the 10- to 30-mile area begins at 10 hours, this response is typical of a staged evacuation that would be employed in the case of a chemical release. The EPZ evacuation impacts the evacuation speeds of the 10- to 30-mile area. The following cohorts were established for this evacuation model:

0 to 10 Miles, 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 Miles, Early Evacuees: This population begins to evacuate before receiving an evacuation order. Focus group work suggested that some residents are prepared and ready to evacuate at the first indication of an accident at the nuclear power plant (NRC, 2008c). Because the accident is initiated by a severe earthquake, it is assumed 30 percent of the public evacuate.

0 to 10 Miles, Public: This population group would evacuate over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is modeled as a single cohort, while the rest of the group is captured as different cohorts, such as the tail.

10 to 20 Miles, Shadow: These residents evacuate from areas that are not under an official evacuation order. 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. For simplicity, this cohort is assumed to be distributed uniformly over the 10- to 20-mile area. This cohort begins evacuating as they observe EPZ evacuees traveling through the area.

0 to 10 Miles, 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 Miles, Tail: The tail represents the last 10 percent of the EPZ population who typically take a longer time to begin to evacuate.

10 to 30 Miles, Public: This population group evacuates over a period of time, with some residents leaving promptly and others leaving later. For this analysis, the bulk of this group is modeled as a single cohort that enters the roadway network while EPZ evacuees are travelling through. The rest of the group is captured as different cohorts, such as the tail.

10 to 30 Miles, Special Facilities: Special vehicles needed to evacuate these facilities require additional time to mobilize and support the evacuation.

30 to 40 Miles, Shadow: A shadow evacuation may be expected in the area beyond the evacuation area. For this analysis, it is assumed that 20 percent of the general public from the 30- to 40-mile area evacuate. This cohort begins evacuating as they hear the order to evacuate for the 10- to 30-mile area, or observe evacuees traveling through the area.

10 to 30 Miles, Tail: The tail represents the last 10 percent of the population of this area who typically take a longer time to begin to evacuate.

30 to 40 Miles, Shelter in Place (SIP): For this evacuation model, it is assumed that 80 percent of the public remaining after the shadow evacuation complies with the SIP order.

Nonevacuating Public: A small portion of the public does not follow protective action orders. It is assumed that 0.5 percent of the general public within the 0 to 40 mile area refuse to evacuate. This group, however, is subject to relocation and a portion of this cohort is relocated according to the relocation parameters discussed above.

Table 71 Evacuation Model 3: Evacuation for PAGs exceeded beyond the EPZ (SFP release after 8 hours).

Population			Response Delays (hours)			Phase Duration (hr)		Evacuation Travel Speeds (mph)		
Cohort		Population Fraction	Delay to Shelter*	Delay to Evacuation	Total (Depart time)	Early (DURBEG)	Middle (DURMID)	Early (ESPEED)	Middle (ESPEED)	Late (ESPEED)
1	0 to 10 miles Schools	.172	0.25	1.25	1.5	0.25	2	20	15	20
2	0 to 10 miles Early Evacuees	.2	0.5	0.5	1	1	2	20	15	20
3	0 to 10 miles General Public	.517	1	2	3	0.25	3	5	2	20
4	10 to 20 miles Shadow	.3	2	2	4	0.25	6	20	15	20
5	0 to 10 miles Special Facilities	.006	0	5	5	3	2	2	5	20
6	0 to 10 miles Evacuation Tail	.1	3	3	6	2	2	2	5	20
7	10 to 20 miles General Public	.552	6	4	10	2	18	2	1	20
	20 to 30 miles General Public	.852								
8	10 to 30 miles Special Facilities	.043	0	20	20	1	10	1	1	20
9	30 to 40 miles Shadow	.2	6	6	12	1	6	15	5	20
10	10 to 30 miles Evacuation Tail	.1	10	20	30	10	2	1	10	20
11	30 to 40 miles Shelter in Place	.795	NA	NA	NA	NA	NA	NA	NA	NA
12	0-40 miles Non-evacuating Public	.005	NA	NA	NA	NA	NA	NA	NA	NA

*Delay to shelter is from the start of the accident (i.e. OALARM set to zero)

For this evacuation model, the hotspot relocation is 5 rem at 26 hours and normal relocation is 1 rem at 38 hours. The plume's initial release for these sequences begins between 8 and 18 hours after accident initiation. The assumed inability of the licensee to halt this release is the basis for expecting decision makers to expand the evacuation. Sequences that develop rapidly could challenge ORO resources to assess radiological conditions beyond the evacuated areas and delay relocation of affected people. In this evacuation model, hotspot relocation does not begin until 26 hours after the release arrives, in order to account for the relatively earlier release and the evacuation of the public from the 10-30 mile area (which is expected to take as long as 24 hours).

APPENDIX B: A QUALITATIVE RISK COMPARISON OF SPENT FUEL STORAGE STRATEGIES

B.1 Introduction

In Staff Requirements Memorandum (SRM) M120607C, dated July 16, 2012, the Commission directed the staff to conduct a comparative assessment of the results of the Spent Fuel Pool Study (SFPS) against previous studies of the safety consequences associated with loading, transfer, and long-term storage in dry cask storage systems (DCSS). Since the SFPS only includes a consequence study of certain seismic events, it is necessary to create a step-by-step model that can be used to compare safety consequences associated with the various stages of onsite spent fuel management. As part of the response to this SRM, this analysis (1) defines several fuel storage strategies to be compared, (2) develops a structure for calculating the difference in risks between these strategies, (3) identifies what relevant information exists, and (4) identifies what new information may be needed.

B.2 Spent Fuel Storage Strategies

For the purpose of studying this issue, two distinct spent fuel storage strategies commonly considered are defined: (1) current practice and (2) expedited transfer of spent fuel into dry storage. A large amount of variation exists in current spent fuel storage practices at various sites. Expedited transfer strategies, if implemented, would also be expected to vary considerably from site to site. Rather than attempting to bound all of the practices that are or may be implemented at various sites, this appendix will focus on the key elements of spent fuel storage strategies covered by existing risk analyses. Current practice generally consists of loading casks only when the pool, in a high density configuration, is nearly full. Just enough casks are loaded to maintain the capability to unload one full core into the pool. Expedited transfer of spent fuel into dry storage involves loading casks at a faster rate for a period of time to achieve a low density configuration in the spent fuel pool (SFP). The expedited process maintains a low density pool by moving all fuel cooled longer than 5 years out of the pool.

B.3 Spent Fuel Storage Stages

The risks associated with spent fuel storage will vary throughout the lifetime of a plant site and will depend on how the fuel is stored, and in what quantities. To analyze the lifecycle risk of spent fuel storage at a plant site, this appendix defines five fuel storage stages, beginning with a low-density pool approaching high density and ending with the final core offload being loaded into casks. The current practice strategy will not include the expedited transfer stage defined below.

Stage 1 consists of the fuel being offloaded into a low-density pool that eventually reaches high density. It is assumed that no casks are loaded during this stage as is generally industry practice. Stage 2 is when the pool is full in a high-density configuration and only as many casks are loaded as necessary. Stage 3 is during the expedited transfer period when the amount of cask loading is increased so as to decrease the inventory in the SFP. Stage 4 begins when expedited loading has been completed and the pool has returned to a low-density configuration and a lower rate of cask loading. Stage 5 begins when the reactor is permanently shut down and the last core is offloaded to the SFP. Stage 5 ends when all fuel has been placed in dry cask storage.

B.4 Risk of Spent Fuel Storage

This section presents generalized equations for the risk of spent fuel storage. These equations will serve as a guide to a subsequent discussion of the relative risk between storage stages and what drives the changes in risk.

The total annual risk of storing spent fuel during any stage can be expressed as the sum of the risk from the SFP and the dry casks. This can be expressed as,

$$R = R_{casks} + R_{sfp}$$

where: R = annual risk of spent fuel
R_{casks} = annual risk of loading and storing fuel in dry casks
R_{sfp} = annual risk of the spent fuel pool

The risk from loading and storing each dry cask is assumed to be constant and only dependent on the number of casks loaded or stored. The total risk of loading and storing casks is given by,

$$R_{casks} = r_{cask,load} * N_{load} + r_{cask,store} * N_{store}$$

where: $r_{cask,load}$ = risk per cask loaded
N_{load} = number of casks loaded per year
$r_{cask,store}$ = risk per cask in storage
N_{store} = number of casks being stored

Section 1.5 of the SFPS report provides an overview of contributors to SFP risk. The majority of SFP risk is thought to emanate from a loss of water from a leak or a boiloff. The risk from the SFP can then be characterized as the frequency of fuel uncovery multiplied by the consequences of the accident. The uncovery frequency is the sum of the frequency of uncovery from cask drops, seismic events, and other initiators. The frequency of a cask drop damaging the pool and leading to uncovery is the product of the number of casks loaded, the probability of a drop, and the probability of pool damage and uncovery given a drop. This value is given by,

$$R_{sfp} = (N_{load} * P_{drop} * P_{damage} + F_{seismic} + F_{other}) * C_{uncovery}$$

where: P_{drop} = probability of a cask drop per cask loaded
P_{damage} = probability of a dropped cask leading to fuel uncovery
$F_{seismic}$ = frequency of uncovery from seismic events
F_{other} = frequency of uncovery from sources other than cask drops and seismic
$C_{uncovery}$ = consequences of fuel uncovery

The SFPS provided a detailed analysis of the consequences, $C_{uncovery}$, for a particular site and a calculation of $F_{seismic}$ for seismic bin 3. To fully calculate $F_{seismic}$, seismic bin 4 would need to be analyzed as well. The SFPS did not analyze other initiators for pool accidents that contribute to SFP risk.

B.5 Risk during Each Stage

Figure 139 is an illustration of the spent fuel risks during each stage for both the current practice and expedited transfer strategies. Though the “current practice” strategy does not include expedited loading, it is divided into the same stages (time periods) for comparison purposes. The figure depicts the SFP risk, dry cask loading risk and dry cask storage risk. The SFP risk includes the risk to the pool from dropped casks.

Figure 141 includes the following major assumptions and limitations:

- The figure is only intended to show trends, not absolute differences in risk. No specific numbers were used to generate the figure.

- The type of risk used will significantly affect the relative values of different portions of the figure. Table 37 gives the ratio of consequences between a high- and low- density pool for several types of risk, with the risk reduction from a low-density pool varying from a factor of 2.1 for individual latent cancer fatality risk for 0–10 miles to 56 for land interdiction.

- The amount of time spent in each stage will affect a calculation of the total risk.

- Changes in N_{load}, N_{store}, and $C_{uncovery}$ are the drivers for the change in risk between stages. Other terms in the above equations are assumed to be constant.

- The specific shape of the figure will depend on site specific parameters such as the pool’s susceptibility to be drained from a cask drop event (P_{damage}).

- Risks are averaged over the operating cycle to demonstrate general trends rather than short-term changes in risk.

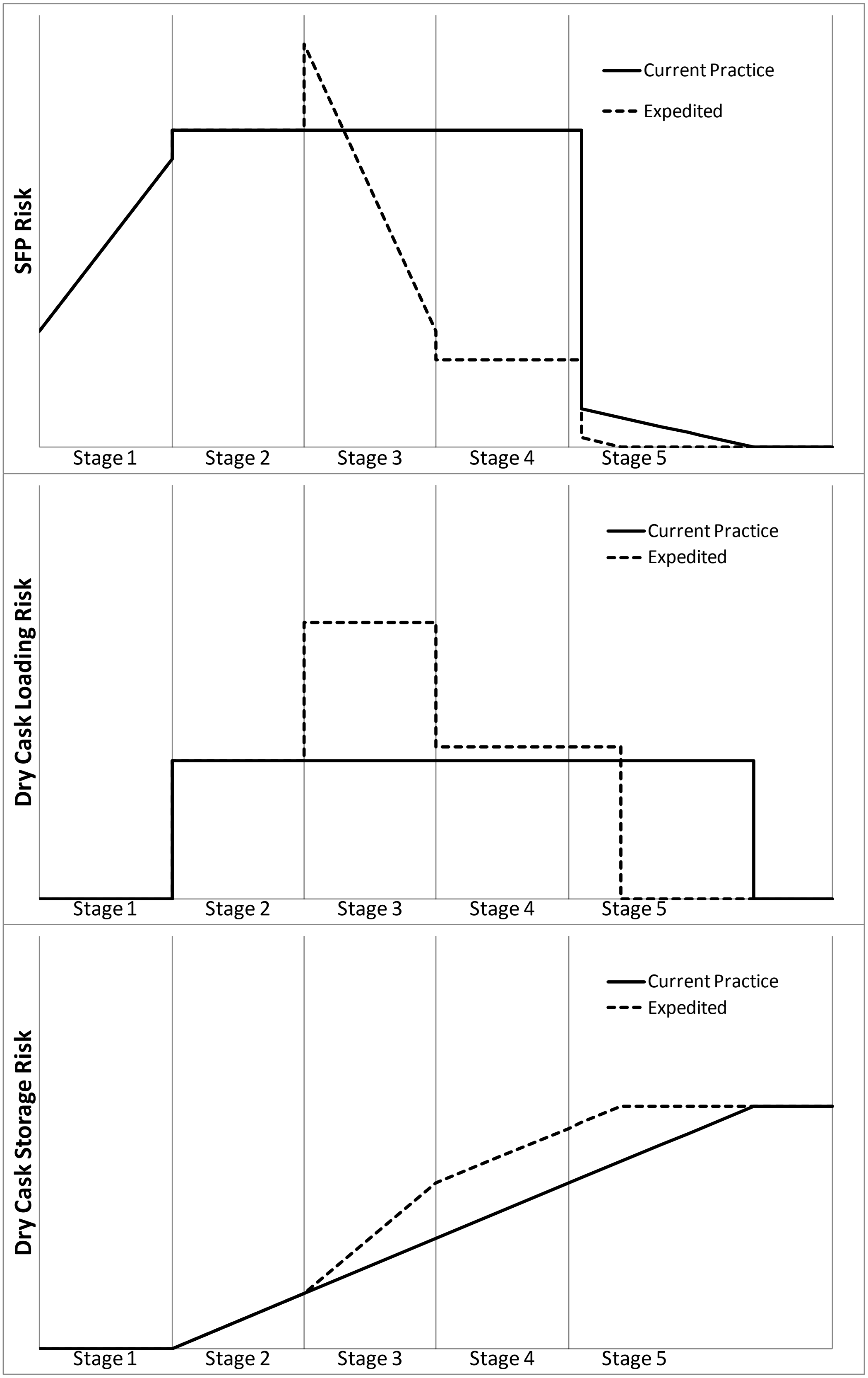

Figure 141 Graphical representation of spent fuel risks

During Stage 1, for both the current practice and expedited transfer scenarios, the amount of spent fuel being stored in the SFP is increasing. As the pool moves from low-density loading to high-density loading, the consequences of fuel uncovery, $C_{uncovery}$, and thus SFP risk, R_{sfp}, increase. Dry cask loading and storage risk is zero since no casks are loaded during this stage (N_{load} and N_{store} = 0).

At the beginning of Stage 2, the pool reaches a high-density configuration and cask loading begins. SFP risk is greater than at the end of Stage 1 because of the possibility of cask drops ($N_{load} > 0$). It is assumed that the rate of dry cask loading is constant throughout this stage, leading to a constant loading risk and a gradually increasing storage risk as more casks are stored (N_{store} is increasing). For the current practice spent fuel storage strategy, the same loading rate is maintained in Stages 2, 3 and 4 and the pool is maintained at maximum loading.

For the expedited transfer strategy, Stage 3 is the beginning of an increased cask loading rate (N_{load} increases). The SFP risk undergoes another step increase (from the increased frequency of cask drop events) and then declines as the pool approaches a low-density configuration and the consequence of fuel uncover, $C_{uncovery}$, decreases. Cask loading risk increases as the rate of loading, N_{load}, increases. Storage risk increases at a faster rate while more casks are being loaded.

Stage 4 marks the end of the expedited transfer phase when the pool has reached a low-density configuration. The cask loading rate and risk decrease to levels slightly higher than in Stage 2. The hotter fuel being loaded requires more lower capacity casks. The decrease in cask loading rate, N_{load}, and consequences of uncovery, $C_{uncovery}$, decrease the SFP risk which remains at a constant, lower level for the rest of the stage. Cask storage risk continues increasing at a slower rate.

At the beginning of Stage 5, the reactor ends its final operating cycle and fuel in the reactor core is offloaded to the SFP. After several months, the fuel in the SFP is generally capable of being air cooled, and the risk decreases for both the current practice and expedited transfer strategies. The risk is nonzero because of the possibility of events which may impede air cooling of the fuel. It is assumed that casks continue being loaded at a constant rate until the pool is empty. The SFP risk continues decreasing gradually as the fuel cools and is removed from the pool. When the cask loading is complete, the pool risk and the cask loading risk go to zero, and the cask storage risk stabilizes. The low-density pool in the expedited transfer case contains less fuel and is emptied sooner since much of the fuel was removed in Stage 3.

B.6 <u>Total Risk over Time</u>

Two components compromise the total risk over a period of time, (1) the amount of time spent in each stage and (2) the risk in each stage. The time spent in each stage will vary depending on how soon expedited transfer is initiated (if at all), how long it takes, and how long the reactor continues to operate with a low-density pool. The risks during each stage will depend on the relative hazards at each site.

The recent EPRI report analyzing spent fuel transfer (EPRI, 2012) estimates that expedited transfer for fuel cooled longer than five years would take between 8-15 years at most sites. The amount of fuel in the SFP, whether the site has multiple units sharing a single cask handling crane, and the availability of trained personnel, and equipment affect this estimate. The amount of time then spent in a low-density configuration depends on how much longer the reactor is operated. It is expected that most reactors will apply for and receive extensions of their operating licenses to 60 years.

The site-specific risks during each stage will drive whether expedited transfer decreases risk, and over what timeframe. An accounting of the risk increases and decreases of expedited transfer compared to current practice will illustrate this point.

For expedited transfer, risk increases relative to current practice are seen in the following stages:

- SFP risk in the beginning of Stage 3 from cask drop events,
- Cask loading risk during Stage 3, and to a lesser extent in Stages 4 and 5,
- Storage risk in Stages 3 and 4 and the beginning of Stage 5

Risk decreases occur in:

- SFP risk later in Stage 3 and in Stage 4,
- cask loading risk in Stage 5

Since the total number of casks loaded is likely to be only slightly higher for the expedited strategy, the increase in cask loading risk during Stage 3 is expected to be mostly offset by the decrease in risk in Stage 5. Furthermore, previous studies such as NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System At a Nuclear Power Plant," issued March 2007, and an EPRI report entitled, "Probabilistic Risk Assessment (PRA) of Bolted Storage Casks," indicate that dry cask storage risk is likely lower than SFP risk. Hence, this leaves a comparison of the increase in SFP risk from cask drops to the decrease in risk from a low-density pool.

The risk increase will depend on the pool's susceptibility to drops (discussed below). For example, if the SFP risk at a particular site were dominated by cask drop risk, it could take many years of operating with a low-density pool to "pay back" the temporary risk increase seen at the beginning of Stage 3. This increase in risk could be mitigated by only loading casks during operating cycle phases 4 or 5 when the SFP is typically air coolable for at least 72 hours, for a complete draindown. In contrast, a pool with low susceptibility to cask drops and high seismic risk will see a greater risk benefit sooner.

One aspect not included in the above figure is the potential need to repackage casks that have already been loaded, before interim storage or permanent disposal. Given the uncertainty in the national strategy for spent fuel, the specifications for disposal at a long-term repository or interim storage site are currently unclear but may be developed in the future. Earlier movement of fuel into current cask designs increases the probability that fuel will have to be repackaged to meet these specifications.

B.7 Availability of Information

The equations defined earlier identify the variables needed to calculate the risk of spent fuel storage. The general shape of
Figure 141 is believed to generally apply to operating reactors. However, no analysis has attempted to quantify the horizontal or vertical axes. The discussion below points to existing

information that could be useful in quantifying these variables as well as what further information could be useful.

B.7.1 Cask Risks ($r_{cask,load}$ and $r_{cask,store}$)

Two major studies have addressed the risk of dry cask loading and storage, NUREG-1864 and EPRI's probabilistic risk assessment of bolted storage casks. NUREG-1864 analyzed a welded cask at a particular boiling-water reactor site. In a complementary effort, the EPRI study analyzed a bolted cask at a generic pressurized-water reactor site. NUREG-1864 identified cask drops and aircraft strikes as the major contributors to risk during cask loading and cask storage, respectively. The EPRI study found the major contributors to risk during cask loading to be drops, failure of the refueling building, and high temperature fires. During storage, major risk contributors were high temperature and forces (e.g., explosions) or heavy loads (e.g., high winds) leading to cask tipover. The difference in major contributors to risk is likely because of differences in the methods used in the analysis as well as differences in the analyzed systems. Regardless, both studies found the risk of dry cask loading and storage to be extremely small. Key factors contributing to this result include the robustness of the analyzed casks and the availability of the refueling building ventilation system, which is capable of significantly decreasing the source term for many accident sequences that result in a cask release.

Several additional factors may affect a calculation of dry cask risk. Considerable uncertainty exists in the source term expected from cask accident sequences resulting in a significant range in consequences, as discussed in Chapter 10. Different cask designs will vary in their ability to resist hazards and may have failure modes not considered in previous studies. Since no standard for performing a dry cask PRA exists, these issues will have to be addressed on a case-by-case basis. The applicability of the assumptions and limitations of previous studies to any future analysis will have to be carefully considered.

B.7.2 Number of Casks (N_{load} and N_{store})

The number of casks loaded, N_{load}, and stored, N_{store}, affects the total cask risk, R_{cask}. The number of casks loaded also affects the SFP risk, R_{sfp}, because of the potential for cask drops. As discussed above, cask loading is assumed to begin in Stage 2, increase during Stage 3 (expedited transfer), and, in Stage 4, return to a lower level necessary to maintain a low-density configuration in the pool.

NUREG-1353 (NRC, 1989a) and the EPRI report on spent fuel transfer (EPRI, 2012) include estimates for the number of casks loaded. NUREG-1353 initially assumes two casks are loaded per week resulting in 104 loads per reactor year. Using assumptions based on more updated information regarding the number of assemblies discharged per reload, the length of the fuel cycle, and the capacity of storage casks in use at the time, the analysis revised this estimate downward by a factor of 10. The EPRI report contains a more detailed analysis considering multi-unit sites and possible expedited loading scenarios. Dependent on these factors, the number of casks loaded annually was estimated to average 3 to 4 annually for current sites and up to 15 to 19 annually for some sites during expedited loading.

At the end of 2011, more than 1,500 casks had been loaded (EPRI). For a comparative risk calculation, site-specific information would have to be collected or estimated. Given the uncertainties in the calculation of risk per cask, and the fact that the risk of loading and storing casks has been estimated to be lower than the risk associated with SFPs, a precise number of casks loaded and stored is not expected to drive the results.

As a first approximation, one might assume that the total number of casks loaded from the SFP would be the same no matter the fuel management strategy. However, expedited fuel transfer requires loading fuel with a higher heat load into casks. Therefore, expedited fuel transfer may result in more casks being loaded with different accident consequences than the current package. The EPRI report estimates the increased number of casks required.

B.7.3 Pool Uncovery Frequency from Cask Drop Events (N_{load}, P_{drop} and P_{damage})

Heavy load drops have the potential to damage the SFP, possibly leading to uncovery of the fuel. In general, casks are considered to be the only loads handled over the pool heavy enough to have the potential to cause structural damage. Other heavy loads are usually prevented from traveling directly over the pool.

SFPs can have a variety of configurations affecting their susceptibility to cask drop events. Some pools contain cask loading pits with floors at a higher elevation than the bottom of the pool. Damage from a cask drop event would only drain the pool to a certain level, potentially giving operators sufficient time to align a makeup water source and continue keeping the fuel covered. The cask loading pit may be separated from the pool by a gate. In other pools, casks are loaded directly in the SFPs in a section which may or may not be reinforced to reduce the risk of damage in a cask drop event.

The total frequency of uncovery will be a function of how many casks are loaded, the estimated probability of a drop per loading, and the probability of damage given a drop. Expedited transfer of spent fuel will lead to increased cask loading for a number of years, increasing the risk of a dropped cask damaging the pool. The number of casks loaded is discussed above.

Several studies have addressed the issue of heavy load drops and the anticipated effect on the SFP. NUREG-1353 estimated a drop rate of 3.1×10^{-4} per reactor year assuming two lifts per week without consideration of Generic Safety Issue (GSI) A-36, "Control of Heavy Loads Near Spent Fuel." The reduction in the probability of a cask drop for a plant which complies with the resolution of GSI A-36 was estimated to be a factor of 0.001 for a revised probability of $3.1\text{x}10^{-7}$ per reactor year. A LLNL analysis reported in NUREG/CR-5176 in support of NUREG-1353 considered worst-case cask drops on the pool wall for a boiling-water reactor and a pressurized-water reactor (Prassinos, 1989). The analysis concluded that it is likely that the liner would be severely damaged, so a value of 1 was used for P_{damage}. Based on updated information, NUREG-1353 judged two lifts per week (104 per year) to be an overestimate by about a factor of 10. The final estimate of the frequency of a cask being dropped and damaging the SFP is 3.1×10^{-8} with an upper bound estimate of 3.1×10^{-7}. This analysis only considered casks dropped on the SFP wall.

NUREG-1738 considered drops that would catastrophically fail the pool, leading to a rapid draindown and failure modes other than drops onto the SFP wall (NRC, 2001). The analysis assumed that only casks are heavy enough to cause catastrophic damage to the pool. Data from NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," issued July 1980, and NUREG/CR-5176 were combined with a calculation of the fraction of the load path spent over the pool and the fraction of the total path the load is lifted high enough to damage the pool to estimate the probability of a drop that damages the pool. For a nonsingle-failure-proof crane, the drop frequency was estimated, based on NUREG-0612 information, to have a mean value of 2.1×10^{-5} per year (using 100 lifts per year). For single-failure-proof cranes or plants that conform to the NUREG-0612 guidelines, the drop frequency was estimated to have a mean value of 2×10^{-7} per year (for 100 lifts per year). The analysis assumed that licensees with a non-single-failure-proof crane took appropriate mitigative actions to reduce the expected frequency of catastrophic damage to the same range as for facilities with a single-failure-proof crane.

NUREG-1864 used empirical drop data reported in NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," issued July 2003, to estimate the probability of dropping a cask. Three load drop events were identified from an estimated 54,000 lifts in the 1968–2002 time period, giving a probability of 5.6×10^{-5} per lift. This probability was considered conservative given that, of the three events, only one was a freefall while the other two were uncontrolled descents. The probability of pool damage was not estimated.

The EPRI dry cask PRA constructed a fault tree of a crane to address a range of factors and to account for specific crane features. Data from NUREG-0612 and other sources were used to estimate failure probabilities for basic components as well as human error probabilities. The final probability of a cask drop given a lift was estimated to be 5.3×10^{-6}. The probability of pool damage was not estimated.

Data cataloguing the susceptibility of SFPs to cask drops for the reactor fleet is not readily available. To address this issue, a risk analysis would need to either perform a site-specific analysis of cask drops, or conservatively assume that most (if not all) drops will damage the pool.

B.7.4 Pool Uncovery Frequency from Seismic Events ($F_{seismic}$)

The frequency of seismic events damaging the pool liner and leading to fuel uncovery depends on both the seismic hazard (i.e., the frequency of the initiating event) and the fragility of the SFP (i.e., the probability that the liner fails given that the event occurs). Chapter 3 of the SFPS report discusses the availability of seismic hazard information.

In contrast, seismic fragility data has not been characterized for most SFPs. NUREG/CR-5176 used a fragility analysis approach involving seismic loads derived from floor response spectra for the reactor building based on design response spectra. These loads were then combined with analytical methods for the calculation of the fundamental period of vibration of SFP floors and walls, as well as approximate methods for the calculation of the strength of these floors and walls. The approach used to derive the SFP fragility was generally consistent with methods used for seismic margin assessments at the time of that study. Since NUREG-1738 was not a site-specific analysis, an attempt was made to generalize this information. NUREG-1738 convolved a generic fragility (roughly corresponding to the fragilities calculated in NUREG/CR-5176) with EPRI and LLNL seismic hazard estimates to estimate the seismic risk. The study also developed a screening checklist such that a plant passing the checklist would have confidence of having a pool fragility of at least the assumed amount.

Finally, the SFPS performed a detailed analysis for the reference plant employing a combination of the approach used in NUREG/CR-5176 to estimate seismic loads with finite element analyses of the SFP structure to calculate hydrodynamic impulsive loads, nonlinear response mechanisms and strain concentrations in the liner. Chapter 4 of the SFPS report describes the structural analysis and estimated SFP performance. Chapter 10 provides a comparative assessment of the estimated performance for the SFP considered in the study with the performance of SFPs in recent major earthquakes in Japan including the SFPs at the Fukushima Daiichi nuclear power plant under the Tohoku earthquake of March 11, 2011.

The most robust way to estimate the seismic risk would be to utilize existing hazard estimates, and perform a site-specific fragility analysis. For some analyses, particularly those considering multiple sites, this may be time and cost prohibitive since the staff and licensees do not generally have fragility analyses of the pools. A generalized analysis for all plant sites would have to address the uncertainty in the variation of SFP responses to seismic events. One approach would be to use a conservative fragility estimate and to develop a checklist to ensure that the estimate is appropriate for the pools being considered.

B.7.5 Pool Uncovery Frequency from Other Events (F_{other})

As discussed in Section 1.5 of the SFPS report, the majority of SFP risk is believed to emanate from pool leakage events such as cask drops and seismic events discussed above, as well as events that preclude water injection for a long period of time (e.g. days). Table 1 in the SFPS report shows the frequency of fuel uncovery from various contributors calculated in NUREG-1353 and NUREG-1738.

B.7.6 Pool Consequences ($C_{uncovery}$)

The SFPS is the most detailed analysis to date of SFP consequences. As discussed in the SFPS report, the study was performed for a specific site for a specific initiating event. However, the consequence results will largely hold for other initiating events and may offer insights applicable to other sites. When applying the consequence results to one or several other sites, the assumptions used in the study, discussed in Chapter 2 of this report, must be considered along with which factors drove the results, discussed in Chapter 10 of this report. Some of these drivers and how they are expected to vary between plants are discussed below.

Once fuel in the pool has become uncovered, it may still be coolable from natural circulation of air, depending on the amount of decay heat and the amount of cooling. In the SFPS, the fuel is estimated to not be air coolable for 10 percent of the operating cycle. Factors affecting this include the amount of fuel in the pool, its configuration, burnup, geometry of the fuel racks, etc. A partial draindown event with channeled fuel could impede airflow and increase the time to coolability.

A significant driver of the amount of radioactivity released is whether a hydrogen combustion event occurs. The SFPS results predict these events in some high density loading situations, but not in any low density loading scenarios. It’s not clear whether this result will hold true for other reactor sites and what level of pool loading is sufficient to achieve this result. Furthermore, the SFPS did not consider hydrogen events from hydrogen originating from a concurrent reactor accident.

The consequence metric used will significantly affect the outcome of any comparative risk calculation. Comparisons of results using different consequence metrics are seen in Table 38 of the SFPS for high-density versus low-density fuel loading and are discussed in Chapter 10 for previous SFP and dry cask risk studies. These results demonstrate that the individual latent cancer fatality risk metric is relatively insensitive to changes in release magnitude for spent fuel accidents. Other metrics, such as land contamination, are much more affected by the amount of radioactivity released. Specific reasons for this are discussed in more detail in Sections 7.6 and Chapter 10.

Other site-specific factors that may affect the consequences of pool uncovery include SFP inventory, mitigative measures, and the surrounding population density and land value. The SFPS analysis for these aspects may have varying levels of applicability to other sites.

B.7.7 Other Spent Fuel Risk Considerations

Several additional events are not believed to significantly contribute to spent fuel risk. Dropping a single assembly is not expected to challenge the integrity of the pool, but may release some radiation. NUREG-1864 analyzes this event. Criticality events, which NUREG-1353 assumes not to be an issue, are considered in Section 3.6 of NUREG-1738. Although this report does not explicitly evaluate criticality events, Section 2.3 does discuss them.

B.8 Conclusions

This appendix discusses some of the information needed to perform a risk comparison of spent fuel storage strategies. NUREGs 1353, 1738 and 1864 provide much of the information for certain plants, and could be supplemented by the site-specific analyses described in the SFPS report. A complete comparison depends on several factors, including the relative, site-specific risks, and the time spent in each stage of fuel storage. The benefit of lower SFP risk from low density loading may be offset by increases in other risks, such as the risk from cask drop events damaging fuel in the cask or the SFP. However, the magnitude of that offset has not been completely calculated for any single plant. Additional work would be necessary to evaluate the applicability of existing information to a particular site or a group of sites.

APPENDIX C: COMMISSION AND ACRS CORRESPONDENCE

Letter from the Advisory Committee on Reactor Safeguards on April 25th 2012 (ADAMS Accession No. ML1208A216): In this letter the ACRS describes the results of a meeting between the Advisory Committee on Reactor Safeguards and the Office of Nuclear Regulatory Research, to the NRC Chairman.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

April 25, 2012

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SPENT FUEL POOL SCOPING STUDY

Dear Chairman Jaczko:

During the 593rd meeting of the Advisory Committee on Reactor Safeguards, April 12-14, 2012, we reviewed the methods and approaches being used in the Office of Nuclear Regulatory Research (RES) Spent Fuel Pool Scoping Study (SFPSS). Our Materials, Metallurgy, and Reactor Fuels and Reliability and PRA Subcommittees jointly reviewed the methods and approaches as well as preliminary results of this study on March 6, 2012. During these meetings, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS

1. The SFPSS is being performed in an organized and systematic manner, and is using modern NRC codes to evaluate the change in consequences from seismically induced spent fuel pool accidents with high and low-density loading.

2. The SFPSS consists of a detailed deterministic analysis of the consequences of a severe seismic event on a spent fuel pool at a single boiling water reactor (BWR) site.

3. The study will contribute to the technical basis for making decisions regarding expedited transfer of older irradiated spent nuclear fuel (SNF) from spent fuel pools.

BACKGROUND

The SFPSS was initiated in July 2011 and is planned for completion by June 2012. The staff was tasked with developing updated information on key aspects of potential spent fuel pool accident consequences on an aggressive one year schedule. The staff was directed to move expeditiously in a technically rigorous manner, using modern codes and methods.

To accomplish this task, the staff reviewed past consequence and risk assessments related to SNF storage, as well as other reports of relevance that have been developed by other organizations. The staff identified seismic hazard as the logical starting point to assess the continued applicability of past studies and to develop insights for the SFPSS. Depending on the results gained from the SPFSS, additional work may be appropriate to reach generally applicable conclusions for the U.S. BWR and pressurized water reactor (PWR) fleet.

Along with providing general updates to past information within the current operational and regulatory environment, the staff indicated that for the scenarios investigated, the study can address key questions and provide insights, such as:

- Do accident progression timelines for SNF pools proceed more slowly than previously thought?

- Do seismically induced station blackout scenarios contribute significantly to the overall consequences, or are these consequences dominated by seismically induced pool drain down?

- Do low-density loadings in spent fuel pools produce substantially different results in terms of public health effects and offsite consequences compared to high-density loadings?

- Do successful post event mitigation actions substantially reduce offsite consequences?

The staff indicated that answers to these questions are expected to be helpful in determining whether expedited transfers of SNF from pools to dry cask storage systems (DCSS) produce substantial safety benefits, thereby informing future regulatory decision making. Other ongoing efforts, such as planned Level 3 probabilistic risk analyses, will complement and build on this work.

DISCUSSION

The technical approach selected by the staff is focused on a detailed analysis of the spent fuel pool in a General Electric BWR-4 reactor at a single site during five phases of an operating cycle. Two conditions in the pool are considered: one representative of the current high-density loading in a relatively full SNF pool, and another representative of low-density loading in which older SNF has been removed to a dry cask storage facility. The elements of the study will include:

- seismic and structural assessments of the integrity of the pool and liner following seismic events with up to six times greater peak ground acceleration than the design basis safe shutdown earthquake (SSE) for the examined site;

- analysis of reactor building dose rates using the SCALE code package;

- accident progression analyses of fuel damage, fission product release, benefits of mitigation, and other effects using the MELCOR code modified to handle spent fuel pool accidents;

- emergency planning assessment;

- offsite consequences analyses of health effects and land contamination using the MACCS2 code; and

- probabilistic considerations.

The approach taken by the staff is capable of producing useful assessments of the consequences of severe seismic events on the structural integrity of the selected spent fuel pool design. The study is also capable of producing quantitative assessments of the safety benefits of low density fuel pool loading on the extent of fuel damage, land contamination, and off site health effects. However, since the study will not address the safety consequences of the same severe seismic events on cask loading, transportation, or long-term storage; the overall safety benefit will not be quantified. The possibility that there could be negative safety consequences associated the expedited loading, transfer, and long-term storage of possibly thousands of DCSS would need to be considered.

For the reasons noted above, the conclusions of the study may not be broadly applicable to the variety of reactor and pool designs in operation in the United States.

We look forward to a future review of the results and conclusions of the SFPSS.

Additional comments by ACRS Members J. Sam Armijo, Michael T. Ryan, Stephen P. Schultz, and Gordon R. Skillman are presented below.

Sincerely,

/RA/

J. Sam Armijo
Chairman

Additional Comments by ACRS Members J. Sam Armijo, Michael T. Ryan, Stephen P. Schultz, and Gordon R. Skillman

The staff's approach is rightly focused on the effects of severe seismic events on the structural integrity of U.S. spent fuel pools. Absent a failure of the pool structure and liner, there can be no rapid or uncontrollable draining of the pool, overheating and failure of the fuel cladding, release of radioactive fission products, and exposure to workers and the public. In the absence of pool failure and drain down, fuel cooling will be maintained in either the high density or low density loading scenarios to be studied in the SFPSS.

In view of the importance of pool structural integrity following seismic events, the SFPSS should be broadened to consider the performance of the spent fuel pools at the Fukushima Daiichi, Fukushima Daini, Onagawa, and Tokai sites following the severe Tohoku earthquake of March 2011, as well as the performance of the spent fuel pools at the Kashiwazaki-Kariwa site following the severe Chuetsu earthquake of July 2007. None of these pools suffered structural failure or drain down. The demonstrated robustness of the spent fuel pools at Fukushima Daiichi was noteworthy. These pools were subjected to the initial M9 earthquake, followed by several aftershocks greater than M7, and hundreds of lesser magnitude. In addition, the potentially weakened spent fuel pools in Units 1, 3 and 4 survived further structural loading from hydrogen explosions without significant damage or draining of the pool water. Although limited in scope, inspection of the fuel and sampling of the spent fuel pool water in the badly damaged Unit 4 revealed that the fuel had not suffered significant damage. By any reasonable standard, the performance of spent fuel pools protected the fuel from significant damage.

Since the spent fuel pools at Fukushima Daiichi were of the same design and vintage as the design chosen for the SFPSS, this broader approach could provide valuable data to confirm or correct the findings of the study.

REFERENCES

1. Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, NUREG-1738, February 2001(ML010430066)

2. RES Memorandum, Subject: Project Plan for Spent Fuel Pool Scoping Study, July 26, 2011 (ML111570370)

Letter from the NRC Staff (ML12137A271): Letter was to the ACRS Chairman, Dr. J. Sam Armijo in response to his earlier letter to the Commission.

May 23, 2012

Dr. J. Sam Armijo, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS LETTER, DATED APRIL 25, 2012, ON THE SPENT FUEL POOL SCOPING STUDY

Dear Dr. Armijo:

I am responding to your letter of April 25, 2012, in which you provided the comments of the Advisory Committee on Reactor Safeguards (ACRS) on the staff's Spent Fuel Pool Scoping Study that was presented to the ACRS on April 12–14, 2012.

The staff agrees with ACRS' summary of the limitations of the study. The study results and the limitations will be considered with other factors when the staff evaluates the Fukushima Lessons Learned Tier 3 issue of whether spent fuel should be transferred to dry cask storage earlier than currently planned by the nuclear power plant licensees. As summarized in your letter, such factors include, but are not limited to, the risks of additional fuel handling when loading and transferring the spent fuel to the casks. The staff will also consider the relevant operating experience related to the integrity of spent fuel pools subjected to severe seismic events.

We appreciate the comments on the staff's plans for the study and look forward to further interactions on this topic.

Sincerely,

/RA Michael F. Weber for/

R. W. Borchardt
Executive Director
for Operations

cc: Chairman Jaczko
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
SECY

Staff Requirements Memo of July 16th 2012 (ADAMS Accession No. ML121980043): SRM directing staff to include additional scope to the Spent Fuel Pool Scoping Study report.

July 16, 2012

ML121980043

IN RESPONSE, PLEASE
REFER TO: M120607C

MEMORANDUM TO: Edwin M. Hackett, Executive Director
Advisory Committee on Reactor Safeguards

R. W. Borchardt
Executive Director for Operations

FROM: Andrew L. Bates, Acting Secretary ***/RA/***

SUBJECT: STAFF REQUIREMENTS – MEETING WITH THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, 9:30 A.M., THURSDAY, JUNE 7, 2012, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission met with the Advisory Committee on Reactor Safeguards (ACRS) to discuss the Committee's recent accomplishments and its ongoing and future activities. The ACRS presented updates on the following specific issues: 1. Spent Fuel Pool Scoping Study (SFPSS), 2. Implementation of Fukushima Recommendations, 3. State-of-the-Art Reactor Consequences Analyses (SOARCA), and 4. NRC Research Program.

As the ACRS noted in its April 25, 2012, letter on the SFPSS and reiterated during its meeting with the Commission, "since the study will not address the safety consequences of the same severe seismic events on cask loading, transportation, or long-term storage, the overall safety benefit will not be quantified. The possibility that there could be negative safety consequences associated with the expedited loading, transfer, and long-term storage of possibly thousands of DCSS [dry cask storage systems] would need to be considered."

The Office of Nuclear Regulatory Research should conduct a comparative assessment of SFPSS results against previous studies of safety consequences associated with loading, transfer, and long-term dry storage. These previous studies should be updated as necessary to conduct the comparative assessment.

The staff should also conduct a human reliability analysis focused on the capability to implement effective spent fuel pool cooling mitigating strategies, such as those required by 10 CFR 50.54(hh) or the recently issued Order EA-12-49, "Mitigation Strategies for Beyond-Design-Basis External Events."

In addition, the SFPSS should consider the evidence from the performance of the spent fuel pools during the real incidents identified in the additional comments by ACRS members in the April 25, 2012, letter.

The results of the SFPSS and the comparative assessment should be provided to the ACRS for its review, and subsequently provided to the Office of Nuclear Reactor Regulation for use in disposition of the Near-Term Task Force Tier 3 item on spent fuel storage, and sent to the Commission as an information paper after the staff has addressed the ACRS's comments.

cc: Chairman Macfarlane
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
OGC
CFO
OCA
OIG
OPA
Office Directors, Regions, ASLBP (via E-Mail)
PDR

ACRS letter to Honorable Allison M. Macfarlane, NRC Chairman, from J. Sam Armijo, ACRS Chairman, dated July 18, 2013 (ADAMS Accession No. ML13198A433).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 18, 2013

The Honorable Allison M. Macfarlane
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SPENT FUEL POOL STUDY

Dear Chairman Macfarlane:

During the 606th meeting of the Advisory Committee on Reactor Safeguards, July 9-12, 2013, we reviewed the NRC staff's report, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor." This study is commonly referred to as the Spent Fuel Pool Study (SFPS). Our Materials, Metallurgy, and Reactor Fuels and Reliability and PRA Subcommittees jointly reviewed the methods, approaches, and findings of this study on May 8, 2013. During these meetings we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The SFPS has been performed in a thorough and systematic manner, and provides a state-of-the-practice assessment of the consequences of a beyond-design-basis seismic event on the spent fuel pool in a reference boiling water reactor containing either high-density or low-density fuel loading.

2. The study has demonstrated that health effects from seismically initiated spent fuel pool damage scenarios are very low for both low-density and high-density pool loadings.

3. We agree with the staff's conclusion that expedited transfer of spent fuel from the pool to dry cask storage does not provide a substantial safety enhancement for the reference plant.

4. An important insight from the SFPS is that the less conventional (1x8) high-density fuel loading configuration actually used in the reference plant can significantly reduce the consequences of seismically induced damage. This approach should be further explored as a measure to provide additional defense in depth against spent fuel pool accidents.

5. The SFPS provides sound approaches, tools, and insights for a broader evaluation of the consequences of severe seismic events on spent fuel pools of different design and will be valuable in determining whether expedited transfers to dry cask storage systems (DCSSs) produce substantial safety benefit for U.S. BWRs and PWRs.

6. The SPFS should be issued.

BACKGROUND

One of the early NRC concerns following the March 11, 2011 Great East Japan Earthquake and tsunami was the possibility that the spent fuel pools at Fukushima had been severely damaged and drained, that fuel cladding had failed due to overheating, and could be releasing very large amounts of radioactive material. This early concern was not unreasonable in view of the unexpected detonation at Unit 4 in which all the fuel was in the spent fuel pool. Within a short time it became clear that the spent fuel pools at Fukushima were not damaged or leaking, the fuel cladding was undamaged, and uncontrolled radiological releases had not occurred. Despite these observations, concerns remained that spent fuel in the pools constituted a major threat to public health and safety. As a result, expedited transfer of spent fuel from the pools in U.S. reactors to DCSSs has been proposed.

The primary objective of the SFPS was to determine the safety benefit of expedited transfer of spent fuel from the pool of a reference BWR (of similar design to those at Fukushima) to DCSSs. The staff reviewed past consequence and risk assessments related to spent fuel storage. These studies showed that risk of fuel uncovery was low, but that the consequences could be large. These prior studies also showed that seismic hazard was the dominant contributor to spent fuel pool risk. For these reasons, the staff selected a study of the consequences of a beyond-design-basis seismic event at a General Electric BWR-4 Mark I reference plant as the starting point to assess the continued applicability of past studies and to evaluate the safety benefit of low-density versus high-density fuel loading in spent fuel pools. Depending on the results gained from the study, it was recognized that additional work would be needed to reach generally applicable conclusions for U.S. BWRs and PWRs.

In a Staff Requirements Memorandum (SRM), dated July 16, 2012, the Commission directed the staff to incorporate human reliability analysis (HRA) and conduct a comparative assessment of the results of the SFPS with previous studies of the safety consequences associated with loading, transfer, and long-term storage in DCSSs. In addition, the SRM stated that the SFPS should consider comparisons of the performance of spent fuel pools during the real incidents identified by the ACRS members in the April 25, 2012, letter with the predicted behavior of the reference spent fuel pool.

DISCUSSION

The SFPS is a detailed analysis of the consequences of a severe, beyond-design-basis, seismic event on the spent fuel pool in a BWR-4 Mark 1 reference plant at a specific site. The analysis was intended to be as realistic as possible with minimal deliberate additions of conservatism. The study examines the benefits of 10 CFR 50.54(hh)(2) mitigation procedures and equipment required by the Power Reactor Security Rulemaking, although it does not consider the capabilities associated with the FLEX program or the requirements of Orders EA-12-051 and EA-12-049.

Seismic and Structural Analyses

The seismic initiator selected for the study was an event with a peak ground acceleration (PGA) range of 0.5 to 1.0 g, a geometric mean PGA of 0.7 g, and a likelihood (based on PGA) of 1 in 60,000 years. The staff concluded that this event provided a good compromise between events with higher occurrence frequencies that would lead to little or no damage versus higher consequence events with very low frequencies. This selection was consistent with prior spent fuel pool probabilistic risk analyses and similar to the seismic event chosen for the State-of-the-Art Reactor Consequences Analysis short-term station blackout study. The event is considerably more severe than the current design basis Safe Shutdown Earthquake (SSE) of the reference plant (PGA of 0.12 g) and the SSEs of most U.S. plants. The horizontal response spectrum is the Generic Issue-199 spectral shape, which used the 2008 U.S. Geological Survey model, scaled to a PGA of about 0.7 g.

Structural modeling and analyses were performed to determine the damage states that would be produced by the reference seismic event and to define the initial conditions for the damage progression analysis. Cracking of the reinforced concrete pool and tearing of its stainless steel liner at the bottom of the pool had the greatest potential for creating leaks and potentially draining the pool, but other damage states were not ignored. The study included quantitative assessments of: the amount of water lost by sloshing, damage to the refueling gate, damage to support systems and penetrations, damage to the reactor building, and damage to other relevant structures. Qualitative assessments were also made of damage to spent fuel racks and stored fuel assemblies.

In-structure response spectra (ISRS) at elevations of interest were obtained by scaling the ISRS developed for the reference plant for the NUREG-1150 study. The loads on the spent fuel pool structure included the static weight of concrete, steel, water, storage racks, and fuel; the dynamic forces from the ground motion; and hydrodynamic forces from the water in the pool. These loads and best-estimate mechanical properties of all pool materials including concrete, structural steel, and the liner were used as inputs to a three dimensional nonlinear finite element analysis of the pool structure and its supports to determine the extent of cracking of the concrete and deformation or tearing of the liner.

The integrity of the spent fuel pool liner is the most important factor influencing the rate and extent of damage following the seismic event. To address liner integrity, the staff constructed a detailed finite element model of the reference plant liner, including floor and wall attachments, using elements as small as 0.15 inches. This model was integrated into the concrete base model to calculate the strains in the liner. The maximum liner strains were concentrated at the attachments to the backup plates embedded in the concrete. These maximum strains however were less than the expected failure strain for the liner material. The staff performed an analysis which considered variability in the failure strain, uncertainties in the ISRS accelerations, uncertainties in concrete properties, and reductions in spectral accelerations to account for ground motion incoherency and nonlinear effects.

The results of these analyses confirmed that a state with no liner tearing was most likely with a relative likelihood in excess of 90%. For the state in which there was a 10% likelihood of liner tearing, the staff assumed that there was an equal likelihood that tearing of the liner would result in small leaks or moderate leaks. The staff estimated the average flow rate for a small leak to a height of about 16 ft above the spent fuel pool floor is about 200 gallons per minute. Their estimate of the average flow rate for the moderate leak to a height of about 16 ft above the spent fuel pool floor is about 1,500 gallons per minute.

The analyses of the likelihood of liner tearing and the resulting flow rates are subject to considerable uncertainty associated with the characteristics of the cracking in the concrete, the loading on the structure, the initial condition of the liner and the concrete, the prediction of failure in the liner material under complex stress states, and the leak rates through cracks in concrete and steel.

The staff believes that this assessment is somewhat conservative. We concur with this assessment. The impact of such uncertainties should be considered in the context of the overall behavior of the system. Caution should be exercised in extrapolating the fragility results from the analysis of this particular spent fuel pool to other pool configurations.

The results of the seismic and structural analysis can be described in terms of three initial damage states which are used as inputs to follow-on analyses of accident progression:

- A state with no leakage. In this state, the liner would be intact, but through-wall cracking of the concrete would occur at the junction of floor and wall of the pool. The likelihood of this state was determined to be 90%.

- A state with small leakage. In this state the rate of leakage would be controlled by the number and dimensions of localized tears in the liner. The likelihood of this state was assumed to be 5%. Absent any mitigation, the pool would drain completely in approximately 42-62 hours.

- A state with moderate leakage. In this state the liner would be severely damaged and the rate of leakage would be controlled solely by the dimension of cracks in the concrete. The likelihood of this state was assumed to be 5%. Absent any mitigation, the pool would drain completely in approximately 6-9 hours.

Accident Progression Analyses and Radiological Releases

Detailed analyses of accident progression were performed using the MELCOR code. Various models within the code were used to evaluate major factors affecting accident progression. These included the pool geometry, fuel design, fuel loading configurations, rack design, decay heat, radionuclide inventory, thermal radiation, air oxidation, hydraulic resistance, fission product release and transport, and the effects of mitigative sprays.

To address the varying heat load and fission product content of the spent fuel pool, the plant operating cycle, was divided into five operating cycle phases (OCPs):

- OCPs 1 and 2 include the refueling period during which the spent fuel pool is connected to the reactor, and the decay heat generation rate is at its highest. These phases account for 3% of the plant operating cycle.

- OCP 3 is a post outage phase in which the spent fuel pool is no longer connected to the reactor, and decay heat is decreasing but still high. This phase represents 5% of the plant operating cycle.

- OCPs 4 and 5 cover the remaining 92% of the plant operating cycle during which decay heat is slowly decreasing. A 1x4 configuration (one recently discharged assembly surrounded by four low power assemblies) was used throughout the study for the high-density pool loading cases, although the reference plant uses a more favorable 1x8 configuration.

For each OCP, the staff analyzed scenarios in which combinations of damage state (no leak, small leak, or moderate leak) mitigation (effective or ineffective) and fuel loading (high-density or low-density) were evaluated. The analyses showed:

- No radiological release occurred for the most likely, no-leak, damage state even with high-density fuel loading. Coolant losses due to seismic induced sloshing and boil-off were small. Even during OCP 1 the fuel remained covered with 15 feet of water after 72 hours.

- For the small leak damage state, no releases occurred for any operating cycle phase for mitigated high-density or low-density fuel loading scenarios. This was not the case for unmitigated, high-density or low-density loadings during OCPs 1, 2 and 3. For these scenarios timely deployment of 10 CFR 50.54(hh)(2) spent fuel pool makeup was required to prevent release. If left unmitigated, radiological release would begin within 40 to 60 hours of the seismic event.

- A significant advantage of the low-density loading was that hydrogen deflagrations were not predicted for any of the small leak unmitigated cases, whereas deflagrations and substantially higher releases were predicted for unmitigated high-density pool loading using the 1x4 configuration. In separate sensitivity analyses the staff evaluated the 1x8 high-density configuration actually used by the reference plant and found that this configuration resulted in major reductions in heat-up rates and maximum fuel temperatures, no deflagrations, and no radiological releases. Such loadings should be evaluated further to determine whether they are broadly beneficial to BWR and PWR spent fuel pools.

- For the moderate leak damage state, mitigation was effective in preventing release during OCPs 2 and 3 for both high-density and low-density fuel loading but not effective during the first week of the plant operating cycle (OCP 1).

- For the moderate leak damage state, mitigation was not necessary to prevent release during OCPs 4 and 5. Both high-density and low-density fuel loadings were air-coolable.

<u>Mitigation and Human Reliability Analyses</u>

For scenarios in which mitigation is successful, the study credited the use of 10 CFR 50.54(hh)(2) equipment and the use of both on-site capabilities and off-site resources to extend the utilization of this equipment until on-site capabilities are recovered. None of these capabilities were credited in the assessment of unmitigated scenarios.

The SFPS shows that the 10 CFR 50.54(hh)(2) equipment is capable of mitigating most scenarios. However, such equipment may not be available due to deployment for other purposes. Order EA-12-049, while not evaluated in the SFPS, specifically requires a capability to maintain spent fuel pool cooling. Responses to the order should be evaluated to ensure that they provide the capability to deal with leakage on the order of that expected in a beyond-design-basis seismic event such as that considered in the SFPS.

To gain insights into the likelihood that successful mitigation could be accomplished, the staff performed a simplified HRA. Unfortunately the analysis does not address complexities expected to affect human performance during a severe seismic event, and it does not account for the effects of uncertainties. Therefore, the calculated human error probabilities are optimistic and cannot be considered as realistic estimates of the probability of success. Nevertheless, the descriptive analysis of the actions required to implement mitigation provides a useful structure for understanding these activities.

Offsite Consequence Analyses

The SFPS shows that for all scenarios no early fatalities are predicted to occur. Even in the unmitigated scenarios, the predicted releases are neither fast enough nor large enough to significantly exceed offsite dose levels that can cause early fatalities. As modeled in the SFPS, personnel evacuation and hotspot relocation are always effective in preventing significant exposure to all individuals in the areas of most concern. A more thorough consideration of uncertainties could identify cases where evacuations are not as effective.

The SFPS also shows that average individual latent cancer fatality risk based on the linear no threshold dose response model is low for both high and low-density loadings without successful mitigation. These individual risks are dominated by long-term exposures to contaminated areas for which doses are small enough to be considered habitable. Use of alternate dose models results in significant reduction in the individual latent cancer fatality risk estimates.

The amount of land interdiction for the studied scenarios could be up to two orders of magnitude greater for the high-density loading configuration compared to the low-density loading configuration. However, successfully deployed mitigation in the high-density loading configuration is predicted to reduce the amount of land interdiction.

Overall, the SFPS provides a technically sound basis for the staff's conclusion that expedited transfer of spent fuel from the pool to dry cask storage does not provide a substantial safety enhancement for the reference plant. The study also provides approaches, tools, and insights for a broader evaluation of the consequences of severe seismic events on spent fuel pools of different design, and will be valuable in determining whether expedited transfers to DCSSs produce substantial safety benefit for U.S. BWRs and PWRs. We commend the staff for these accomplishments.

Sincerely,

/RA/

J. Sam Armijo
Chairman

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APPENDIX D: REGULATORY ANALYSIS AND BACKFITTING DISCUSSION TO DETERMINE THE SAFETY BENEFIT OF EXPEDITED TRANSFER OF SPENT FUEL AT A REFERENCE PLANT

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

ABBREVIATIONS AND ACRONYMS

ADAMS	Agencywide Documents Access and Management System
Bq	Becquerel
BLS	Bureau of Labor Statistics
BWR	boiling-water reactor
CDF	Core Damage Frequency
CFR	*Code of Federal Regulations*
CoC	certificate of compliance
Cs	cesium
DOE	U.S. Department of Energy
DSC	dry storage cask systems
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
FR	*Federal Register*
FTE	Full-Time Equivalent
ISFSI	independent spent fuel storage installation
LCF	latent cancer fatality
LERF	Large Early Release Frequency
LNT	linear no-threshold
LOOP	loss of offsite power
MACCS2	MELCOR Accident Consequence Code System, Version 2
MELCOR	not an acronym
NPV	Net Present Value
NRC	Nuclear Regulatory Commission
NTTF	Near-Term Task Force
OCP	operating cycle phase
OMB	Office of Management and Budget
PAG	protective action guides
PGA	peak ground acceleration
PRM	petition for rulemaking
RA	Regulatory Analysis
SFP	spent fuel pool
SRM	Staff Requirements Memorandum
USGS	United States Geological Survey

D.1 INTRODUCTION

This appendix, which is organized into five sections, presents the regulatory analysis and backfitting discussion to determine the safety benefit of expedited transfer of spent fuel at a reference plant:

- Section D.1 describes the nature of the problem and a clear statement of the objective of the proposed action.
- Section D.2 describes and clearly explains the alternative approaches considered.
- Section D.3 describes the attributes affected, the methodology used to evaluate benefits and costs, the analysis model, key data and assumptions, and results for the alternatives evaluated.
- Section D.4 presents the analytical results and findings including discussion of supplemental considerations, uncertainties in estimates, and results of sensitivity analyses on the overall costs and benefits.
- Section D.5 presents the preferred alternative and the basis for selection, discusses any decision criteria used, identifies and discusses the regulatory instrument to be used (as applicable), and explains the statutory basis for the action.

D.1.1 Statement of the Problem

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 spent fuel pools is safe and that the risk is appropriately low. These studies used simplified and sometimes bounding assumptions and models to characterize the likelihood and consequences of beyond-design-basis spent fuel pool accidents[46]. As part of NRC's post-9/11 security assessments, spent fuel pool 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 issued a new regulation in 2009, 10 CFR 50.54(hh)(2), that requires reactor licensees to develop and implement guidance and strategies intended, in part, to maintain or restore spent fuel pool cooling capabilities following certain beyond design basis events.

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 (73 FR 46204, August 8, 2008) 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 events in Japan following the March 2011 earthquake has rekindled public and industry interest in understanding the consequences from postulated accidents associated with high-density spent fuel pool storage and the relative benefits of low-density spent fuel pool storage. This study provides an analysis of the health and safety benefits, if any,

[46] An overview of previous studies is provided in section 10.2 to the Spent Fuel Pool Study.

from moving from high-density to low-density spent fuel pool storage at the reference plant. This appendix assesses the costs and benefits of this activity, and then, for illustrative purposes, assesses whether the benefits are cost justified and substantial enough to justify a backfit to impose these requirements on the reference plant.

In response to these recent events, the staff has determined that it should confirm that high-density spent fuel pool configurations continue to provide adequate protection, and assess whether any safety benefits (or detriments) would occur from expedited transfer of spent fuel to dry cask storage at the reference plant.

The purpose of this regulatory analysis is to help ensure that:

- Appropriate alternatives to regulatory objectives are identified and analyzed.
- No clearly preferable alternative is available to this action.
- The costs of implementation are justified by its effect on overall protection of the public health and safety.

D.1.2 Objective of Proposed Action

Following the March 2011 accident at the Fukushima Daiichi nuclear power plant in Japan that resulted after the Tohoku Earthquake and subsequent tsunami, several stakeholders submitted comments to the Commission and staff requesting that regulatory action be taken to require the expedited transfer of spent fuel stored in spent fuel pools to dry cask storage. The rationale was that transferring the spent fuel to dry storage would lessen the potential consequences associated with a loss of spent fuel pool coolant inventory by decreasing the amount of spent fuel stored in these pools and thereby decreasing the heat generation rate and radionuclide source term associated with the spent fuel in pool storage.

As directed by the Commission in SRM-SECY-12-0025, dated March 9, 2012, the staff has undertaken regulatory actions that originated from the NTTF recommendations to enhance reactor and spent fuel pool safety. On March 12, 2012, the staff issued Order EA-12-051, which requires that licensees install reliable means of remotely monitoring wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event. In addition, the staff issued Order EA-12-049 which requires that licensees develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities following a beyond design basis external event. Upon full implementation of these Orders, spent fuel pool safety will be significantly increased.

While the staff has concluded, based on previous studies without these enhancements, that both spent fuel pools and dry casks provide adequate protection of public health and safety, the staff has determined that it should confirm that spent fuel pools continue to provide adequate protection.

This analysis uses information contained within the “Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor” (Spent Fuel Pool Study or main document), to evaluate whether there is a benefit at the reference plant in the study to change from high- to low-density storage configurations in the spent fuel pool.

This analysis calculates the potential benefit per reactor year resulting from expedited fuel transfer by comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage. The comparison uses the initiating frequency and consequences from the Spent Fuel Pool Study as an indicator of any changes in the NRC's understanding of safe storage of spent fuel. The staff also used calculated results from previous spent fuel pool studies (i.e., NUREG-1353 and NUREG-1738) to extend the applicability of this evaluation to include other initiators, which could challenge spent fuel pool cooling or integrity.

D.2 IDENTIFICATION AND PRELIMINARY ANALYSIS OF ALTERNATIVE APPROACHES

This section presents the analysis of the alternatives that the NRC considered to meet the regulatory goals identified in the previous section. The NRC considered the regulatory baseline and one alternative to change this baseline as discussed below.

D.2.1 Alternative 1 – Regulatory Baseline – Maintain the Existing Spent Fuel Storage Requirements

This alternative reflects a Commission decision not to expedite the storage of spent fuel to dry cask storage, but to continue with NRC's existing licensing requirements for spent fuel storage. Under this alternative, spent fuel is moved into dry storage only as necessary to accommodate fuel assemblies being removed from the core during refueling operations. It also assumes that all applicable requirements and guidance to date have been implemented, but no implementation is assumed for related generic issues or other staff requirements or guidance that is unresolved or still under review.

The condition represented by this alternative is the storage of spent fuel in high-density racks[47] in the spent fuel pool, a relatively full spent fuel pool, compliance with all current regulatory requirements including those requirements associated with the following:

- 10 CFR 50.54(hh)(2) with respect to spent fuel configuration, and spent fuel pool preventive and mitigative capabilities,
- Order EA-12-051 that requires licensees to install reliable means of remotely monitoring wide-range spent fuel pool levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event, and
- Order EA-12-049 that requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities following a beyond design basis external event.

Furthermore, because spent fuel pools have a limited amount of available storage, even after licensees expanded their storage capacity using high-density storage racks, the current practice

[47] Most nuclear power plant spent fuel pools were originally designed for temporary storage of spent fuel. Starting in the 1980s, most pools were "re-racked" to utilize hardware that stores the assemblies in a more closely-spaced arrangement, thus allowing the storage of more assemblies in a high-density configuration.

of transferring spent fuel to dry storage in accordance with 10 CFR 72 is assumed to continue.[48] This alternative represents the baseline for estimating the incremental costs of alternative 2.

D.2.2 Alternative 2 – Low-density Spent Fuel Pool Storage

Under this alternative, older spent fuel assemblies[49] are expeditiously moved from spent fuel pool storage to dry cask storage beginning in year 2014 to achieve and maintain a low-density loading of spent fuel in the existing high-density racks within five years. The situation where the spent fuel pool is re-racked to a low-density rack configuration was not evaluated because such a situation would be inefficient in terms of regulatory benefit given that much of the benefit could be achieved by storing less fuel in the existing high-density racks. Because of the low-density spent fuel pool loading, this alternative has less longer-lived radionuclide inventory in the spent fuel pool, a lower overall heat load in the pool, and a slight increase in the initial water inventory that displaces the removed spent fuel assemblies. In certain situations, this additional water could delay the clearing of the baseplate, which would temporarily inhibit natural air circulation cooling under and up through the racks should the spent fuel pool completely drain.

Due to the uncertainty associated with the schedule for the availability of a spent fuel repository, the reference plant has a plan to have sufficient on-site storage capacity (in-pool capacity and dry storage) to store all of the spent fuel discharged over the operating life of the plant until sufficient repository capacity becomes available. As a result, the analyzed incremental increase in costs results primarily from the increase in net present value cost for the early transfer of spent fuel into dry storage resulting from the earlier capital costs for new casks and for a dry storage facility.

The staff recognizes that there are cost and risk impacts associated with the transfer of spent fuel from the spent fuel pool to cask storage after five years of cooling and during long-term cask storage[50]. These cost and risk impacts, if included, would reduce the overall net benefit in relation to the regulatory baseline. These effects (e.g., the added risks of handling and moving casks) were conservatively ignored to calculate the potential benefit per reactor year by only comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and its implementation costs.

D.3 <u>ESTIMATION AND EVALUATION OF VALUES AND IMPACTS</u>

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48 Maintenance of the existing spent fuel pool storage requirements would not limit the Commission's authority to add new requirements or update regulatory guidelines, as necessary. These actions and activities are a part of the regulatory baseline. However, these activities would be pursued as separate regulatory actions to resolve particular technical issues. Under this alternative, the NRC would take no action to require facilities to expedite the movement of spent fuel to achieve low-density loading in the spent fuel pool.

49 Older spent fuel assemblies are those that have been placed in the spent fuel pool to cool for at least five years after discharge from the reactor core.

50 EPRI report TR-1021049 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.

This section discusses the benefits and costs of each action alternative relative to the baseline. Ideally, all costs and benefits are converted into monetary values. The total of benefits and costs are then algebraically summed to determine for which alternative the difference between the values and impacts was greatest. However, in some cases the assignment of monetary values to benefits is not provided because meaningful quantification is not possible.

D.3.1 Identification of Affected Attributes

This section identifies the factors within the public and private sectors that the regulatory alternatives (discussed in section D.2) are expected to affect. These factors are classified as attributes using the list of potential attributes provided by the NRC in Chapter 5 of its Regulatory Analysis Technical Evaluation Handbook. The basis for selecting each attribute is presented below.

Affected attributes are the following:

- Public Health (Accident). This attribute measures expected changes in radiation exposure to the public due to changes in accident frequencies or accident consequences associated with the proposed action. The expected changes in radiation exposure are measured over a 50-mile radius from the plant site. The dose to the public is from reoccupation of the land and other activities following a severe accident. In addition, the dose to the public includes the occupational dose to workers for cleanup and decontamination of the contaminated land not onsite. The calculation for each alternative is made by subtracting the alternative from the regulatory baseline.

- Occupational Health (Accident). This attribute measures occupational health effects, for both immediate and long-term, associated with site workers because of changes in accident frequency or accident mitigation. Within the regulatory baseline, the short-term occupational exposure related to the accident occurs at the time of the accident and during the immediate management of the emergency and during decontamination and decommissioning of the onsite property. The radiological occupational exposure resulting from cleanup and refurbishment or decommissioning activities of the damaged facility to occupational workers are found within the long-term occupational exposure.

- Occupational Health (Routine). This attribute accounts for radiological exposures to workers during normal facility operations (i.e., non-accident situations). These occupational exposures occur during DSC loading and handling activities; ISFSI operations, maintenance, and surveillance activities; and preparing to ship the spent fuel offsite.

  This attribute represents an estimate of health effects incurred during normal facility operations so accident probabilities are not relevant. As is true of other types of exposures, a net decrease in worker exposures is taken as positive; a net increase in worker exposures is taken as negative. This exposure is also subject to the dollar per person-rem conversion factor.

- Offsite Property. This attribute measures the expected total monetary effects on offsite property resulting from the proposed action. Changes to offsite property can take various forms, both direct, (e.g. land, food, and water) and indirect (e.g. tourism). This attribute is typically the product of the change in accident frequency and the property consequences from the occurrence of an accident.

For the regulatory baseline, the offsite property costs are any property consequences resulting from any radiological release from the occurrence of an accident. Normal operational releases and those releases before severe accident are outside the scope of this regulatory analysis.

- Onsite Property. This attribute measures the expected monetary effects on onsite property, including replacement power costs, decontamination, and refurbishment costs, from the proposed action. There are two forms of onsite property costs that are evaluated. The first type is the cleanup and decontamination costs for the unit. The second type is the cost to replace the energy from the damaged or shutdown units.

- Industry Implementation. This attribute accounts for the projected net economic effect on the affected licensees to implement the mandated changes. Costs include procedural and administrative activities. Additional costs above the regulatory baseline are considered negative and cost savings are considered positive.

- Industry Operation. This attribute accounts for the projected net economic effect due to routine and recurring activities required by the proposed alternative on all affected licensees.

- NRC Implementation. This attribute accounts for the projected net economic effect on the NRC to place the proposed alternative into operation. NRC implementation costs and benefits incurred in addition to those expected under the regulatory baseline are included. Additional rulemaking, policy statements, new or expedited revision of guidance documents, and inspection procedures are examples of such costs.

- NRC Operation. This attribute accounts for the projected net economic effect on the NRC after the proposed action is implemented. Additional inspections, evaluations, or enforcement activities are examples of such costs.

Attributes that are not expected to be affected under any of the alternatives include the following: public health (routine), other government, general public, antitrust considerations, safeguards and security considerations, regulatory efficiency, improvements in knowledge, and environmental considerations.

D.3.2 Methodology Overview

This section describes the process used to evaluate benefits and costs associated with the proposed regulatory framework alternatives. The benefits (values) include desirable changes in affected attributes (e.g., monetary savings and improved security and safety). The costs (impacts or burdens) include undesirable changes in affected attributes (e.g., increased monetary costs, and decreased security and safety).

The regulatory analysis methodology is specified by various guidance documents. The two documents that govern the NRC's voluntary regulatory analysis process are NUREG/BR-0058, Revision 4, "Regulatory Analysis (RA) Guidelines of the U.S. Nuclear Regulatory Commission," dated September 2004 (RA Guidelines), and NUREG/BR-0184, “Regulatory Analysis Technical Evaluation Handbook,” dated January 1997 (RA Handbook). The regulatory analysis identifies all attributes impacted by the proposed alternative and analyzes them either quantitatively or qualitatively as described in the previous section.

For the quantified regulatory analysis, the NRC staff develops expected values for each cost and benefit. The expected value is the product of the probability of the cost or benefit occurring and the consequences that would occur assuming the event actually happens. For each alternative, the staff first determines the probabilities and consequences for each cost and benefit, including the year the consequence is incurred. The NRC staff then discounts the consequences in future years to the current year of the regulatory action. Finally, the NRC staff sums the costs and the benefits for each alternative and compares them.

After performing a quantitative regulatory analysis, the NRC staff adds attributes that could only be qualified[51]. Based on the qualification of each attribute, uncertainties, sensitivities, and the quantified costs and benefits, the staff makes a recommendation for each alternative. If the benefits, both quantified and qualified, are greater than the quantified and qualified costs, then the staff recommends the alternative should be implemented. If the benefits, both quantified and qualified, are less than the quantified and qualified costs, then the staff recommends the alternative should not be implemented.[52]

D.3.2.1 Analysis Model

This regulatory analysis measures the incremental impacts of the proposed regulatory framework alternative to the “continue with the existing regulatory framework” baseline, which reflects anticipated behavior in the event that the proposed alternatives are not adopted. Section D.4 presents the estimated incremental costs and savings of each alternative relative to continuing with NRC’s existing regulatory framework (alternative 1).

Key inputs into the analysis model are discussed in the following subsections.

D.3.2.1.1 Baseline for the Analysis

The regulatory baseline used in the analysis is to continue with NRC’s existing approach to spent fuel pool storage. This baseline assumes full compliance with existing NRC requirements, including current regulations and relevant orders. This is consistent with NUREG/BR-0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,” Rev. 4, which states that, “in evaluating a new requirement…, the staff should assume that all existing NRC and Agreement State requirements have been implemented.”

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[51] See NRC’s Regulatory Analysis Technical Evaluation Handbook, Section 4.3, “Estimation and Evaluation of Values and Impacts.”

[52] See NRC’s Regulatory Analysis Technical Evaluation Handbook, Section 4.5, “Decision Rationale.” Non-quantifiable attributes can only be factored into the decision in a judgmental way; the experience of the decisionmaker will strongly influence the weight that they are given. Qualitative attributes may be significant factors in regulatory decisions and should be considered, if appropriate.

D.3.2.1.2 Discount Rates

In accordance with guidance from the Office of Management and Budget (OMB) and NUREG/BR-0058, Rev. 4, present-worth calculations are used to determine how much society would need to invest today to ensure that the designated dollar amount is available in a given year in the future. By using present-worth, costs and benefits, regardless of when averted in time, are valued equally. Based on OMB guidance Circular No. A-4, September 17, 2003, present-worth calculations are presented using both 3 percent and 7 percent real discount rates. The 3 percent rate approximates the real rate of return on long-term government debt, which serves as a proxy for the real rate of return on savings. This rate is appropriate when the primary effect of the regulation is on private consumption. Alternatively, the 7 percent rate approximates the marginal pretax real rate of return on an average investment in the private sector, and is the appropriate discount rate whenever the main effect of a regulation is to displace or alter the use of capital in the private sector.

Although the NRC is not bound to follow OMB guidance, the NRC has voluntarily complied with the present-worth calculations developed in OMB Circular No. A-4 and has stated so in the RA Guidelines and RA Handbook.

D.3.2.2 Data

The data and assumptions used in analyzing the quantifiable impacts associated with the proposed alternative are discussed in this subsection. Information on attributes affected by the proposed regulatory framework alternatives were obtained from experienced NRC staff and other sources as referenced. The NRC staff considered the potential differences between the new requirements and the current requirements and has incorporated the proposed incremental changes into this regulatory analysis.

Available cost information is included in the backfitting analysis of the reference plant, which is provided for illustrative purposes. The NRC plans to use the insights from this analysis to inform a broader regulatory analysis to support decisionmakers in determining whether NRC's regulations should be changed to impose new generic requirements on all operating nuclear reactors.

D.3.2.2.1 Spent Fuel Pool Initiator Release Frequency

Section 1.5 of the Spent Fuel Pool Study provides an overview of contributors to spent fuel pool risk. The majority of spent fuel pool risk emanates from a loss of water from a sizeable leak in the spent fuel pool or a boil off in which operator action to inject water into the pool for an extended period is precluded. The release frequency from the spent fuel pool can then be characterized as the frequency of the initiator causing fuel uncovery multiplied by the probability of a release given fuel uncovery for the specific initiating event. The total release frequency is the sum of the frequency of releases from cask drops, seismic events, and other initiators. This value is given by:

$$F_{release} = \sum_i F_{initiator_i} \, x \, P_{release_i}$$

Where $F_{initiator}$ includes

F_{drop} = frequency of spent fuel uncovery from cask drops

$F_{seismic\text{-}bin\ 3}$ = frequency of spent fuel uncovery from seismic bin 3 event

$F_{seismic\text{-}bin\ 4}$	=	frequency of spent fuel uncovery from seismic bin 4 event
F_{other}	=	frequency of spent fuel uncovery from sources other than cask drops and seismic
$P_{release}$	=	probability of release given spent fuel uncovery for specific initiators

Source: Derived from Spent Fuel Pool Study, section B.4.

The Spent Fuel Pool Study provides a detailed analysis of the consequences, for a particular site and a calculation of $F_{seismic}$ for seismic bin 3, a hazard exceedance frequency range provided in Table 4 of the Spent Fuel Pool Study and reproduced in Table 72.

Table 72 Seismic Bins and Initiating Event Frequencies

Bin No.	Bin Range (g)	Bin PGA (g)	Approximate Initiating Event Frequency (USGS 2008 model) (/yr)
1	0.05 - 0.3	0.12	$5.2x10^{-4}$
2	0.3 - 0.5	0.4	$2.7x10^{-5}$
3	0.5 - 1.0	0.7	$1.7x10^{-5}$
4	> 1.0	1.2	$4.9x10^{-6}$

The Spent Fuel Pool Study did not analyze initiators that contribute to spent fuel pool risk other than for seismic events defined by seismic bin no. 3. However past studies, such as NUREG-1353 and NUREG-1738, evaluated additional events that could contribute to risk and consequences from spent fuel pool fires. Table 74 summarizes these initiating-event-class fuel uncovery frequencies. Uncovery frequencies taken from past studies depend on the assumptions stated in those studies. Additionally, seismic bin no. 4 is included by extrapolating the results of this study. For seismic bin no. 3 and bin no. 4 events, the uncovery frequency is the product of the initiating event frequency, ac power fragility, and the liner fragility.

The main report uses an ac power fragility value of 0.84 taken from NUREG-1150 as a surrogate for the conditional probability of normal spent fuel pool cooling and makeup not being available following a 0.7g earthquake. This simplifying assumption was made in light of the fact that the main report is not a probabilistic risk assessment (but rather a consequence analysis with probabilistic considerations) and that this value already approximates the upper bound value of 1.00. For the seismic bin no. 4 event, ac power fragility upper bound value of 1.00 was used in this regulatory analysis. In reality, the availability of normal spent fuel pool cooling and makeup would be a combination of the ac power fragility, the fragility of the actual equipment and its support equipment, and operator actions to recover spent fuel pool cooling capabilities using additional mitigation equipment and strategies implemented in response to Order EA-12-049, which were not considered in the main report. The modeling and consideration of these guidance and strategies to maintain or restore spent fuel pool cooling capabilities following a beyond design basis external event could result in a smaller value for spent fuel pool cooling and makeup failure conditional probability than the values used here and a resulting smaller initiating event fuel uncovery frequency.

Section 4.1.5 of the main report describes the results from the nonlinear finite element analysis to estimate the likelihood of leakage from concrete cracking and related spent fuel pool liner failure for the 0.7g earthquake. Figure 27 shows that the maximum membrane effective strain is about 3.7 percent. Based on this calculated liner strain for the 0.7g earthquake, a structural analysis of the pool estimates that the spent fuel pool in this study has a 90% probability of surviving the 0.7g earthquake with no liner leakage (or conversely, a 10% probability of damaging the liner such that leakage will occur). As a result, a liner fragility value of 0.1 is used for the seismic bin no. 3 initiating event. For the seismic bin no. 4 initiating event (i.e., 1.2g earthquake), a comparable structural analysis was not performed to determine the liner fragility value. As detailed in section 4.1.1 of the main report, the specific conditions for liner failure vary according to site conditions and spent fuel pool design. NUREG-1353 predicted the likelihood of liner failure from all potential earthquakes to be between about two and six times in a million years. NUREG-1738 predicted the likelihood of liner failure from all potential earthquakes to be between about two times in a million years and two times in 10 million years. Because a documented liner fragility value for a 1.2g earthquake for the reference plant is not readily available, a conservative bounding approach was used. A liner fragility value of 1.00 is used in this regulatory analysis for the best estimate, even though a realistic analysis may be able to justify a value a factor of 2 or more lower.

Past studies have reached generally similar conclusions about the relative contribution to risk from the seismic initiating events considered. Table 73 Frequency of Spent Fuel Pool Fuel Uncovery for Seismic Events summarizes the impact of the above modeling assumptions when comparing the seismic initiating event fuel uncovery frequencies from previous spent fuel pool accident regulatory analyses.

Table 73 Frequency of Spent Fuel Pool Fuel Uncovery for Seismic Events

Reference	Spent Fuel Pool Fuel Uncovery (per reactor-year)	Percent Increase in Fuel Uncovery Frequency Value
NUREG-1353 (1989) (BWR, best estimate)[1]	$7x10^{-6}$	(10%)
NUREG-1738[2]	$2x10^{-6}$	315%
This regulatory analysis[3]	$6.3x10^{-6}$	100%

1. This number was not multiplied by the stated conditional probability of having a zirconium fire of 0.25.
2. NUREG-1738 presented results for the two different seismic hazard models in wide use at the time (the Electric Power Research Institute and Lawrence Livermore National Labs models). The larger of the two values is listed above.
3. The initiating event frequency values are from Table 72. The likelihood of fuel uncovery is a product of initiating event frequency (e.g., $1.6x10^{-5}$ for seismic bin no. 3), ac power fragility (0.84), and liner fragility (0.1). For seismic bin no. 4, the likelihood of fuel uncovery is a product of initiating event frequency ($4.9x10^{-6}$), ac power fragility of 1.0, and a liner fragility of 1.0 (i.e., 100-percent likelihood of ac power and pool liner failure).

As discussed in the SFPS report, the study was performed for a specific site and for a specific initiating event. Once fuel in the pool has become uncovered, it may still be coolable from natural circulation of air, depending on the amount of decay heat and the amount of cooling. In section 12.1 of the main report, the fuel is estimated to be air coolable for at least 72 hours for all but roughly 10 percent of the operating cycle. Factors affecting this value include the amount of fuel in the pool, its configuration, burnup, geometry of the fuel racks, etc. A partial draindown event with channeled fuel could impede airflow. In this case with no natural circulation of air through the racks, the cooling of the fuel by the spray flow would be the only effective cooling mechanism until the decay heat of the fuel is reduced.

For the seismic bin no. 4 event, the spent fuel is assumed to retain an air coolable geometry following this event that causes a moderate to large crack in the pool and results in full pool draindown. Information provided in NUREG/CR-5176 (Prassinos et al, 1989), which concludes that there is high confidence that spent fuel pool racks are sufficiently robust to remain generally intact with their fuel channels open supports this assumption. Furthermore, prior studies conclude that severe earthquakes are not expected to result in catastrophic failure of spent fuel pool structural walls and floor or fuel racks. Section 4.2 of this study cites median fragility for the reactor building of about 1.6g. However, the main report did not perform a structural analysis to verify that the reference plant spent fuel and racks retain their structural integrity and air-coolable geometry following a 1.2g peak ground acceleration seismic event. Given the uncertainties involved, a bounding approach was used to evaluate the sensitivity of assuming the spent fuel is not air-coolable following a seismic bin no. 4 earthquake that causes a rapid draindown of the spent fuel pool. This was done by assuming a value of 1.0 for the high estimate of the conditional probability of release for the seismic bin no. 4 unsuccessful mitigation event.

For the cask drop event, spent fuel is assumed to retain an air coolable geometry because a postulated cask drop accident would most likely affect the fuel pool floor in the cask loading area. The overhead crane used to move the casks is designed to meet single failure proof criteria, and has interlocks and administrative controls that limit the motion of the crane over the spent fuel pool to the cask loading area, where no fuel is stored. Although improbable, crane failure is more likely to occur during hoisting operations when many components contribute to holding the cask than during translational motion when the hoist holding brakes are set. The hoisting activities occur over the cask loading area, and, in that location, the cask, if dropped, could have sufficient potential energy to damage the spent fuel pool floor. . However, the main report did not perform a structural analysis to verify that the reference plant spent fuel and racks retain their structural integrity and air-coolable geometry following a cask drop event. Given the uncertainties involved, a bounding approach was used to evaluate the sensitivity of assuming the spent fuel is not air-coolable following a cask drop accident. This was done by assuming a value of 1.0 for the high estimate of the conditional probability of release for the cask drop unsuccessful mitigation event.

To calculate the total release frequency, the uncovery frequencies are multiplied by the conditional probability of release for each initiating event class. The conditional probability of release depends on the fraction of the operating cycle where the fuel is not air coolable. For the seismic bin no. 3 event analyzed in the SFPS, this was calculated to be the ratio of 60/730 days or 8.2% of the operating cycle. See Section 5.6.3 of the main document for further discussion. For the non-seismic and non-cask drop events taken from previous studies, the nature of the events may lead to a situation similar to a partial draindown where the rack baseplate is not cleared and airflow is impeded. For these events, the conditional release probability is assumed to be 100%.

When mitigation is credited, this study found that successful mitigation decreased the conditional probability by a factor of 19 for the seismic bin no. 3 event analyzed using mitigation measures required under 10 CFR 50.54 (hh)(2). The main report does not consider the post-Fukushima mitigation equipment and mitigation strategies for their use required under Order EA-12-049 and being implemented by the operating plants that are intended to increase the likelihood of restoring or maintaining power and mitigation capability during severe accidents. For the purposes of this regulatory analysis, it was assumed that successful mitigation decreased the conditional probability by a factor of 19 for all initiating events as determined in the main report. In reality, the effectiveness of post-Fukushima improvements to severe accident mitigation measures will depend on a variety of factors, which the SFPS did not consider, and which are expected to be more effective than what is assumed here. Although the likelihood of successful mitigation deployment is uncertain.

Table 74 summarizes the initiating event class fuel uncovery frequencies, the conditional probability of release, and the total release frequency with and without mitigation.

Table 74 Release Frequencies for Spent Fuel Pool Initiators

Spent fuel loading configuration		1x4		1x4	
Initiating Event Class	**Initiating Event Fuel Uncovery Frequency (per r-yr)**	**Conditional Probability of Release (Unsuccessful mitigation)**	**Release Frequency (Unsuccessful mitigation) (per r-yr)**	**Conditional Probability of Release (successful mitigation)**	**Release Frequency (successful mitigation) (per r-yr)**
Seismic bin no. 3 event	1.4×10^{-6} (3)	8.2%	1.18×10^{-7}	0.43% (4)	6.18×10^{-9}
Seismic bin no. 4 event	4.9×10^{-6} (3)	8.2% – 100%	4.03×10^{-7} – 4.9×10^{-6}	0.43% (4)	2.12×10^{-8} – 2.58×10^{-7}
Cask / heavy load drop	2×10^{-7} (2)	8.2% – 100%	1.64×10^{-8} – 2×10^{-7}	0.43% (4)	8.65×10^{-10} – 1.05×10^{-8}
LOOP – severe weather	1×10^{-7} (2)	100%	1.00×10^{-7}	0.43% (4)	5.26×10^{-9}
LOOP – other	3×10^{-8} (2)	100%	3.00×10^{-8}	0.43% (4)	1.58×10^{-9}
Internal fire	2×10^{-8} (2)	100%	2.00×10^{-8}	0.43% (4)	1.05×10^{-9}
Loss of pool cooling	1.5×10^{-8} (1)	100%	1.50×10^{-8}	0.43% (4)	7.89×10^{-10}
Loss of coolant inventory	3×10^{-9} (2)	100%	3.00×10^{-9}	0.43% (4)	1.58×10^{-10}
Inadvertent aircraft	3×10^{-9} (2)	100%	3.00×10^{-9}	0.43% (4)	1.58×10^{-10}
Missiles – general	2.5×10^{-9} (1)	100%	2.50×10^{-9}	0.43% (4)	1.32×10^{-10}
Missiles - tornado	1×10^{-9} (2)	100%	1.00×10^{-9}	0.43% (4)	5.26×10^{-11}
Pneumatic seal failures	n/a (5)				
Total			7.11×10^{-7} – 5.39×10^{-6}		3.74×10^{-8} – 2.84×10^{-7}

1. Values from NUREG-1353. These numbers were multiplied by the stated conditional probability of having a zirconium fire of 0.25.
2. Values from NUREG-1738
3. Initiating event frequency values from Spent Fuel Pool Study, Table 4. The likelihood of fuel uncovery is a product of initiating event frequency (e.g., 1.6×10^{-5} for seismic bin no. 3), ac power fragility (0.84), and liner fragility (0.1). For seismic bin no. 4, the likelihood of fuel uncovery is a product of initiating event frequency (4.9×10^{-6}), ac power fragility of 1.0, and a liner fragility of 1.0 (e.g., 100-percent likelihood of ac power and pool liner failure).
4. The conditional probability of release with successful mitigation with deployed 50.54(hh)(2) equipment is the quotient of OCP probability (60/730 or 8.2%) divided by the mitigation benefit in reducing the release likelihood (factor of 19). See Section 5.6.3 of the main document for further discussion. Additional mitigation equipment and mitigation strategies under Order EA-12-049 would further enhance the likelihood of successful mitigation, thereby further reducing the value for the conditional probability of release with successful mitigation.
5. As discussed in Table 3 of the main report, the reference plant has gates with mechanical seals to prevent leakage. These seals are kept under pressure by passive mechanical means (i.e., do not depend on air pressure, ac power, or dc power). Therefore, pneumatic seal failures are not applicable for the reference plant.

Based on this information, the values used in this regulatory analysis for $F_{release}$ is are summarized in Table 75.

Table 75 Spent Fuel Pool Release Frequency Estimates

Parameter	Unsuccessful mitigation			Successful mitigation		
	Low	Best	High	Low	Best	High
$F_{release}$	7.11×10^{-7}		5.39×10^{-6}	3.74×10^{-8}		2.84×10^{-7}

These release frequency values are subject to the assumption of unsuccessful deployment of mitigation and the other assumptions contained in this analysis and those stated in Table 3 of the main report. A comparison of the release frequencies (total and delta) used in this regulatory analysis to the release frequencies used for only seismic bin no. 3 in the Spent Fuel Pool Study is provided in Table 76.

Table 76 Release Frequency Comparison Between Inclusion of All Initiator Event Classes to the Seismic Bin No. 3 Event

Mitigation Case	Release Frequency for All Initiator Events Classes (per r-yr)	Release Frequency for the Seismic Bin No. 3 Event (per r-yr)	Net Percent Increase in Release Frequency
Unsuccessful Mitigation	$7.11 \times 10^{-7} – 5.39 \times 10^{-6}$	1.18×10^{-7}	505% – 4489%
Successful Mitigation	$3.74 \times 10^{-8} – 2.84 \times 10^{-7}$	6.18×10^{-9}	505% – 4489%
Delta change	$6.74 \times 10^{-7} – 5.11 \times 10^{-6}$	1.11×10^{-7}	505% – 4489%

D.3.2.2.2 Duration of On-site Spent Fuel Storage Risk

The reference plant operating license expires in 2034. For this analysis, it is assumed that the plant operates through the term of its operating license and that the licensee continues to store spent fuel in the pool following commercial operation[53] to allow the spent fuel to cool sufficiently before placing into dry storage. For all cases analyzed, it was assumed that spent fuel stored in the spent fuel pool is susceptible to the risk of spent fuel fires for up to one year after permanent cessation of operations.

D.3.2.2.3 Cost/Benefit Inflators

The consequences for some attributes are estimated based on the values published in the NRC Regulatory Analysis Handbook. Within the NRC Regulatory Analysis Handbook, the information in relation to severe reactor accident consequences is provided in previous year dollars. To evaluate the costs and benefits consistently, the consequences are inflated. The most common inflator is the Consumer-Price Index for all urban consumers (CPI-U), developed by the U.S. Department of Labor, Bureau of Labor Statistics. Using the CPI-U, the previous year dollars were converted to the year 2012. The formula to determine the amount in 2012 dollars is

$$\frac{\text{CPIU}_{2012}}{\text{CPIU}_{\text{Base Year}}} * \text{Consequence}_{\text{Base Year}} = \text{Consequence}_{2012}.$$

Values of CPI-U used in this regulatory analysis are summarized in Table 77.

Table 77 Consumer Price Index – All Urban Consumers Inflator

Base Year	CPI-U Inflator for Year 2012
2005	1.1756
2006	1.1389

53 Decommissioning of the unit must be completed within 60 years of permanent cessation of operations under 10 CFR 50.82, “Termination of License.” Completion of decommissioning beyond 60 years will be approved by the Commission only when necessary to protect public health and safety.

Base Year	CPI-U Inflator for Year 2012
2007	1.1073
2008	1.0664
2009	1.0702
2010	1.0529
2011	1.0207

Source: www.bis.gov/data/inflation_calculator.htm

D.3.2.2.4 Dollar per Person-Rem Conversion Factor

Using the dollar value of the health detriment and a risk factor that establishes the nominal probability for stochastic health effects attributable to radiological exposure (fatal and non-fatal cancers and hereditary effects) provides a dollar per person-rem of $2,000, rounded to the nearest thousand, according to NUREG-1530, “Reassessment of NRC’s Dollar per Person-Rem Conversion Factor Policy,” dated December 1995.

The NRC currently use a value of statistical life (VSL)[54] of $3 million based on NUREG-1530, and a cancer risk factor of 7.0 x 10^{-4}, which is a reduction to the closest significant digit of a recommendation by the International Commission on Radiation Protection (ICRP) in Publication No. 60. Therefore, the dollar per person-rem is equal to $3 million times 7.0 x 10^{-4} rounded to the nearest thousand (due to uncertainties) or $2,000.

D.3.2.2.5 Onsite Property Decontamination, Repair, and Refurbishment Costs

Spent fuel pool accident risks have significant contributions from onsite property monetary losses (e.g., repair and refurbishment) and plant decontamination. The risk dominant accident sequences involve the failure of the pool due to seismic or load drop events resulting in the loss of pool integrity. This scenario results in loss of spent fuel pool water inventory, zircaloy cladding fire initiation with propagation through the spent fuel assemblies stored in the pool, and an uncontrolled radiological release from the reactor building. The NRC assumes that, based on the current regulatory framework, with insights from the Fukushima Dai-ichi accident, that onsite property would be radiologically affected in the following way. The consequences of a spent fuel fire are expected to be similar to the Category II accident as defined in NUREG/CR-5281, section 3.2.4. Based on this reference, the cleanup and decontamination costs are estimated to be approximately $165 million (1983 dollars) and the cost for permanent disposal of the damaged fuel is $26 million (1983 dollars). Using Table C.95 from the RA Handbook, the pool repair to is expected to cost $72 million (1983 dollars). Adjusting these estimated costs using the CPI-U inflator formula and using a multiplier of three to model the high estimate and a divider of two to model the low estimate results in the values provided in Table 78.

[54] The value of a statistical life (VSL) is the monetary value of a mortality risk reduction that would prevent one statistical (as opposed to an identified) death (Jones-Lee, 2004). The VSL is a key component in the calculation of the dollar per person-rem value, which is the product of the VSL multiplied by a risk coefficient.

Table 78 Onsite Property Decontamination, Repair, and Refurbishment Costs

Onsite Property Cost Element	1983 dollars			2013 dollars		
	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
Cleanup and decontamination	$165,000,000	$495,000,000	$82,500,000	$371,250,000	$1,113,750,000	$185,625,000
Repair Pool	$72,000,000	$216,000,000	$36,000,000	$162,000,000	$486,000,000	$81,000,000
Disposal of damaged fuel	$26,000,000	$78,000,000	$13,000,000	$58,500,000	$175,500,000	$29,250,000
Total	$263,000,000	$789,000,000	$131,500,000	$591,750,000	$1,775,250,000	$295,875,000

D.3.2.2.6 Replacement Energy Costs

Replacement energy costs are the costs for replacing the energy from the nuclear power plant due to a plant shutdown to install required equipment or due to an accident.[55] The NRC assumes that replacement energy costs would be required until onsite decontamination and repair efforts are completed or the unit is retired.

The NRC assumes that licensees engage in power purchase agreements (PPA)[56] to economically purchase replacement power. A PPA is a legal contract between an electricity generator (licensee) and a power purchaser. The NRC assumes that a licensee will not be able to replace the power through other generation for seven years and would have to buy power from the market. Although not all licensees may have PPAs, the licensee will still replace the lost energy any time that the nuclear power plant is not operating to meet its electrical power supply obligations. The NRC assumes that after 7 years, the onsite decontamination and repair efforts are completed or the unit is retired and other power sources will be developed to replace the unit’s lost electrical generation capability.

For the replacement energy cost calculation in this regulatory analysis, the NRC assumes that the reference plant is located on a multi-unit site. For the high estimate case, the NRC assumes that replacement energy is purchased for both the accident unit and the co-located unit at the site.

D.3.2.2.7 Occupational Worker Exposure (Accident)

There are two types of occupational exposure related to accidents: short-term and long-term. The first occurs at the time of the accident and during the immediate management of the emergency. The second is a long-term exposure, presumably at significantly lower individual rates, associated with the cleanup and refurbishment or decommissioning of the damaged facility. The value gained in the avoidance of both types of exposure is conditioned on the change in frequency of the accident's occurrence.

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55 The replacement energy cost is only the cost to buy the energy for production on the market. Therefore, the cost would be the cost of buying the cheapest energy. These estimates do not include transmission or distribution costs.

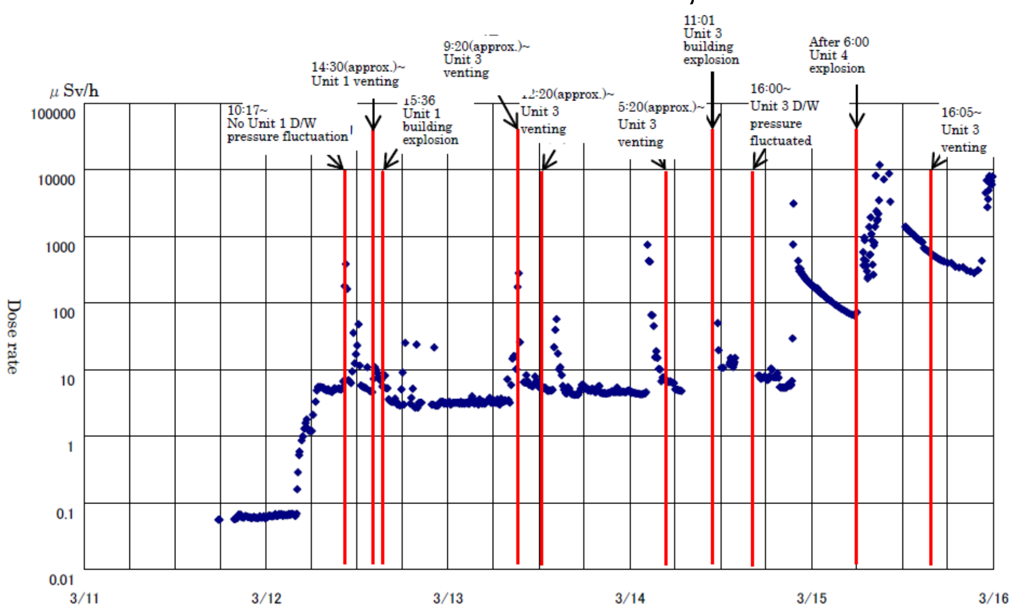
56 A power purchase agreement is a contract between two parties, one who generates electricity for the purpose of sale (the seller) and one who is looking to purchase electricity (the buyer). The PPA defines all of the commercial terms for the sale of electricity between the two parties, including when the project will begin commercial operation, schedule for delivery of electricity, penalties for under delivery, payment terms, and termination.

The experiences at the Three Mile Island Unit 2 (TMI-2), the Chernobyl, and the Fukushima nuclear power plants illustrated that significant occupational exposures could result from performing activities outside the control room during a power reactor accident. At TMI-2, the average occupational exposure related to the incident was approximately 1.0 rem, with a collective dose of 1,000 person-rem occurring over a 4-month span, after which time occupational exposure approached pre-accident levels. For Chernobyl, the average dose for persons closest to the plant was 3.3 person-rem (RA Handbook p. 5.30), yielding an average value of 3,300 person-rem.

After the Fukushima unit 1 building explosion on March 12, 2011, the unit 3 building explosion on March 14, and the unit 4 building explosion and the exposure of the unit 2 reactor fuel rods on March 15, radioactive materials were release into the environment and surrounding areas of the Fukushima Dai-ichi nuclear power plant. Measurement and evaluation of radiation exposure levels for workers engaged in emergency work at the Fukushima Daiichi NPS have been implemented continuously since the Tohoku Earthquake.

As shown in Figure 142, the dose rate in the vicinity of the main gate at the Fukushima Dai-ichi site near the time of the Unit 4 explosion varied between 20 mrem and 1.0 rem per hour (between 200 and 10,000 µSv per hour).

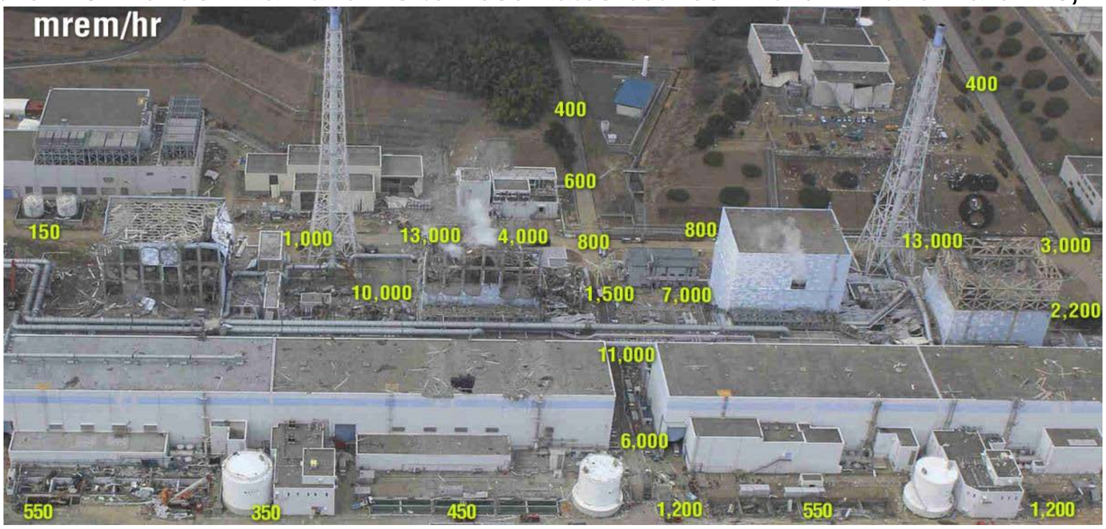
Figure 142: Dose Rate in Vicinity of Fukushima Daiichi Nuclear Plant Site Main Gate between March 11 and March 16, 2011



Source: Fukushima Nuclear Accident Analysis Report p. 371.

On March 22 and 23, surveys of the airborne radioactivity and dose rates around the Fukushima Daiichi site were collected and documented. The dose rates are shown on Figure 143.

Figure 143: Fukushima Daiichi Site Dose Rates between March 22 and March 23, 2011



Source: INPO 11-005, p 41

The distribution of total monthly exposure for workers engaged in radiation work at the Fukushima Daiichi NPS for the first three months following the March 2011 accident is provided in Table 79.

Table 79 Average Accident Occupational Exposure at Fukushima Dai-ichi Nuclear Power Plant from March to May 2011

Total Radiation Exposure (mSv)	Number of Plant Workers Exposed		
	March 2011[1]	April 2011[2]	May 2011[3]
≥ 250	6	0	0
200 - 249	2	0	0
150 - 199	14	0	0
100 - 149	77	0	0
50 - 99	309	3	0
20 - 49	859	81	19
10 - 19	1041	310	144
< 10	1434	3214	2854
Total number of workers	3742	3608	3017

1. Maximum March 2011 occupational exposure was 670.4 mSv.
2. Maximum April 2011 occupational exposure was 69.3 mSv.
3. Maximum May 2011 occupational exposure was 41.6 mSv.
4. One mSV is equal to 0.1 rem.

Source: Wada et al, Occupational and Environmental Medicine, 2012 August; 69(8): p. 600.

To estimate the monthly total occupational radiation exposure received by all workers, a high estimate, best estimate, and low estimate were calculated based on the maximum category value, the midpoint category value, and the first quartile category value. The results are tabulated in Table 80.

Table 80 Estimated Immediate Accident Occupational Monthly Exposure at Fukushima

Radiation Exposure (mSv)	Best Estimate Category Radiation Exposure (mSv)			High Estimate Category Radiation Exposure (mSv)			Low Estimate Category Radiation Exposure (mSv)		
	March 2011	April 2011	May 2011	March 2011	April 2011	May 2011	March 2011	April 2011	May 2011
≥ 250	460.2			670.4			355.1		
200 - 249	224.5			249			212.25		
150 - 199	174.5			199			162.25		
100 - 149	124.5			149			112.25		
50 - 99	74.5	69.3		99	69.3		62.25	62.25	
20 - 49	34.5	34.5	34.5	49	49	41.6	27.25	27.25	27.25
10 - 19	14.5	14.5	14.5	19	19	19	12.25	12.25	12.25
< 10	5	5	5	10	10	10	2.5	2.5	2.5
Total Monthly Dose	90,200	23,600	17,000	125,600	42,200	32,100	72,500	14,200	9,400
Avg Worker Dose	24.1	6.5	5.6	33.6	11.7	10.6	19.4	3.9	3.1

The immediate accident occupational exposure for a spent fuel pool accident shown in Table 81 is estimated based on the Fukushima data and the following assumptions:

- The immediate accident period lasts for one year,
- The workforce during the immediate accident period is 3,700 workers, and
- The average worker radiation exposure remains constant at the May 2011 value from May 2011 through February 2012.

Table 81 Immediate Accident Occupational Exposure for a Spent Fuel Pool Fire

Case	Immediate Accident Occupational Exposure (averted person-rem)
Low Estimate	18,070
Best Estimate	28,380
High Estimate	48,880

After the immediate response to a spent fuel pool fire, a long process of cleanup and refurbishment or decommissioning will follow. The Fukushima Nuclear Accident Analysis Report states, "The average value for 5,128 people in April of 2012 was 1.07 mSv per worker due to decreasing trends in environment dose rates (p 415). The NRC assumes that the process of cleanup and refurbishment or decommissioning will begin one year after the accident and will take seven years to complete. During those seven years, the NRC assumes that each occupational worker at the damaged reactor site will be exposed to 1.07 mSv per month (0.107 rem per month) for the duration of the cleanup and refurbishment or decommissioning. Assuming the average value for 5,128 workers would remain for the duration yields a cumulative long-term occupational dose of 46,000 person-rem.

In NUREG/CR-5281, Jo et al. (1989) conducted what essentially amounted to a regulatory analysis of a non-reactor nuclear fuel cycle facility using the 1983 Handbook (Heaberlin et al. 1983) as guidance. The accidental occupational exposure was assumed to be similar to that from TMI-2, which is 4,580 person-rem.

As described in the RA Handbook (p 5.30), the DOE (1987) summarized results on the collective dose received by the populace surrounding the Chernobyl accident. Average dose equivalents of 3.3 rem per person, 45 rem per person, and 5.3 rem per person were estimated for residents within 3 km, between 3 km and 15 km, and between 15 km and 30 km of Chernobyl, respectively (Mubayi et al. 1995, p. A-5). Assuming 1,000 workers and a 4.2 multiplier, an estimate radiation exposure of 14,000 person-rem results.

Site worker exposures following a spent fuel pool accident could be greater than that of a reactor core melt accident. This is because a spent fuel pool stores significantly more fuel assemblies than a reactor core. Additionally, radionuclides released during a spent fuel pool accident have longer half-lives (e.g., Cesium-137) than those that would be released during a reactor accident. Given the uncertainties in existing data and variability in severe accident parameters and worker response, Table 82 provides the long-term occupational dose used in this regulatory analysis to analyze spent fuel pool fires.

Table 82 Long-Term Accident Occupational Exposure for a Spent Fuel Pool Fire

Case	Immediate Accident Occupational Exposure (averted person-rem)
Low Estimate	4,580
Best Estimate	14,000
High Estimate	46,000

D.3.2.2.8 Long-Term Habitability Criteria

The long-term phase is the period following the seven-day emergency phase and is modeled for 50 years to calculate consequences from exposure of the average person. Radiation exposure during this phase is mainly from external radiation from trace contaminants that remain after the land is decontaminated, or in lightly contaminated areas where no decontamination was required. Internal radiation exposures may also occur during this period, including inhalation of resuspended radionuclides and ingestion of food and water with trace contaminants. Depending on the relevant protective action guides (PAGs) and the level of radiation, food, and water below a certain limit could be considered adequately safe for ingestion, and lightly contaminated areas could be considered habitable.

A long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be made by a number of local, state, and federal jurisdictions and would most likely 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 may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations.

Site-specific values are used to determine long-term habitability. For habitability, most states adhere to EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter. This habitability criterion was used in previous spent fuel pool studies, which used 4 rem in 5 years to represent these PAG levels (e.g., 2 rem in year one, followed by 0.5 rem each successive year). However, consistent with the location of the reference plant, the Spent Fuel Pool Study analysis utilizes the State of Pennsylvania habitability criterion of 500 mrem beginning in the first year (and each following year). The use of this long-term habitability criterion reduces the predicted long-term population doses and health effects and increases the costs associated with interdiction, decontamination, and condemnation.[57]

Given the uncertainties in which long-term habitability criterion would be used, Table 83 provides the long-term phase habitability criterion used in this analysis to analyze the consequences of spent fuel pool fires on public health (accident).

Table 83 Long-Term Habitability Criterion

Case	Long-Term Habitability Criterion	Protective Action Basis
Low Estimate	500 mrem annually	Pennsylvania dose limit to the public
Best Estimate	2 rem in the first year and 500 mrem each year thereafter	EPA intermediate phase PAGs
High Estimate	2 rem annually	EPA intermediate phase PAG: first year

Based on the average population dose for a release estimated using a sensitivity analysis, the public dose for the two EPA protective action bases was estimated by scaling population dose calculated using the Pennsylvania dose limit. The habitability criterion scaling factors used are provided in Table 84.

Table 84 Habitability Criterion Scaling Factors

	500 mrem	2 rem in the first year and 500 mrem each year thereafter	2 rem
Population Dose within 50 miles	100%	207%	278%
Total Population Dose	100%	165%	192%

The use of these habitability criteria also affects the values of offsite property damage used in this analysis. Certain metrics such as offsite property damage, the number of displaced individuals (either temporarily or permanently) and the extent to which such actions may be needed are inversely proportional to changes in collective dose resulting from changes in habitability criteria.

The impacts for alternate protective action levels were produced by examining the sensitivity analyses used to evaluate the effect of alternate protective action levels on land contamination, which were based on the results for a release from a high-density loading without credit for mitigation during OCP3. Scaling factors for different protective action levels were derived from this case. For a very large release that led to economic impacts beyond 50 miles, the sensitivity of the results within 50 miles to different protective action levels is less than the sensitivity of results beyond 50 miles. For significantly lower release magnitudes associated with the low density and successful mitigation cases, the scaling approach used can predict higher economic consequences within 50 miles than for the total. This implies that the economic impacts beyond 50 miles would be small relative to the economic impacts within 50 miles, and the total scaled

57 Interdiction and condemnation refer to the relocation of people from contaminated areas according to the habitability criterion. Interdiction is the temporary relocation of the affected population while decontamination, natural weathering, and radioactive decay reduce the contamination levels. Condemnation is the permanent relocation of the affected population if decontamination, natural weathering, and radioactive decay cannot adequately reduce contamination levels to habitability limits within 30 years.

economic impact is therefore set equal to the scaled economic impact within 50 miles. The economic consequences scaling factors used are provided in Table 85.

Table 85 Economic Consequences Scaling Factors as a Function of Habitability Criteria

	500 mrem	2 rem in the first year and 500 mrem each year thereafter	2 rem
Economic Consequences within 50 miles	100%	67%	56%
Total Economic Consequences	100%	43%	31%

These criteria provide a benchmark for understanding the nature and the extent of the relationship between collective dose, economic consequences, and habitability criteria following a severe spent fuel pool accident. These measures are subject to large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies, the loss of infrastructure on the general U.S. economy, or the details of how long-term protective actions would be performed.

D.3.2.2.9 Other Key Data

All monetized costs are expressed in 2012 dollars. Ongoing costs of operation related to the alternatives are assumed to begin in 2014 unless otherwise stated, and are modeled on an annual cost basis.

Estimates were made for one-time implementation costs. The staff assumes that these costs will be incurred in the first year of the analysis unless otherwise noted.

Estimates were made for recurring annual operating expenses. The values for annual operating expenses are modeled as a constant expense for each year of the analysis horizon. An annuity calculation was performed to discount these annual expenses to 2012 dollar values.

Reference plant site population data was projected to year 2011 using the latest version of the computer code SECPOP2000. SECPOP2000 uses 2000 census data and applies a multiplier value of 1.1051 from the U.S. Census Bureau to account for the average population growth in the United States from 2000 to 2011 as discussed in section 7.1.3 of the main report. No further population growth was evaluated in this appendix.

D.3.2.3 Assumptions

The Spent Fuel Pool Study is used to inform this analysis is a consequence study based on the occurrence of a postulated beyond-design-basis earthquake (with an estimated frequency of occurrence of one event in 60,000 years) to a selected U.S. Mark I BWR spent fuel pool with a unit-specific spent fuel pool. The Spent Fuel Pool Study major assumptions are listed in section 2 of the main document. Additional assumptions used for this analysis are discussed below. The costs presented in this analysis are based on estimates by the authors or cited documents. It should be noted that this is a generic cost estimate and should be used accordingly. Site-specific features may result in higher or lower costs than those estimated.

D.3.2.3.1 Projected Number of Outages and Spent Fuel Assemblies

The reference plant is on a 24-month refueling cycle and is estimated to require eleven refueling outages between 2012 and the end of its operating license in 2034. It is assumed that

284 assemblies are offloaded to the spent fuel pool during each outage based on information in section 5.1 of the main document. The full core of 764 assemblies is offloaded to the spent fuel pool upon operating license expiration.

The analysis for the reference plant is based on a high-density spent fuel pool inventory of 3,055 assemblies in a high-density 1x4 loading configuration, a number based on the pool capacity of 3,819 assemblies, reduced by 764 assemblies to accommodate a full core offload capability using the existing high-density racking. In a low density 1x4 with empties configuration, the spent fuel pool stores 852 assemblies. The number of spent fuel assemblies required up to operating license expiration is calculated based on the existing high-density spent fuel pool inventory, the number added from refueling outages, and the full reactor core inventory and is provided in Table 86.

Table 86 Number of Spent Fuel Assemblies Remaining through Operating License Expiration

Category	Inventory	Number	No. of spent fuel assemblies
Current spent fuel pool inventory	3,055	1	3,055
refueling	284	11	3,124
reactor core	764	1	764
		Total	6,943

D.3.2.3.2 Dry Storage Capacity

Three companies supply most of the dry storage technologies to U.S. commercial nuclear power plants. These companies are Holtec International, Inc. (Holtec), NAC International, Inc. (NAC), and Transnuclear, Inc. (Transnuclear). The dry storage cask systems[58] (DSCs) for all three companies are certified by the NRC for storage of high burnup spent fuel (i.e., burnups greater than 45 GWd/MTU), using both regional and uniform loading of spent fuel in the packages. A summary of a representative sampling of dry storage canisters commercially available to the reference plant for BWR fuel storage is provided in Table 87.

Table 87 Representative Sampling of Commercially Available BWR Spent Fuel Dry Storage Technology

Vendor Package	Fuel Type	Canister Type	Capacity (Assemblies)	Maximum Decay Heat Per Package[1] (kW)
Holtec HI-STORM	BWR	MPC-68	68	34
Holtec HI-STORM FW	BWR	MPC-89	89	46.36
NAC MAGNASTOR	BWR	87B	87	33
Transnuclear NUHOMS	BWR	61BTH	61	31.2
Transnuclear TN-68	BWR	Bolted	68	30

The maximum decay heat per assembly for uniform loading is estimated by dividing the package decay heat by the number of assemblies. The maximum decay heat per assembly under regional loading schemes will generally be higher than the maximum decay heat per assembly assuming uniform loading for a smaller number of assemblies. Cask certificates of compliance provide the specific maximum assembly decay heat limits for each storage location in the basket.
Source: EPRI TR-1025206, p. 2-11.

[58] The term dry storage cask system (DSC) includes dual-purpose canister based systems, dual-purpose casks, and storage-only dry storage casks and canister systems.

D.3.2.3.3 Fuel Assembly Decay Heat as a Function of Burnup and Cooling Time

As fuel assembly burnups increase, the decay heat of the fuel assembly (watts per assembly) increases. Decay heat also can vary significantly with initial enrichment and assembly irradiation parameters. Spent fuel burnups have gradually increased since the 1990s with average BWR burnups about 43 GWd/MTU and range between 40 and 50 GWd/MTU. Spent fuel assembly average decay heat for a 40 GWd/MTU BWR assembly that has cooled for five years is approximately 360 watts/assembly. The average decay heat for a 50 GWd/MTU assembly that has cooled for five years is approximately 520 watts per assembly (EPRI TR-1021049, p. 2-3, Regulatory Guide 3.54). The average BWR spent fuel assembly that has cooled for five years is approximately 410 watts/assembly.

Table 88 Canister Storage Capacity Based on Heat Rate Limitations

Vendor Package	Capacity (Assemblies)	Maximum Decay Heat Per Package1 (kW)	Max. Capacity based on decay heat		
			410w per assembly	520w per assembly	% Additional Canisters
Holtec HI-STORM	68	34	68.00	65.38	4.0%
Holtec HI-STORM FW	89	46.36	89.00	89.00	0.0%
NAC MAGNASTOR	87	33	80.49	63.46	37.1%
Transnuclear NUHOMS	61	31.2	61.00	60.00	1.7%
Transnuclear TN-68	68	30	68.00	57.69	17.9%

Based on the average BWR spent fuel assembly that emits 410 watts after it has cooled for five years, Table 88 shows that all of the dry storage canisters can be filled to capacity with the exception of the NAC MAGNASTOR, without exceeding the maximum decay heat per package rating, subject to restrictions on loading pattern. For 50 GWd/MTU assemblies that emit 520 watts after they have cooled for five years, fewer assemblies can be stored in a cask to ensure that it does not exceed the maximum decay heat rating. The number of additional dry storage casks required depends on the vendor package selected and range between no additional canisters to almost 40% additional canisters. Additional DSCs, which are required because of high heat load, are estimated in this appendix. For this regulatory analysis, the Transnuclear TN-68 dry casks are evaluated because the reference plant's ISFSI for dry cask storage utilizes the TN-68 cask design as discussed in section 1.3 of the main document. The currently approved minimum cooling time for fuel stored in the TN-68 dry casks is seven years (10 years for some fuel types), and Transnuclear would need to demonstrate, in an amendment request, that spent fuel that was cooled for a shorter period can be stored safely. The costs for Transnuclear to prepare such an amendment request and for the NRC review are not included in this regulatory analysis. The methodology used to estimate the capacity of the DSCs for spent fuel that has cooled for five years is subject to uncertainties resulting from decay heat and loading pattern restrictions. As a result, the actual DSC capacity may be higher or lower than those estimated.

D.3.2.3.4 Dry Storage Upfront Costs

Upfront costs include engineering, design, and licensing costs; equipment costs; construction costs; and start up and testing costs. Each of these cost components are further described in EPRI TR-1021048, "Industry Spent Fuel Storage Handbook." As noted in EPRI TR-1025206, "Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage after Five Years of Cooling, Rev. 1," the independent spent fuel storage installation

(ISFSI) upfront costs vary widely from site to site and the upfront costs for those in operation vary from several million to tens of millions of dollars. (EPRI TR-1025206, p. 2-23) Values for upfront costs were estimated based on two publically available cost estimates that identified the specified number of DSC to be stored. The estimate amortized upfront costs for each site is provided in Table 89.

Table 89 Amortized DSC Upfront Costs

ISFSI Facility	Upfront Cost Estimate (base year)	Upfront Cost Est. (2012 $)	DSC Storage Capacity	Attributed Upfront Cost per DSC (2012 $)
Monticello	$21.5 million (2005 $)	$25,275,400	30	$842,500
Pilgrim	$22 million (2006$)	$25,055,800	53	$472,800
Average (Best Estimate)		$25,165,600		$657,700

D.3.2.3.5 Incremental Costs Associated with Earlier DSC Purchase and Loading

Incremental costs are the costs associated with the purchase and loading of DSCs on a periodic basis. These costs include the capital costs for the DSC and the loading costs for the storage systems. The unit cost estimates used in this analysis are provided in Table 90. These cost estimates are based on the DSC unit costs that EPRI used for a Generic Interim Storage Facility (EPRI TR-1018722) and documented in EPRI TR-1025206. Operating nuclear power plants sites may experience incremental DSC purchase and loading costs that are higher or lower than the amount assumed in this analysis.

Table 90 Incremental Unit Cost Estimates

Item	Unit Cost (Constant $2012)
Canister	$780,000
Concrete overpack	$208,000
Loading of canister-based storage	$312,000
Total	$1,300,000

D.3.2.3.6 Incremental Annual ISFSI Operating Costs

Annual operating costs for an ISFSI during reactor operation include the costs associated with NRC inspections; security; radiation monitoring; ISFSI operational monitoring; technical specification and regulatory compliance, including implementation of new certificate of compliance (CoC) amendments; personnel cost and code maintenance associated with fuel selection for dry storage; personnel costs for spent fuel management and fabrication surveillance activities; electric power usage for lighting and security systems; road maintenance to the ISFSI site; and miscellaneous expenses associated with ISFSI maintenance. NRC license fees for dry storage are included as part of the 10 CFR 50 operating license fees.

Because the reference plant has already implemented dry storage, there are no incremental annual ISFSI operating costs expected to implement dry storage at an earlier date if a policy decision is made to accelerate the transfer of spent fuel stored in spent fuel pools to dry storage. Annual operating costs are a function of when a company begins dry storage.

Therefore, incremental costs associated with annual ISFSI operating costs are insignificant for this analysis.

D.3.2.3.7 Dry Storage Occupational Exposure (Routine)

Routine occupational exposure associated with dry storage of spent fuel includes worker dose associated with additional DSC loading, unloading and handling activities; additional ISFSI operations, maintenance, and surveillance activities; additional DSC storage at an ISFSI; and additional transportation cask loading, unloading, and handling activities.

Worker dose associated with DSC loading operations vary depending upon the cask technology being loaded, the characteristics of the fuel being loaded (e.g., fuel age and burnup), and fuel loading patterns in the DSC (e.g., the location of short-cooled, high burnup spent fuel or colder spent fuel within DSC baskets using regional loading). For the regulatory baseline, a worker dose of 400 person-mrem per DSC loaded was assumed. This radiation dose is consistent with that used in EPRI TR-1021049 and in EPRI TR-1018058, which analyzed worker impacts associated with loading spent fuel for transport to the proposed Yucca Mountain repository. Some sites achieve per package dose ranges in the range of 200 to 300 person-mrem per package loaded, while other sites experience higher per package dose rates. For the low-density storage case, each cask loaded in addition to the number required by the regulatory baseline is estimated to result in an incremental 400 person-mrem dose.

There is routine occupational dose associated with ISFSI annual operation and maintenance activities (i.e., inspection, surveillance, and security operations). The regulatory baseline assumes an annual dose of 120 person-mrem per site per year for inspection, surveillance, and security activities and 1,500 person-mrem per site per year for ISFSI operations and maintenance. These estimated radiation doses are consistent with assumptions used by EPRI in EPRI TR-1021049 and TR-1018058. Because additional shielding is assumed to be provided by concrete overpacks, the worker dose associated with ISFSI operations and maintenance is not expected to increase. Therefore, there is no incremental occupational dose predicted for performing annual ISFSI operation and maintenance.

There is routine occupational dose associated with the storage of each DSC at an operational ISFSI. The regulatory baseline assumes a worker dose of 170 person-mrem for each additional DSC loaded at an ISFSI site. This estimated radiation dose is consistent with assumptions used by EPRI in EPRI TR-1021049 and TR-1018058. Because additional shielding is assumed to be provided by concrete overpacks, the worker dose associated with each DSC stored at an operational ISFSI is not expected to increase. For the low-density spent fuel pool storage case, each cask stored in addition to the number required by the regulatory baseline is estimated to result in an incremental 170 person-mrem dose.

Table 91 summarizes the occupational dose estimates for each activity.

Table 91 Incremental Occupational Dose (Routine) Estimates

Activity	Incremental Occupational Dose (Routine) (person-mrem per activity)
Load a DSC	400
ISFSI Operation and maintenance	0
Loading a DSC at an ISFSI	170
Total	570

D.3.2.3.8 Number of Dry Storage Casks

In 2012, the reference plant has 3,819 fuel assemblies stored in the spent fuel pool in a high-density 1x4 loading configuration. During each refueling outage, 284 assemblies are offloaded from the reactor vessel to the spent fuel pool. For the regulatory baseline, the plant is expected to load the required number of Transnuclear TN-68 DSCs with a 68-assembly capacity each refueling outage to retain sufficient space in the spent fuel pool to discharge one full core of fuel (full core reserve). The estimated inventory for use by this regulatory analysis is shown in Table 92.

Table 92 Regulatory Baseline Loading of Dry Storage Casks

Year	Initial SFP inventory	Refueling	Placed into dry storage	Final SFP Inventory	No. of casks loaded	Cask Capacity
2012	3055	284	-340	2999	5	68
2014	2999	284	-272	3011	4	68
2016	3011	284	-272	3023	4	68
2018	3023	284	-272	3035	4	68
2020	3035	284	-272	3047	4	68
2022	3047	284	-340	2991	5	68
2024	2991	284	-272	3003	4	68
2026	3003	284	-272	3015	4	68
2028	3015	284	-272	3027	4	68
2030	3027	284	-272	3039	4	68
2032	3039	284	-272	3051	4	68
2034	3051	764	0	3815	0	68
2040	3815	0	-816	2999	12	68
2041	2999	0	-816	2183	12	68
2042	2183	0	-816	1367	12	68
2043	1367	0	-680	687	10	68
2044	687	0	-687	0	11	68
Total number of casks					103	

At the expiration of the operating license in 2034, the full core is offloaded into the spent fuel pool. The analysis further assumes that the entire spent fuel pool inventory will gradually be placed into dry storage beginning in 2040 and completed by 2044, 10 years after termination of unit commercial operation.

For the low-density spent fuel pool storage case, it is assumed that there is an NRC policy decision that requires licensees to offload the spent fuel inventory to dry storage to obtain a low-density 1x4 with empties configuration within five years (e.g., by end of 2019). In this configuration, the reference plant spent fuel pool stores 852 assemblies (Spent Fuel Pool Study, Table 15). Using the same initial conditions as above, and using the DSC with a 57-assembly derated capacity beginning in year 2019, the inventory model is provided in Table 93.

Table 93 Low-density Spent Fuel Pool Case Loading of Dry Storage Casks

Year	Initial SFP inventory	Refueling	Placed into dry storage	Final SFP Inventory	No. of casks loaded	Cask Capacity
2012	3055	284	-340	2999	5	68
2013	2999	0	0	2999		
2014	2999	284	-544	2739	8	68
2015	2739	0	-544	2195	8	68
2016	2195	284	-544	1935	8	68
2017	1935	0	-544	1391	8	68
2018	1391	284	-544	1131	8	68
2019	1131	0	-285	846	5	57
2020	846	284	-285	845	5	57
2022	845	284	-285	844	5	57
2024	844	284	-285	843	5	57
2026	843	284	-285	842	5	57
2028	842	284	-285	841	5	57
2030	841	285	-285	841	5	57
2032	841	286	-285	842	5	57
2034	842	764	-798	808	14	57
2043	808	0	-408	400	6	68
2044	400	0	-400	0	6	68
Total number of casks					111	

At the expiration of the operating license in 2034, the full core is offloaded into the spent fuel pool. The analysis further assumes that the entire spent fuel pool inventory will gradually be placed into dry storage beginning in 2043 and completed by 2044, taking only two years because of the smaller remaining inventory. Additionally, in years 2038 and 2039, the spent fuel has cooled for a sufficient length of time that the DSC is no longer derated.

D.3.3 Sensitivity Analysis

D.3.3.1 Present Value Calculations

Current trends in the marketplace have provided returns on investments well below the 3 percent and 7 percent discount rates, which OMB Circular No. A-4 is based. The NRC is providing a zero discount rate (e.g., undiscounted values) as a sensitivity analyses. Historically, regulatory analyses have provided the undiscounted values for the costs and benefits for information purposes, but have not provided them as a sensitivity analysis. However, the NRC is reporting the undiscounted costs and benefits as part of the sensitivity analysis based on current market trends and future predictions.

D.3.3.2 Dollar per Person-Rem Conversion Factor

The NRC is currently revising the dollar per person-rem averted conversion factor based on recent information regarding the value of a statistical life (VSL). However, until the NRC completes the update and publishes the appropriate guidance documents, the NRC will perform sensitivity analysis to estimate the impact on the calculated results when more current VSL and cancer risk factor are used. The NRC used the U.S. Environmental Protection Agency's (EPA) VSL as an interim value in the sensitivity analysis. The EPA's VSL was developed through a rigorous process, reviewing many published academic papers, and includes review from the Scientific Advisory Board, an independent review board.

The EPA's VSL in 2009 dollars is approximately $7.2 million.[59] The VSL is derived from "using a mixed effects model (random intercept with fixed effects for study characteristics), the authors regressed the VSL estimates on average income, probability of death, and several study design variables" (EPA, page 41). Therefore, using the CPI-U based inflator to adjust from 2009 dollars to 2012 dollars yields a VSL of approximately $7.7 million. The International Commission on Radiation Protection (ICRP) updated the mortality risk factor in ICRP Publication No. 103, the updated risk coefficient is 5×10^{-4}. Using the updated ICRP risk coefficient and escalated EPA-based VSL, the dollar per person-rem conversion, rounded to the nearest thousand, is $4,000 per person-rem.

Therefore, the NRC will provide the $2,000 per person-rem conversion value for the recommendation and the $4,000 per person-rem conversion value as a sensitivity analysis for this regulatory analysis.

D.3.3.3 Replacement Energy Costs

The NRC is currently updating its estimates for replacement energy costs based on a U.S. competitive electricity market area model. The updated model provides the replacement energy costs by day, week, and year, based on market area, in 2010 dollars. For each U.S. power market area, a lowest cost and highest cost replacement energy cost estimate was calculated, normalizing for reactor megawatt rating differences. The estimated replacement energy cost per reactor per year ranges from a high estimate of $54.4 million to a low estimate of $692,000 across all U.S. power markets. The average estimated cost per reactor per year across all U.S. power markets is $9.6 million and the median estimated cost is $6.4 million in 2010 dollars. Using the CPI-U inflator formula and the 2010 CPI-U inflator value from Table 77, the estimated replacement energy costs range from $57.3 million to $729,000 in 2012 dollars. The average estimated cost per reactor per year across all US power markets is $10.1 million and the median estimated cost is $6.7 million in 2012 dollars.

D.3.3.4 Consequences Extending Beyond 50 Miles

NUREG/BR-0184 states that in the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile distance from the plant site. However, in this circumstance it is beneficial for the analysis to include supplemental information (e.g., analyses and results) that go beyond the guidance provided in this document. The Spent Fuel Pool Study uses a plume release model that predicts slow deposition of aerosols. This results in public health consequences that extend beyond 50 miles from the postulated accident site. While the accuracy of the model decreases with distance, the amount of public exposure beyond 50 miles in the event of a release is expected to be significant. To capture effects beyond 50 miles, this regulatory analysis

[59] Environmental Protection Agency, National Center for Environmental Economics, "Valuing Mortality Risk Reductions for Environmental Policy: A White Paper", dated December 2010.

evaluates the public health and safety and economic consequences estimated by the plume model beyond the 50-mile distance from the plant site as a sensitivity analysis.

D.3.4 Alternative – Low-Density Spent Fuel Pool Storage

D.3.4.1 Public Health (Accident)

This attribute measures expected changes in radiation exposure to the public due to change in accident frequencies or accident consequences associated with the proposed action. The expected changes in radiation exposure are predicted over a 50-mile radius from the plant site. The calculated radiation dose to the public is primarily from reoccupation of the land and other activities following the spent fuel pool accident. In addition, the calculated radiation dose to the public includes the occupational dose to workers for cleanup and decontamination of contaminated land not onsite. The incremental radiation doses are calculated by subtracting the values for the alternative from those of the regulatory baseline. The difference (delta) is the averted dose benefit of this alternative in units of person-rem. The quantitative results for public health (accident) considering the contribution of all initiators that could affect spent fuel pool risk is provided in Table 94. These values are based on the MACCS2 analyses and probabilistic considerations described in further detail in the Spent Fuel Pool Study and other referenced documents. The assumptions with regard to the release frequencies are discussed in section D.3.2.2.1 and with regard to the habitability criteria are found in section D.3.2.2.8 of this regulatory analysis.

Table 94 Summary of Public Health (Accident) for Low-density Spent Fuel Pool Storage [All Initiators]

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
	Low Est.	Best Est.	High Est.	Undiscounted	3% Net Present Value			7% Net Present Value		
				Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Low-density storage	60	124	1,260	$247,700	$86,700	$179,500	$1,825,500	$60,200	$124,600	$1,267,000

As Table 94 shows, the best estimate of the delta benefit for averted public health (accident) radiation exposure from a spent fuel pool accident, which results in spent fuel damage, is 124 person-rem. This dose represents the reduction of public health risk that results from a policy decision to transfer spent fuel from the spent fuel pool to dry storage in order to achieve low-density spent fuel loading in the pool at the reference plant. This value is based on a spent fuel pool accident that results in an averted delta dose exposure of approximately 5.6 person-rem per reactor-year over a remaining licensed lifetime of 22 years (until year 2034). The best estimate values are based on the reference site's population density of 722 people per square mile within a 50-mile radius from the site and result from the uncontrolled release of radionuclides from a full spent fuel pool. The low estimate case reflects the health benefit of a spent fuel pool with low-density storage compared to a pool with high-density storage if the more stringent Pennsylvania protective action guides are used following an event challenging spent fuel pool cooling. The high estimate case reflects the calculated health benefits that result if a less stringent 2 rem annual dose protective action guide is used.

A case to evaluate the sensitivity of the results to a change in the dollar per person-rem conversion value from $2,000 to $4,000 per person-rem averted was performed. The results of this case are provided in Table 95.

Table 95 Sensitivity Analyses of Public Health (Accident) Benefits for Low-density Spent Fuel Pool Storage for All Initiating Events (within 50 miles)

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
				Undiscounted	3% Net Present Value			7% Net Present Value		
	Low Est.	Best Est.	High Est.	Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Dollar per person-rem value	60	124	1,260	$495,500	$173,400	$358,900	$3,650,900	$120,400	$249,100	$2,534,000

Because a spent fuel pool fire under certain scenarios and environmental conditions could result in impacts to public health that extend beyond 50 miles, the next two cases evaluate the sensitivity of averted public health exposures extending beyond 50 miles from the site. The first sensitivity case extends the analysis beyond 50 miles from the plant site and uses the same low, best, and high estimate case assumptions for habitability described above and uses the standard $2000 per person-rem conversion factor. The second sensitivity case evaluates the sensitivity of extending the analysis beyond 50 miles and uses a $4,000 per person-rem conversion factor. Table 96 shows the sensitivity on public health (accident) benefits for these two cases.

Table 96 Sensitivity Analyses of Public Health (Accident) Benefits for Low-density Spent Fuel Pool Storage for All Initiating Events (extending beyond 50 miles)

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
	Low Est.	Best Est.	High Est.	Undiscounted	3% Net Present Value			7% Net Present Value		
					Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Base case extended beyond 50 miles	541	892	7,868	$1,783,450	$783,250	$1,291,900	$11,399,100	$543,650	$896,700	$7,911,700
Dollar per person-rem value	541	892	7,868	$3,566,900	$1,566,500	$2,583,800	$22,798,200	$1,087,300	$1,793,400	$15,823,400

D.3.4.2 Occupational Health (Accident)

Occupational health measures both short-term and long-term health effects associated with site workers as a result of changes in accident frequency or accident mitigation. Within the regulatory baseline, the short-term occupational exposure related to the accident occurs at the time of the accident and during the immediate management of the emergency and during decontamination and decommissioning of the onsite property. The radiological occupational exposure resulting from cleanup and refurbishment or decommissioning activities of the damaged facility to occupational workers are estimated within the long-term occupational exposure. The quantitative results for occupational health (accident) considering the contribution of all initiators that could affect spent fuel pool risk is provided in Table 97 and is based on the release frequencies discussed in section D.3.2.2.1 and the occupational health (accident) assumptions found in section D.3.2.2.7.

Table 97 Occupational Health (Accident) Benefits for Low-density Spent Fuel Pool Storage Considering All Initiating Events

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
	Low Est.	Best Est.	High Est.	Undiscounted	3% Net Present Value			7% Net Present Value		
				Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
short-term	0.3	0.4	5.5	$841	$388	$628	$7,959	$269	$453	$5,524
long-term	0.1	0.2	5.2	$415	$98	$310	$7,490	$68	$223	$5,198
Total	0.3	0.6	10.7	1,260	490	940	15,450	340	680	10,720

As Table 97 shows, the total delta benefit for short- and long-term occupational health (accident) is 0.6 person-rem averted per reactor. The estimated total benefit of the occupational health (accident) attribute for low-density spent fuel pool storage relative to the regulatory baseline, using the $2,000 per person-rem averted conversion factor, net present value ranges are insignificant for this analysis and do not warrant further sensitivity analysis.

D.3.4.3 Occupational Health (Routine)

Occupational health (routine) accounts for radiological exposures to workers during normal facility operations (i.e., non-accident situations). These occupational exposures occur during DSC loading and handling activities, ISFSI operations, and maintenance and surveillance activities. The assumptions in relation to the exposures for occupational health (routine) are found in section D.3.2.3.7 of this regulatory analysis.

Table 98 Occupational Health (Routine) Costs for Low-density Spent Fuel Pool Storage

	No. of DSCs			Dose (person-rem)		Costs (2012 dollars)		
Year	Low-Density SFP Loading	Regulatory Baseline	Difference	Exposure per DSC	Additional Dose	No Discount	3% NPV	7% NPV
2012	5	5	0	0.57	0	$0	$0	$0
2013	0	0	0	0.57	0	$0	$0	$0
2014	8	4	-4	0.57	-2.28	-$4,560	-$4,298	-$3,983
2015	8	0	-8	0.57	-4.56	-$9,120	-$8,346	-$7,445
2016	8	4	-4	0.57	-2.28	-$4,560	-$4,052	-$3,479
2017	8	0	-8	0.57	-4.56	-$9,120	-$7,867	-$6,502
2018	8	4	-4	0.57	-2.28	-$4,560	-$3,819	-$3,039
2019	5	0	-5	0.57	-2.85	-$5,700	-$4,635	-$3,550
2020	5	4	-1	0.57	-0.57	-$1,140	-$900	-$663
2022	5	5	0	0.57	0	$0	$0	$0
2024	5	4	-1	0.57	-0.57	-$1,140	-$800	-$506
2026	5	4	-1	0.57	-0.57	-$1,140	-$754	-$442
2028	5	4	-1	0.57	-0.57	-$1,140	-$710	-$386
2030	5	4	-1	0.57	-0.57	-$1,140	-$670	-$337
2032	5	4	-1	0.57	-0.57	-$1,140	-$631	-$295
2034	14	0	-14	0.57	-7.98	-$15,960	-$8,329	-$3,602
2040		12	12	0.57	6.84	$13,680	$5,979	$2,058
2041		12	12	0.57	6.84	$13,680	$5,805	$1,923
2042		12	12	0.57	6.84	$13,680	$5,636	$1,797
2043	6	10	4	0.57	2.28	$4,560	$1,824	$560
2044	6	11	5	0.57	2.85	$5,700	$2,214	$654
				Total:	**-4.56**	**-$9,000**	**-$24,000**	**-$27,000**

As Table 98 shows, the delta benefit for occupational health (routine) is an increase of 4.56 person-rem in worker exposure resulting from DSC loading and handling activities; ISFSI operations; and maintenance and surveillance activities. The estimated cost to the occupational health (routine) for low-density spent fuel storage relative to the regulatory baseline and calculated in accordance with the current regulatory framework, ranges from $24,000 (3 percent net present value) to $27,000 (7 percent net present value) using the $2,000 per person-rem averted conversion factor. These ranges are insignificant for this analysis and do not warrant further sensitivity analysis.

D.3.4.4 Offsite Property

The offsite property attribute measures the expected total monetary effects on offsite property resulting from the proposed action. Changes to offsite property can take various forms, both direct, (e.g. land, food, and water) and indirect (e.g. tourism). This attribute is the product of the change in accident frequency and the property consequences from the occurrence of a spent fuel pool accident at the reference plant.

For the regulatory baseline, the offsite property costs are any property consequences resulting from any radiological release from the occurrence of an accident. Normal operational releases and any plant releases not related to the severe accident analyzed are outside the scope of this regulatory analysis.

The cost offsets for the analyzed spent fuel pool accident are quantified relative to the regulatory baseline based on the MACCS2 calculation results and probabilistic considerations provided in the main document. The results for the consequences from a low-density spent pool accident are compared to those from the regulatory baseline spent fuel pool accident. The calculation is the difference between the calculated consequences resulting from a low-density and a high-density spent fuel pool accident and are provided in Table 99.

Table 99 Offsite Property Cost Offsets for Low-density Spent Fuel Pool Storage

Case	Offsite Property Cost Offsets (2012 dollars)						
	Undiscounted	3% Net Present Value			7% Net Present Value		
	Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Base case, consequences within 50 miles	$723,300	$777,500	$524,000	$3,323,400	$539,700	$363,700	$2,306,700
Sensitivity study, consequences extend beyond 50 miles	$2,139,300	$3,599,100	$1,549,700	$8,393,400	$2,498,000	$1,075,600	$5,825,500

As Table 99 shows the estimate of offsite property damage can vary significantly with the criterion used to measure or estimate the level of contamination. This regulatory analysis uses three protective action levels – the Pennsylvania PAG of 500 mrem annually for the low estimate, the EPA intermediate phase PAG level of 2 rem in the first year, and 500 mrem annually thereafter for the best estimate, and 2 rem annually for the high estimate – to evaluate post-accident collective dose and offsite property costs. As discussed in section D.3.2.2.8, offsite property costs are inversely proportional to changes in collective dose resulting from changes in habitability criteria (i.e., lower PAG guidelines result in lower collective dose value and higher offsite property costs). Furthermore, the high estimate is also affected by the bounding assumption used in establishing the high estimate spent fuel pool release frequency shown in Table 75. As shown in Table 99 the estimated total cost offsets for the low-density storage option relative to the regulatory baseline range from $0.5 to $3.3 million (3 percent net present value) and from $0.4 to $2.3 million (7 percent net present value) considering consequences within 50 miles from the site. As a sensitivity study, the analysis of potential consequences was extended beyond 50 miles from the site and were quantified based on the MACCS2 model. These estimate results are also shown in Table 99 and result in cost offsets approximately 2.5 to 4.6 times greater than those in the base case result.

This analysis does not address potential changes to current methodologies and tools to regulatory analysis guidance that may result from applying SOARCA insights and improving guidance and analysis tools (such as the MACCS2 computer code) based on up-to-date data in addition to advancements in accident consequence assessment knowledge as it relates to this attribute.

D.3.4.5 Onsite Property

This attribute measures the expected monetary effects on onsite property, including replacement power costs, decontamination, and refurbishment costs, from the proposed action. There are two forms of onsite property costs that each alternative must disposition. The first type of onsite property costs are the cleanup and decontamination costs for the unit. The second type of onsite property costs is the cost to replace the energy from the damaged or shutdown unit(s). The cost offsets for low-density spent fuel pool storage are quantified relative to the regulatory baseline based on the probabilistic considerations provided in the main document and the onsite property estimates described in section D.3.2.2.5.

As stated in section D.3.2.2.6, another unit is co-located on the reference plant's site. Therefore, both units may not operate (e.g., due to significant site damage or contamination resulting in high occupational exposure to the undamaged unit) due to the spent fuel pool accident. In modeling the replacement energy costs based on this scenario, it is assumed for the high estimate that replacement energy would be purchased for both units.

Based on these modeling assumptions, the onsite property results are provided in Table 100.

Table 100 Summary of Onsite Property Cost Offsets for Low-density Spent Fuel Pool Storage

Case	Onsite Property Cost Offsets (2012 dollars)						
	Undiscounted	3% Net Present Value			7% Net Present Value		
	Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Onsite Property - Replacement Energy	$1,639	$50	$1,091	$117,100	$30	$682	$73,200
Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$8,800	$2,900	$5,800	$132,500	$1,800	$3,600	$82,600
Total	$10,440	$2,950	$6,890	$249,600	$1,830	$4,280	$155,800

As Table 100 shows, based on these calculations, the delta cost offset for probability weighted onsite property best estimate ranges from $6,890 (3 percent net present value) to $4,280 (7 percent net present value). Low and high estimates are also provided in Table 100.

D.3.4.6 Industry Implementation

Industry implementation accounts for the projected net economic effect on the affected licensees to implement the mandated changes. Costs evaluated for dry storage include upfront and incremental DSC capital and loading costs. Additional costs above the regulatory baseline are considered negative and cost savings are considered positive. The quantitative results for industry implementation are given in terms of expected costs if a policy decision is made to accelerate the transfer of spent fuel stored in spent fuel pools to dry storage. These expected costs are not frequency weighted. Assumptions used for developing the industry implementation cost model are discussed in sections D.3.2.3.2, D.3.2.3.5, and D.3.2.3.6, with the results provided in Table 101.

Table 101 Industry Implementation Cost Model for Low-density Spent Fuel Pool Storage

Year	No. of DSCs			Unit Costs			Costs (2012 dollars)		
	Low-Density	Regulatory Baseline	Difference	One Time ISFSI Mod	Upfront costs per	DSC Purchase and Loading	No Discount	3% NPV	7% NPV
2012	5	5	0		$657,632	$1,300,000	$0	$0	$0
2013	0	0	0		$657,632	$1,300,000	$0	$0	$0
2014	8	4	-4		$657,632	$1,300,000	-$7,830,528	-$7,381,024	-$6,839,486
2015	8	0	-8		$657,632	$1,300,000	-$15,661,056	-$14,332,085	-$12,784,087
2016	8	4	-4		$657,632	$1,300,000	-$7,830,528	-$6,957,323	-$5,973,872
2017	8	0	-8		$657,632	$1,300,000	-$15,661,056	-$13,509,364	-$11,166,116
2018	8	4	-4		$657,632	$1,300,000	-$7,830,528	-$6,557,944	-$5,217,811
2019	5	0	-5		$657,632	$1,300,000	-$9,788,160	-$7,958,670	-$6,095,574
2020	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,545,373	-$1,139,360
2022	5	5	0		$657,632	$1,300,000	$0	$0	$0
2024	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,373,044	-$869,212
2026	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,294,225	-$759,203
2028	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,219,932	-$663,118
2030	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,149,902	-$579,193
2032	5	4	-1		$657,632	$1,300,000	-$1,957,632	-$1,083,893	-$505,889
2034	14	0	-14		$657,632	$1,300,000	-$27,406,848	-$14,303,428	-$6,186,086
2040	0	12	12		$657,632	$1,300,000	$23,491,584	$10,267,625	$3,533,186
2041	0	12	12		$657,632	$1,300,000	$23,491,584	$9,968,568	$3,302,043
2042	0	12	12		$657,632	$1,300,000	$23,491,584	$9,678,222	$3,086,022
2043	6	10	4		$657,632	$1,300,000	$7,830,528	$3,132,111	$961,377
2044	6	11	5		$657,632	$1,300,000	$9,788,160	$3,801,105	$1,123,105
		Total:	**-8**			**Total:**	**-$15,660,000**	**-$41,820,000**	**-$46,770,000**

For this analysis, the Transnuclear TN-68 dry casks are evaluated for the best estimate because the reference plant's ISFSI for dry cask storage utilizes the TN-68 cask design as discussed in Section 1.3 of the main report. The results provided in Table 102 show that eight additional DSCs are needed to store the hotter spent fuel.

Table 102 Industry Implementation Costs for Low-density Spent Fuel Pool Storage

Case	Costs (2012 dollars)		
	No Discount	3% Net Present Value	7% Net Present Value
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000

Table 102 shows, the incremental costs associated with DSC upfront costs and the earlier purchasing and loading of DSCs on a periodic basis. The estimated industry implementation costs for low-density spent fuel storage relative to the regulatory baseline and calculated in accordance with the current regulatory framework, ranges from $41.8 million (3 percent net present value) to $46.8 million (7 percent net present value).

D.3.4.7 Industry Operation

Industry operation accounts for the projected net economic effect due to routine and recurring activities required by the proposed alternative. Annual operating costs for an ISFSI during reactor operation include the costs associated with NRC inspections; security; radiation monitoring; ISFSI operational monitoring; technical specification and regulatory compliance, including implementation of new certificate of compliance (CoC) amendments; personnel cost and code maintenance associated with fuel selection for dry storage; personnel costs for spent fuel management and fabrication surveillance activities; electric power usage for lighting and security systems; road maintenance to the ISFSI site; and miscellaneous expenses associated with ISFSI maintenance. NRC license fees for dry storage are included as part of the 10 CFR 50 operating license fees. As discussed in section D.3.2.3.6, incremental costs associated with annual ISFSI operating costs are insignificant for this analysis.

Industry operation also includes annual operating costs following reactor shutdown for decommissioning, which includes the costs associated with transporting spent fuel offsite. These costs were beyond the scope of the evaluation of expedited transfer of spent fuel to dry cask storage and are not included in this analysis.

D.3.4.8 NRC Implementation

These costs, if calculated, would further reduce the calculated net benefit for this reference plant regulatory and backfit analysis.

D.3.4.9 NRC Operation

These costs, if calculated, would further reduce the calculated net benefit for this reference plant regulatory and backfit analysis.

D.3.4.10 Other Considerations

D.3.4.10.1 Modeling Uncertainties

There remain significant uncertainties in estimating the frequency of events for natural phenomena, which are postulated to challenge spent fuel pool cooling or integrity. There are also significant uncertainties in the calculation of event consequences in terms of the dispersion and disposition of radioactive material into the site environs. This is due in part to significant uncertainties regarding the degree to which topographical features and other phenomena are modeled at distances away from the reference plant. Estimating economic consequences also includes large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies and the loss of infrastructure on the general U.S. economy. An example of this is the supply chain disruptions that followed the 2011 Tohoku earthquake and subsequent tsunami on Japan or the 2004 Indian Ocean earthquake and tsunami on Thailand.

D.3.4.10.2 Cask Handling Risk

The NRC recognizes that there are costs and risks associated with the handling and movement of spent fuel casks in the reactor building. These cost and risk impacts, if included in this analysis, would further reduce the overall net benefit in relation to the regulatory baseline. These effects (e.g., the added risks of handling and moving casks) were conservatively ignored in order to calculate the maximum potential benefit by only comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and its implementation costs without consideration of cask movement risk.

D.3.4.10.3 Mitigating Strategies

The release of fission products to the environment from events that may cause the loss of spent fuel pool cooling or integrity, such as seismic events, missiles, heavy load drops, loss of cooling or make-up, inadvertent drainage or siphoning and pneumatic seal failures, are estimated to be approximately 5.5×10^{-7} per reactor-year without successful mitigation. Operator diagnosis and recovery are important factors considered in the development of the event frequencies for these events and portions of this evaluation are premised on licensees having taken appropriate actions to understand the potential consequences of spent fuel pool accident events and develop appropriate procedures and mitigating strategies to respond and mitigate the consequences.

The main report evaluated the potential benefits of mitigation measures required under Title 10, Code of Federal Regulations (10 CFR), Part 50.54 (hh)(2), which were implemented following the September 11, 2001 attacks. These mitigation measures are intended to maintain spent fuel pool cooling in the event of a loss of large areas of the plant due to explosions or fire. The main report does not consider the post-Fukushima improvements required by NRC and being implemented by the plants. These improvements are intended to increase the likelihood of restoring or maintaining power and mitigation capability during severe accidents.

The new spent fuel pool level instrumentation required under Order EA-12-051 and the mitigation strategies now required under Order EA-12-049, significantly enhance the likelihood of successful mitigation because of the following features:

- Portable equipment with redundant sets (e.g., N+1) that is sufficient to supply all functions, simultaneously for the entire site, including equipment for the spent fuel pool. This portable equipment provides additional protection from seismic events, which are a dominant contributor to spent fuel pool risk.

- The mission time for this equipment is indefinite, versus the 12-hour mission time for the 50.54(hh)(2) equipment.[60]

- The new EA-12-049 mitigating strategies are capable of being deployed in all modes, which means that the new strategies can address spent fuel pool cooling issues that could occur in any operating cycle phase.

- The new spent fuel pool level instrumentation required under Order EA-12-051, ensures a reliable indication of the water level in the spent fuel pool for identification of the following pool water level conditions:
    - A level that is adequate to support operation of the normal fuel pool cooling system
    - A level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and
    - A level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

[60] This section of the regulations deals with the development and implementation of guidance and strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with loss of large areas of the plant resulting from explosions or fire.

- The minimum spent fuel pool makeup flow rate under Order EA-12-049 is set to match the design basis heat load for the spent fuel pool, which is typically a full core offload in addition to the recently removed fuel from the last refueling outage. This results in a lower flow rate than that in NEI guidance for Part 50.54 (hh)(2) equipment and an earlier transition to spray, if necessary, due to leaks.

- The method of filling the spent fuel pool is via a connection to the normal spent fuel pool makeup system located away from the spent fuel pool floor, reducing the impacts on human performance due to potentially adverse environmental conditions (e.g., high temperature, humidity, and radiation) following an event.

This additional equipment, strategies, and features provided by Orders EA-12-049 and EA-12-051, provide additional accident mitigation capability and would further enhance the likelihood of successful mitigation, thereby further reducing the value for the conditional probability of release.

D.3.4.10.4 Other Favorable Spent Fuel Loading Configurations

In section 9.2 of the Spent Fuel Pool Study, a sensitivity analysis is provided in which a more favorable fuel pattern is considered. In this more favorable pattern, eight cold assemblies surround each hot assembly (i.e., 1x8 fuel assembly pattern). Although only a few sensitivity analysis were performed using this configuration, the results are promising. The sensitivity calculations for the high-density 1x8 fuel pattern showed a shorter time to air coolability (i.e. no releases in OCP3). Even for the cases that led to the release of radioactive materials in OCP2, the release magnitude was much smaller than for the 1x4 fuel pattern, and comparable to the low-density cases. Furthermore, the high-density loading configuration, which allows for 764 empty cells for a full core offload may result in similar reductions in risk to the low-density storage option evaluated without the significant capital costs for implementation. Further evaluation of this alternative and possibly other loading configurations for all operating cycle phases is recommended.

D.4 <u>PRESENTATION OF RESULTS</u>

This section presents the analytical results, including discussion of supplemental considerations, uncertainties in estimates, and results of sensitivity analyses on the overall benefits. The results are presented in two different ways, in order to address the differing decision criteria between regulatory analyses and backfit analyses (10 CFR 50.109).

D.4.1 Regulatory Analysis

D.4.1.1 Summary Table

Table 103 summarizes the quantified net benefits used to perform a safety goal screening.

Table 103 Summary of Net Benefits for Low-density Spent Fuel Pool Storage Considering All Initiator Events (within 50 miles)

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$247,700	$179,500	$124,600	$119,700	$86,700	$60,200	$2,520,000	$1,825,500	$1,267,000
Occupational Health (Accident)	$1,300	$900	$700	$700	$500	$300	$21,300	$15,400	$10,700
Offsite Property	$723,300	$524,000	$363,700	$1,073,300	$777,500	$539,700	$4,587,800	$3,323,400	$2,306,700
Onsite Property	$10,400	$6,900	$4,300	$4,480	$2,950	$1,830	$378,600	$249,600	$155,800
Total Benefits	***$982,700***	***$711,300***	***$493,300***	***$1,198,200***	***$867,700***	***$602,000***	***$7,507,700***	***$5,413,900***	***$3,740,200***
Occupational Health (Routine)	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***
Net Benefit	**-$15,416,000**	**-$41,385,000**	**-$46,368,000**	**-$15,200,800**	**-$41,228,300**	**-$46,259,000**	**-$8,891,300**	**-$36,682,100**	**-$43,120,800**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

As shown in Table 103, the calculated net benefits for requiring low-density spent fuel pool storage at the reference plant does not achieve a positive net benefit using the current regulatory framework. This means that the calculated licensee costs to implement a low-density spent fuel pool storage alternative at the referenced site outweighs the estimated benefits.

Furthermore, for the seismic event analyzed for the Spent Fuel Pool Study, no offsite early fatalities are calculated to occur. This result is expected for two main reasons:

1. In comparison to reactors, spent fuel pools have a larger proportion of longer-lived radionuclides, which are less likely to cause the significant doses required for acute health effects.

2. Despite the large releases for certain predicted spent fuel pool accident progressions, the release from the most recently discharged fuel (which contains the shorter-lived radionuclides) is predicted to be insufficiently fast and insufficiently large to reach the acute thresholds associated with offsite early fatalities. When doses do exceed minimum levels for early fatalities, emergency response, as treated in the main report, effectively prevents any early fatality risk, at least in part because the modeled accident progression results in releases that are long compared with the time needed for relocation.

In addition, the predicted long-term exposure of the population, which could result in latent cancer fatality risk, is also low for the following reasons:

1. The individual latent individual latent cancer fatality risk within 0-10 miles for the studied scenarios is predicted to be on the order of 10^{-10} to 10^{-11} per year, based on the linear no threshold (LNT) dose response model.

2. The risk within 10 miles of the analyzed accident is dominated by low dose received at a low dose rate. According to alternate dose response models, excluding the uncertain effects of low radiation dose could reduce the quantified individual latent cancer fatality risk within 10 miles to be approximately 10^{-14} per year, a reduction of approximately 3,000 times.

3. Average individual latent cancer fatality risk is low and decreases slowly as a function of distance from the plant. Additionally, the predicted individual risks latent cancer fatalities are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough to be considered habitable. Therefore, the use of alternate dose response models would significantly reduce the quantified latent cancer fatalities by at least an order of magnitude.

D.4.1.2 Implementation and Operation Costs

Table 104 Summary of Total Implementation and Operation Costs for Low-density Spent Fuel Pool Storage for All Initiator Events

Attribute	Costs (2012 dollars in millions)	
	3% Net Present Value	7% Net Present Value
Occupational Health (Routine)	$0.024	$0.027
Industry Implementation	$41.800	$46.770
Industry Operation	$0.252	$0.064
NRC Implementation	nc	nc
NRC Operation	nc	nc
Total	$42.096	$46.861

As shown in Table 104, the total estimated costs for the referenced plant unit to achieve and maintain a low-density spent fuel pool loading range from $42 million (3 percent net present value) to $47 million (7 percent net present value). These costs are dominated by the capital costs for the DSCs and the loading costs for the storage systems to achieve low-density storage in the spent fuel pool than that required for the regulatory baseline.

D.4.1.3 Total Benefits and Cost Offsets

Table 105 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage for All Initiator Events

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	Undiscounted	3% Net Present Value	7% Net Present Value
Public Health (Accident)	$0.12 to $2.52	$0.09 to $1.83	$0.06 to $1.27
Occupational Health (Accident)	$0.001 to $0.021	$0.0005 to $0.015	$0.0003 to $0.011
Offsite Property	$0.72 to $4.59	$0.52 to $3.32	$0.36 to $2.31
Onsite Property	$0.004 to $0.38	$0.003 to $0.25	$0.002 to $0.16
Total	$0.85 to $7.51	$0.61 to $5.42	$0.42 to $3.75

The total benefits, which include the public health (accident) and occupational health (accident) is summed with the cost offsets, which include offsite property and onsite property relative to the regulatory baseline, are shown in the Table 105. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

D.4.1.4 Sensitivity Analysis

This section summarizes the results of the sensitivity analyses that were performed as an additional consideration in performing safety goal screening for requiring low-density spent fuel pool storage at the reference plant.

D.4.1.4.1 Dollar per Person-Rem Conversion Factor

The NRC is currently revising the dollar per person-rem averted conversion factor based on recent information regarding the value of a statistical life. However, until the NRC completes the update and publishes the appropriate guidance documents, the NRC performs sensitivity analysis to estimate the impact on the calculated results when more current VSL and cancer risk factor are used. The NRC used the U.S. Environmental Protection Agency's (EPA) VSL as an interim value in the sensitivity analysis as described in section D.3.3.2. The affect of this variable on the calculated results are provided in Table 106.

Table 106 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-density Spent Fuel Pool Storage Considering All Initiating Events (within 50 miles)

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$495,400	$359,000	$249,200	$239,400	$173,400	$120,400	$5,040,000	$3,651,000	$2,534,000
Occupational Health (Accident)	$2,600	$1,800	$1,400	$1,400	$1,000	$600	$42,600	$30,800	$21,400
Offsite Property	$723,300	$524,000	$363,700	$1,073,300	$777,500	$539,700	$4,587,800	$3,323,400	$2,306,700
Onsite Property	$10,400	$6,900	$4,300	$4,480	$2,950	$1,830	$378,600	$249,600	$155,800
Total Benefits	***$1,231,700***	***$891,700***	***$618,600***	***$1,318,600***	***$954,900***	***$662,500***	***$10,049,000***	***$7,254,800***	***$5,017,900***
Occupational Health (Routine)	-$18,000	-$48,000	-$54,000	-$18,000	-$48,000	-$54,000	-$18,000	-$48,000	-$54,000
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,408,000***	***-$42,120,000***	***-$46,888,000***	***-$16,408,000***	***-$42,120,000***	***-$46,888,000***	***-$16,408,000***	***-$42,120,000***	***-$46,888,000***
Net Benefit	**-$15,176,000**	**-$41,228,000**	**-$46,269,000**	**-$15,089,400**	**-$41,165,100**	**-$46,225,500**	**-$6,359,000**	**-$34,865,200**	**-$41,870,100**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

As shown in Table 106, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530.

D.4.1.4.2 Consequences Extending Beyond 50 Miles

The RA Handbook states that in the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile distance from the plant site, although alternative distances from the plant may be used for sensitivity analyses. For this regulatory analysis, supplemental information (e.g., analyses and results) based on MACCS2 calculated results, which extends the analysis beyond 50 miles from the postulated accident site is provided in Table 107.

Table 107 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-density Spent Fuel Pool Storage Considering All Initiating Events

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$1,783,400	$1,291,900	$896,700	$1,081,200	$783,300	$543,600	$15,735,800	$11,399,100	$7,911,700
Occupational Health (Accident)	$1,300	$900	$700	$700	$500	$300	$21,300	$15,400	$10,700
Offsite Property	$2,139,300	$1,549,700	$1,075,600	$4,968,300	$3,599,100	$2,498,000	$11,586,600	$8,393,400	$5,825,500
Onsite Property	$10,400	$6,900	$4,300	$4,680	$3,150	$2,030	$378,600	$249,600	$155,800
Total Benefits	***$3,934,400***	***$2,849,400***	***$1,977,300***	***$6,054,900***	***$4,386,100***	***$3,043,900***	***$27,722,300***	***$20,057,500***	***$13,903,700***
Occupational Health (Routine)	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***	***-$16,399,000***	***-$42,096,000***	***-$46,861,000***
Net Benefit	**-$12,465,000**	**-$39,247,000**	**-$44,884,000**	**-$10,344,100**	**-$37,709,900**	**-$43,817,100**	**$11,323,300**	**-$22,038,500**	**-$32,957,300**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

As shown in Table 107, calculated net benefits for requiring low-density spent fuel pool storage at the reference plant do not achieve a positive net benefit for eight of the nine cases presented. One case, the undiscounted high estimate, shows a positive net benefit of about $11.3 million, which reflects the value and impacts at the time in which they are incurred with no present worth conversion. It is informative to compare this value to the other high estimate values of ($22.0 million) and ($33.0 million), which differ from this case by adjusting these future costs into year 2012 dollars using 3-percent and 7-percent discount rates as described in section D.3.2.1.2.

D.4.1.4.3 Combined Effect of Consequences Extending Beyond 50 Miles and Dollar per Person-Rem Conversion Factor

This sensitivity analysis considers all initiating events that can challenge the reference plant's spent fuel pool cooling or integrity while taking into account the combined effects of extending the analysis of consequences beyond 50 miles from the site and increasing the dollar per person-rem conversion value from $2,000 to $4,000 per person-rem averted. The combined effects of these two variables on the calculated net benefits are provided in Table 108.

Table 108 Combined Sensitivity Analysis that Analyzes Consequences Beyond 50 Miles using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-density Spent Fuel Pool Storage for All Initiator Events

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$3,566,900	$2,583,800	$1,793,400	$2,162,500	$1,566,500	$1,087,300	$31,471,600	$22,798,200	$15,823,400
Occupational Health (Accident)	$2,500	$1,900	$1,400	$1,300	$1,000	$700	$42,700	$30,900	$21,400
Offsite Property	$2,139,300	$1,549,700	$1,075,600	$4,968,300	$3,599,100	$2,498,000	$11,586,600	$8,393,400	$5,825,500
Onsite Property	$10,400	$6,900	$4,300	$4,680	$3,150	$2,030	$378,600	$249,600	$155,800
Total Benefits	***$5,719,100***	***$4,142,300***	***$2,874,700***	***$7,136,800***	***$5,169,800***	***$3,588,000***	***$43,479,500***	***$31,472,100***	***$21,826,100***
Occupational Health (Routine)	-$18,000	-$49,000	-$54,000	-$18,000	-$49,000	-$54,000	-$18,000	-$49,000	-$54,000
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,408,000***	***-$42,121,000***	***-$46,888,000***	***-$16,408,000***	***-$42,121,000***	***-$46,888,000***	***-$16,408,000***	***-$42,121,000***	***-$46,888,000***
Net Benefit	**-$10,689,000**	**-$37,979,000**	**-$44,013,000**	**-$9,271,200**	**-$36,951,200**	**-$43,300,000**	**$27,071,500**	**-$10,648,900**	**-$25,061,900**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

As shown in Table 108, calculated net benefits for requiring low-density spent fuel pool storage at the reference plant do not achieve a positive net benefit for eight of the nine cases presented. One case, the undiscounted high estimate, shows a positive net benefit of about $27.1 million, which reflects the value and impacts at the time in which they are incurred with no present worth conversion. It is informative to compare this value to the other high estimate values of ($10.6 million) and ($25.1 million), which differ from this case by adjusting these future costs into year 2012 dollars using 3-percent and 7-percent discount rates as described in section D.3.2.1.2.

D.4.2 Backfit Analysis

As discussed above, the NRC has determined that the reference plant would not achieve a substantial increase in the protection of public health and safety from a change to low-density spent-fuel-pool storage. The NRC has therefore determined that imposing a requirement to use only low-density spent fuel pool storage at the reference plant would not meet the requirements of the backfit rule. However, to ensure that there is a complete discussion of these issues, the NRC has drafted an analysis of the costs associated with imposing these requirements as a backfit for illustrative purposes. This analysis of the direct and indirect costs of implementing the new requirements provides an assessment of the costs associated with imposing these requirements and the relative safety benefits in terms of the NRC's backfit rule. This backfit analysis examines the impacts of requiring low-density spent fuel pool storage at the reference plant relative to the baseline used in the regulatory analysis, which consists of existing requirements including the recently issued orders.

This plant-specific backfit analysis differs from most NRC's backfit analyses in that the NRC is not imposing or proposing to impose any requirements on its licensees. Instead, the NRC is assessing the safety benefits and costs of hypothetical requirements that, if implemented, would result in the use of low-density spent fuel pool storage and a corresponding increase in on-site dry cask storage for the reference plant. An NRC rulemaking to impose requirements like the ones analyzed in this appendix would need to include a backfit analysis. This section of the appendix provides a discussion of some of the elements that would be analyzed as part of a backfit analysis of these requirements. Prior to imposing these requirements through a rulemaking the NRC would, at the very least, issue a separate regulatory bases for public comment. If it is determined that rulemaking is required, the NRC would issue a proposed rule for public comment.

Low-density Spent Fuel Pool Storage Alternative Requirements that Constitutes a Plant-Specific Backfit for the Reference Plant

- All spent fuel assemblies that have cooled for at least five years (older spent fuel assemblies) after discharge from the reactor core are expeditiously moved from spent fuel pool storage from spent fuel pool storage to dry cask storage.

- The completion of the initial movement of older spent fuel assemblies to dry cask storage is achieved within five years of the effective date of the requirement.

- Following each refueling outage, the older spent fuel assemblies stored in the pool shall be moved to dry cask storage in a timely manner.

In performing this analysis, the NRC considered the nine factors in 10 CFR 50.109, as described in the following subsections.

D.4.2.1 General Description of the Activity Required at the Reference Plant to Complete the Backfit

The alternative would require that the licensee of the reference plant incur upfront costs, including engineering, design, and licensing costs; equipment costs; construction costs; and start up and testing costs, as necessary for their independent spent fuel storage installation to accept the dry storage cask systems. The licensee would also need to purchase and load dry storage casks on a periodic basis in compliance with the regulatory requirement.

D.4.2.2 Potential Change in the Risk to the Public from the Accidental Offsite Release of Radioactive Material

Table 109 Public Health (Accident) Person-Rem Averted

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
	Low Est.	Best Est.	High Est.	Undiscounted	3% Net Present Value			7% Net Present Value		
				Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
Low-density storage	60	124	1,260	$247,700	$86,700	$179,500	$1,825,500	$60,200	$124,600	$1,267,000

Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

If the NRC were to implement the low-density storage proposal, the storage of spent fuel in dry storage casks would decrease the accidental offsite release of radioactive material from a postulated spent fuel pool accident. As Table 109 shows, dry cask storage at the reference plant would decrease the radiation exposure to the public by between 60 and 1,260 person-rem. The dose to the public mostly comes from the reoccupation of land after decontamination and the exposure to the workers who are decontaminating the public land. This analysis also assumes that 0.5% of the public will not evacuate during the accident. This resultant radiation dose is included within the public health exposure. As shown in the regulatory analysis, the best estimate benefits range from $0.18 million (3 percent net present value) to $0.12 million (7 percent net present value). A more in-depth review of the person-rem exposure to the public is found in section D.3.4.2.

D.4.2.3 Potential Impact on Radiological Exposure of Facility Employees

Table 110 Facility Employee Exposure

Case	Dose (averted person-rem)			Benefits (2012 dollars)						
	Low Est.	Best Est.	High Est.	Undiscounted	3% Net Present Value			7% Net Present Value		
				Best Est.	Low Est.	Best Est.	High Est.	Low Est.	Best Est.	High Est.
accident short-term	0.268	0.421	5.493	$841	$388	$628	$7,959	$269	$453	$5,524
accident long-term	0.068	0.208	5.170	$415	$98	$310	$7,490	$68	$223	$5,198
routine	-4.560	-4.560	-4.560	-$9,000	-$24,000	-$24,000	-$24,000	-$27,000	-$27,000	-$27,000
Total	-4.224	-3.932	6.103	-$7,744	-$23,514	-$23,063	-$8,552	-$26,662	-$26,324	-$16,278

Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

If imposed on licensees, these requirements would provide added assurance that nuclear industry workers are not subjected to unnecessary radiological or hazardous chemical exposures as the result of mitigative and clean-up activities associated with a spent fuel pool accident that results in a radioactive release. Storage of spent fuel in dry storage casks would decrease the post-accidental offsite radiation exposure to facility employees from a postulated spent fuel pool accident. The exposure of facility employees comes from a short-term dose, based on the exposure during the accident, and a long-term dose, based on the exposure from the onsite cleanup costs. Facility employees, however, receive additional radiation exposure during DSC loading and handling activities, ISFSI operations, and maintenance and surveillance activities, resulting in a net increase in radiation exposure as shown in Table 110 for the low and best estimates. A more in-depth discussion of the person-rem exposure to facility employees can be found in sections D.3.4.2 and D.3.4.3.

D.4.2.4 Installation and Continuing Costs Associated with the Backfit, including the Cost of Facility Downtime or the Cost of Construction Delay

Table 111 Installation and Continuing Costs Associated with the Backfit

Case	Costs (2012 dollars)		
	Undiscounted	3% NPV	7% NPV
Implementation costs	-$15,660,000	-$41,820,000	-$46,770,000
Operation costs	-$730,000	-$252,000	-$64,000
Total	-$16,390,000	-$42,072,000	-$46,834,000

Implementation and continuing costs include the upfront costs, which include engineering, design, and licensing costs; equipment costs; construction costs; and start up and testing costs, as necessary, for the reference plant's independent spent fuel storage installation to accept the dry storage cask systems. In addition, the licensee would need to purchase and load dry storage casks on a periodic basis in compliance with regulatory requirements. As these actions are assumed not to affect normal power operations, there are no assumed replacement energy costs or construction delays. A more detailed analysis of the industry implementation and operation costs is provided in sections D.3.4.6 and D.3.4.7.

D.4.2.5 Potential Safety Impact of Changes in Plant or Operational Complexity, including the Relationship to Proposed and Existing Regulatory Requirements

If imposed on licensees, these requirements are not expected to have a significant effect on facility complexity. The scheduling and performance of loading spent fuel assemblies from the spent fuel pool into casks and transporting them to the ISFSI would add additional complexity to plant operations, especially during the initial 5-year loading phase. The added plant operations complexity is not significant and will not substantially affect the reference plant operational practices or result in substantial indirect costs. However, should a cask drop accident occur during plant operation, even though its likelihood is remote, the event could challenge plant safety systems in mitigating the consequences.

D.4.2.6 Estimated Resource Burden on the NRC Associated with the Proposed Backfit and the Availability of Such Resources.

The establishment of the requirements needed to require the reference plant to move expeditiously all spent fuel assemblies that have cooled for at least five years (older spent fuel assemblies) after discharge from the reactor core from spent fuel pool storage to dry cask storage would require rulemaking. The rulemaking would not result in a substantial increase in annual expenditures of agency resources.

D.4.2.7 Potential Impact of Differences in Facility Type, Design, or Age on the Relevancy and Practicality of the Proposed Action

There is no expected significant differentiation in how individual plants would implement the requirement to expeditiously move all spent fuel assemblies that have cooled for at least five years (older spent fuel assemblies) after discharge from the reactor core from spent fuel pool storage to dry cask storage. If imposed on licensees, these requirements do not directly relate to the facility type, design, or age.

D.4.2.8 Whether the Proposed Backfit is Interim or Final and, if Interim, the Justification for Imposing the Proposed Backfit on an Interim Basis

This consideration is not relevant to the analysis at this time because no requirements are being proposed.

D.4.2.9 Other Information Relevant and Material to the Proposed Backfit

Table 112 Summary of Backfitting Net Benefits for Low-density Spent Fuel Pool Storage for All Initiator Events (within 50 miles)

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$247,700	$179,500	$124,600	$119,700	$86,700	$60,200	$2,520,000	$1,825,500	$1,267,000
Occupational Health (Accident)	$1,300	$900	$700	$700	$500	$300	$21,300	$15,400	$10,700
Occupational Health (Routine)	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000	-$9,000	-$24,000	-$27,000
Total Benefits	***$240,000***	***$156,400***	***$98,300***	***$111,400***	***$63,200***	***$33,500***	***$2,532,300***	***$1,816,900***	***$1,250,700***
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***
Net Benefit	**-$16,150,000**	**-$41,916,000**	**-$46,736,000**	**-$16,279,000**	**-$42,009,000**	**-$46,801,000**	**-$13,858,000**	**-$40,255,000**	**-$45,583,000**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

Table 112 summarizes the described benefits and costs associated with the proposed backfit to require the reference plant to expeditiously move all older spent fuel assemblies after discharge from the reactor core from spent fuel pool storage to dry cask storage. The analyzed alternative would also incur onsite and offsite property cost offsets from an accident. These cost offsets are summarized in Table 113

Table 113 Summary of Cost Offsets for Onsite and Offsite Property

Attribute	Total Cost Offsets								
	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Offsite Property	$723,300	$524,000	$363,700	$1,073,300	$777,500	$539,700	$4,587,800	$3,323,400	$2,306,700
Onsite Property	$10,400	$6,900	$4,300	$4,480	$2,950	$1,830	$378,600	$249,600	$155,800
Total Benefits	***$733,700***	***$530,900***	***$368,000***	***$1,077,800***	***$780,500***	***$541,500***	***$4,966,400***	***$3,573,000***	***$2,462,500***

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

Table 114 Combined Sensitivity Analysis of the Backfitting Net Benefits for Low-density Spent Fuel Pool Storage for All Initiator Events (extending analysis beyond 50 miles and using a Revised Dollar per Person-Rem Conversion Factor)

Attribute	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$3,566,900	$2,583,800	$1,793,400	$2,162,500	$1,566,500	$1,087,300	$31,471,600	$22,798,200	$15,823,400
Occupational Health (Accident)	$2,500	$1,900	$1,400	$1,300	$1,000	$700	$42,700	$30,900	$21,400
Occupational Health (Routine)	-$18,000	-$49,000	-$54,000	-$18,000	-$49,000	-$54,000	-$18,000	-$49,000	-$54,000
Total Benefits	***$3,551,400***	***$2,536,700***	***$1,740,800***	***$2,145,800***	***$1,518,500***	***$1,034,000***	***$31,496,300***	***$22,780,100***	***$15,790,800***
Industry Implementation	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000	-$15,660,000	-$41,820,000	-$46,770,000
Industry Operation	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000	-$730,000	-$252,000	-$64,000
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***	***-$16,390,000***	***-$42,072,000***	***-$46,834,000***
Net Benefit	**-$12,838,600**	**-$39,535,300**	**-$45,093,200**	**-$14,244,200**	**-$40,553,500**	**-$45,800,000**	**$15,106,300**	**-$19,291,900**	**-$31,043,200**

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

Table 114 summarizes the results of the combined sensitivity analyses that extended the backfitting net benefit analysis beyond 50 miles from the plant site and used a higher per person-rem conversion factor to monetize averted dose. The analyzed alternative would also incur onsite and offsite property cost offsets from an accident. These cost offsets for the combined sensitivity analysis are summarized in Table 115.

Table 115 Summary of Combined Sensitivity Analysis Cost Offsets for Onsite and Offsite Property

Attribute	Total Cost Offsets								
	Best Estimate			Low Estimate			High Estimate		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Offsite Property	$2,139,300	$1,549,700	$1,075,600	$4,968,300	$3,599,100	$2,498,000	$11,586,600	$8,393,400	$5,825,500
Onsite Property	$10,400	$6,900	$4,300	$4,680	$3,150	$2,030	$378,600	$249,600	$155,800
Total Benefits	***$2,149,700***	***$1,556,600***	***$1,079,900***	***$4,973,000***	***$3,602,300***	***$2,500,000***	***$11,965,200***	***$8,643,000***	***$5,981,300***

1. nc = not calculated
2. Results are expressed in current dollars (year 2012 dollars) except for the undiscounted cases, which are expressed in constant dollars.

D.4.3 Disaggregation

In order to comply with the guidance provided in Section 4.3.2 ("Criteria for the Treatment of Individual Requirements") of the Regulatory Analysis Guidelines, the NRC conducted a screening review to ensure that the aggregate analysis does not mask the inclusion of individual provisions that are not cost-beneficial when considered individually and not necessary to meet the goals of imposing these requirements on the reference plant. Consistent with the Regulatory Guidelines, the NRC evaluated, on a disaggregated basis, each new regulatory provision expected to result in incremental costs. Based on this screening review, the NRC did not identify any requirements needing further consideration. The NRC believes that each of these provisions described in section D.4.2 is necessary in the aggregate for the expedited transfer of spent fuel to DSCs. However, as noted above, the Commission has not found that accelerated transfer to DSCs to provide a substantial safety benefit, nor to be cost justified.

D.4.4 Safety Goal Evaluation

Safety goal evaluations are applicable only to regulatory initiatives considered to be generic safety enhancement backfits subject to the substantial additional protection standard in 10 CFR 50.109(a)(3).

The frequency of damage to the spent fuel is estimated to be range from 7.11×10^{-7} to 5.39×10^{-6} per reactor-year when considering all initiators that could challenge spent fuel pool cooling or integrity. These values, when compared to a target value of 1×10^{-4}, which is the quantitative health objective for latent cancer fatalities derived using reactor accident characterizations, represents a 0.71% to 5.39% of the overall frequency of core damage.

The frequency of a release of radioactive material to the environment is assumed to be the same as the frequency of spent fuel damage. The reactor building, which houses the spent fuel pool, does not provide a containment barrier similar to the containment structure surrounding the reactor core, especially under the conditions postulated to dominate the release of radioactive materials from spent fuel.

It is difficult to compare the estimated 7.11×10^{-7} to 5.39×10^{-6} per reactor-year release frequencies for the postulated spent fuel pool accident when considering all initiators to a target value of 1×10^{-5} per reactor year for a large early release frequency (LERF). The spent fuel pool source term is not similar to the core damage (or melt) source term for which the consequences of a spent fuel pool accident are predicted to have no early fatalities and public health risk is dominated by latent cancer risks resulting from long-term exposures. Because the analyzed spent fuel accident is a slow progression with at least eight hours before an environmental release occurs, and the resultant release is not expected to result in any offsite early fatalities, the analysis suggests that the spent fuel pool release does not fall within the definition of a large early release. Although this analyzed accident is different from a reactor accident, the spent fuel pool estimated release frequencies of 7.11×10^{-7} to 5.39×10^{-6} per reactor-year meets the 1×10^{-5} LERF guidelines.

Societal risk is based on the statistically expected number of early and latent cancer fatalities. The Safety Goals for the Operation of Nuclear Power Plants: Policy Statement (51 FR 28044) defines the early fatality area calculation as that within one mile from the site boundary. As discussed above, the resultant release is not expected to result in any offsite early fatalities. A ten-mile radius is defined for calculating latent cancer fatalities. The second quantitative objective of the Policy Statement is for the risk to the population in the vicinity of a nuclear power plant from an accident at a nuclear power plant should not exceed 0.1 percent of the sum of cancer fatality risks resulting from all other causes. Based on recent data (http://www.cancer.org/research/cancerfactsfigures/index) the total fatality rate from cancer in the U.S. is 580,350 per 315,747,500 persons (http://www.census.gov/popclock/) or a risk of 1.84×10^{-3} per year, which results in a safety goal of 1.84×10^{-6} per year. Using the bounding frequency of damage to the spent fuel of 5.39×10^{-6} per reactor-year, which considers all initiators that could challenge spent fuel pool cooling or integrity, and the conditional individual latent cancer fatality risk within a ten-mile radius is 4.4×10^{-4} yields a bounding latent cancer fatality risk of 2.37×10^{-9} of cancer fatality per year. This calculated value of 2.37×10^{-9} latent cancer fatalities per reactor-year associated with a spent fuel pool accident is less than represents a 0.13% fraction of the 1.84×10^{-6} per year societal risk goal value based on the calculation area specified in the Safety Goal Policy Statement.

Therefore, the risk of a spent fuel pool accident at the reference plant appears to meet the Safety Goal Policy Statement public health objectives. They also meet the $1x10^{-5}$ per reactor-year LERF guideline. Therefore, the Regulatory Baseline is justified for the alternative described in section D.2.2 as evaluated for the reference plant.

D.4.5 CRGR Results

This section addresses regulatory analysis information requirements for rulemaking actions or staff positions subject to review by the Committee to Review Generic Requirements (CRGR). All information called for by the CRGR is presented in this regulatory analysis.

D.5 DECISION RATIONALE

This section presents the decision rationale, including the basis for selection, any decision criteria used, the regulatory instrument to be used (if applicable), and the statutory basis for the selected regulatory action. The decision rationale is presented in two different ways, in order to address the differing decision criteria between regulatory analyses and backfit analyses (10 CFR 50.109).

D.5.1 Regulatory Analysis

Table 103 shows that a requirement for low-density spent fuel storage alternative does not achieve a cost-beneficial increase in public health and safety for the reference plant using the current regulatory framework when all event initiators, which may challenge spent fuel cooling or pool integrity, are considered. Furthermore, the three sensitivity studies provided in section D.4.1.4 also showed that the low-density spent fuel storage alternative was not cost-justified for any of the discounted sensitivity cases.

The NRC believes that there are other considerations discussed in section D.3.4.10 that would further reduce the quantified benefits and make the proposed alternative less justifiable. Based on the NRC's assessment of the costs and benefits, the agency has concluded that the risk due to beyond design basis accidents in spent fuel pools, while not negligible, is sufficiently low that the added costs involved with expediting the movement of spent fuel from the pool to achieve the low-density fuel pool storage alternative evaluated for the reference plant is not warranted.

D.5.2 Backfit Analysis

The NRC conducted a backfit analysis for the reference plant relative to the backfit requirements in 10 CFR 50.109 for illustrative purposes. The NRC does not believe that this alternative results in a cost-justified substantial safety enhancement for the reference plant. First, the risk of a spent fuel pool accident at the reference plant appears to meet the Safety Goal Policy Statement public health objectives. The estimated spent fuel pool accident release frequency is also less than the $1x10^{-5}$ per reactor-year LERF guideline. Second, the cost-justified criteria are not met when evaluating the averted accident consequences within 50 miles of the site consistent with the regulatory framework. Sensitivity analyses that extend the analyses beyond 50 miles also show that the low-density spent fuel storage alternative was not cost-justified for any of the discounted sensitivity cases. Therefore, the Regulatory Baseline is justified for the alternative described in section D.2.2 as evaluated for the reference plant.

In light of the findings above, the NRC concludes that the quantified safety benefits of the proposed requirements that qualify as backfits, considered in the aggregate, would not constitute a substantial increase in protection to public health or safety or the common defense and security, and the costs of these requirements would not be justified in view of the increase in protection to safety and security provided by the backfits embodied in these requirements on the reference plant.

D.5.3 Conclusion

The regulatory screening analysis and the backfitting discussion in this appendix indicate that for the reference plant a requirement for low-density spent fuel pool storage, and an associated requirement for expedited transfer of spent fuel from the spent fuel pool to meet a low-density spent fuel pool storage requirement, are not justified.

The risk due to beyond design basis accidents in the spent fuel pool analyzed in this study, is sufficiently low that the added costs involved with expediting the movement of spent fuel from the pool to achieve the low-density fuel pool storage alternative are not warranted. While the expedited fuel movement alternative evaluated is not cost-beneficial, the report has discovered that an alternative 1x8 high-density fuel configuration may have significantly lower costs in implementation and potentially similar benefits to the low-density configuration. This alternative should be evaluated further, in addition to other possible spent fuel pool loading configurations, as part of the regulatory analysis for expedited fuel movement described in SECY-12-0095 to evaluate the transfer of spent fuel to dry cask storage for existing and new (future) nuclear power plants.

D.6 REFERENCES

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APPENDIX E: SFPS PUBLIC COMMENTS SUMMARY

The Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor (SFPS) draft report was publicized in a press release on June 24, 2013, with a public comment period from July 2, 2013 through August 1, 2013. Comments related to the SFPS project covered a wide range of topics. This Appendix provides excerpts of the comments received along with NRC responses. All comments are located in the NRC's Agencywide Documents Access and Management System (ADAMS) with the below referenced accession numbers.

The comments below are organized by public commenter.

<u>From: Marv Lewis (ADAMS Accession No. ML13196A302, ML13207A141, and ML13211A177):</u>

1) **Comment:**

"The entire report seems to skirt the issue whether putting spent fuel pools on the roof of a reactor is logical or a design error"

NRC Response:

The elevated pool design used with the boiling water reactor (BWR) Mark I and II containment designs provides for the safe transfer of fuel from the reactor and the safe storage of fuel. The pool location allows for underwater transfer of fuel from the reactor vessel with substantial shielding provided by more than 10 feet of water above the fuel during transfer. The storage configuration maintains in excess of 20 feet of water above the stored fuel to provide for cooling of the fuel, protection from credible high-energy missiles that may fall into the pool, and mitigation of any radiological release that may result from a design-basis fuel handling accident. Systems attached to the pool provide for adequate cooling and have design features to prevent a substantial loss of coolant inventory under accident conditions, such as design-basis earthquakes and pipe breaks. The pool itself makes up a small part of the reactor building and, in addition to being constructed from very thick reinforced concrete walls with a leak-tight stainless steel liner, is surrounded by the reactor building structure, which consists of thick reinforced concrete walls at the level of the spent fuel pool and below.   The heavy structural members that support the enclosure above the refueling floor have been designed to withstand extreme natural phenomena, but, for the several facilities in this class that have relatively lightweight sheathing, the sheathing has been designed to separate from the enclosure to prevent over-loading the structure. Thus, the spent fuel pools have been designed to provide for safe storage and transfer of the fuel used in these BWRs.

2) **Comment:**

"Water supply is evaluated only in the proximity of the fuel pool"

NRC Response:

The study assesses situations where deployment of mitigation is successful, and where mitigation is not successful, considering only portable pumping equipment because the study assumes that the normal ac-powered systems will be unavailable. The cases in Chapter 6 with unsuccessful deployment of mitigation subsume instances where the water source (which as the comment indicates is not proximate to the refuel floor) is unavailable because of either an event-driven failure or random failure.

3) **Comment:**

I applaud the use of color and simple way that seismology is explained. The tables and figures are easily read and understood. The problem emerges in that the graphs and explanation may be an oversimplification. I do not see many 2 sigma and 6 sigma confidence levels on the graphs and figures. If there is any confidence in the figures, data and graphs, the confidence levels should be clearly marked. Is the reader to assume the data is unassailable or to assume there is no confidence in the data?"

NRC Response:

The USGS 2008 model does not provide the various confidence levels, thus the report doesn't include this information. However, the point estimates provided by this model and used in the study account for confidence levels by providing for each peak ground acceleration average hazard estimates which are associated with confidence levels above the 50th percentile confidence level of the median point estimates. As compared to the median estimates, the confidence level percentiles for the average hazard curves increase when the uncertainties increase. As shown in Figure 2 (Chapter 3) of the report, the hazard curve used in the study predicts average seismic hazards greater than the average hazard curves used in previous NRC studies for peak ground accelerations greater than about 0.4g.

4) **Comment:**

"I respectfully suggest an extension of the comment period"

NRC Response:

Because this is a research study we did not extend the comment period. This research study does not authorize any licensee action or set regulatory requirements. This study also does not establish any Commission policy. The comment period was appropriate for a research study and provided sufficient opportunity to receive comments.

5) **Comment:**

"Are there any configurations and accidents that can ignite zirconium if the surface, alloying or other parameters are outside of the design due to error?"

NRC Response:

Nuclear fuel is fabricated with a high degree of quality assurance, with very tight design tolerances. The only credible scenario under which a zirconium fire is expected to occur in the spent fuel pool (SFP) is during accidents that lead to loss of water. This is investigated in this report. The air oxidation kinetics models in MELCOR for Zr-based alloys (including Zirlo and M5) are based on the research sponsored by NRC and documented in NUREG/CR-6846.

From: Robert Vandenbosh (ADAMS Accession No. ML13190A353):

6) **Comment:**

"Study terminated after 3 days. One learns from Fukushima that adequate offsite emergency response cannot be guaranteed within 3 days"

NRC Response:

The assumptions regarding offsite support in this study are similar to those used in the State-of-the-Art Reactor Consequence Analyses (NUREG-1935). In NUREG-1935, the staff reviewed available resources and emergency plans for Peach Bottom (same reference plant as for this spent fuel study) and determined that adequate mitigation measures could be brought onsite within 24 hours, and be connected and functioning within 48 hours.

The 3 day truncation time was chosen to perform a detailed assessment of the accident progression and to capture the release characteristics. In all unmitigated cases studied in the report for which a zirconium fire is predicted to occur, fuel uncovery occurs before 48 hours (see page 81) and radiological releases occur before 72 hours.

In the event of a prolonged severe accident, radiation and other hazards could make termination of an ongoing SFP release challenging. The staff included a sensitivity case reported in Section 9.8 to investigate releases that continue beyond 72 hours.

7) **Comment:**

"Too much credit is given to mitigation. It is difficult to predict all scenarios that could compromise mitigation."

NRC Response:

The study analyzed scenarios with and without successful deployment of mitigation which reasonably characterizes the range of possibilities.

8) **Comment:**

“Inclusion in tables of cancer results assuming the existence of threshold doses goes against recommendations of radiation health professional organizations that LNT should be used.”

NRC Response:

Radiation health professional organizations differ in their assessment of how the consequences from low dose radiation should be reported. The SFPS uses the Linear No Threshold (LNT) model as the base for the reporting of results. The dose truncation methodology, introduced in the SOARCA analyses documented in NUREG-1935, is a sensitivity analysis. Because the health effects of low dose radiation are uncertain, this approach yields insights into how results might change if low dose radiation were less effective at cancer induction than predicted by the LNT model.

9) **Comment:**

“You give an example of a translation of a specific ground movement (0.12) into an earthquake magnitude. A similar example for your 0.7 g ground motion would be helpful.”

NRC Response:

The ground motion at a site is a function of both the magnitude of the earthquake and the distance from the source to the specific site as well as other factors such as source characteristics, how the regional geology transmits the seismic energy and how local geologic effects affect the ground motions. Calculation of the probability of exceeding a given peak ground acceleration at the site, accounts for the ground motions that may result from all potential seismic sources in the region around the site. Therefore, the calculated probability is an aggregate of the contribution of earthquakes with different magnitudes and distances to the site. Although a peak ground acceleration at the site is not generally associated with a single magnitude earthquake, probabilistic seismic hazard models provide an indication of the magnitude and range of the earthquakes that would contribute the most to the hazard at a given site. Section 3.3, first paragraph of page 37, of the report provides the magnitude and distance of the earthquakes expected to contribute the most to the 0.7g event by describing them as earthquakes with magnitude less than about 6.0 at distances of less than about 20 km.

<u>From: Union of Concerned Scientists (ADAMS Accession No. ML13210A139):</u>

10) **Comment:**

“In several tables and figures (e.g., Table 37, Figs. 96 and 97) you use the term "population-weighted" for Individual Latent Cancer Fatality Risk. I don't understand what this means for an individual. A search on the term "population-weighted" reveals it is never defined or explained in the text of the Report.”

NRC Response:

“Population-weighted, individual LCF risk" is the total amount of latent cancer fatalities predicted in a specified area, divided by the population that resides within that area. This definition has been added to Table 33 of the report.

11) **Comment:**

“Most of your High Density results are for a (1 x 4) configuration. But not all reactor pools are loaded in this configuration, and the length of time that regulations allow for achieving this density is said to be a secret (p. 208). Thus you are emphasizing a risk less than that actually associated with some pools.”

NRC Response:

The calculations in the report were for a specific reference SFP that implements a pre-arranged fuel loading pattern (see Table 15 and Section 9.3). Even though the specific time requirement for achieving the 1x4 arrangement is not publicly available information, the sensitivity calculations presented in Section 9.3 show that somewhat higher releases are expected in OCP1 and OCP2 for a contiguous fuel pattern, and the large releases for 1x4 in OCP3 are comparable to OCP2 (with contiguous fuel loading). The objective of this study was not to provide releases for all pools, but to specify a detailed accident progression analysis for a specific reference pool with additional sensitivity calculations to provide insights on releases from different loading patterns. In fact, Peach Bottom uses a 1x8 loading pattern, where most plants employ a 1x4 loading pattern.

12) **Comment:**

“The relative risk of these two storage options was the real need being served by the study, but the study utterly fails that goal”

“That was the central question that needed to be answered by the NRC’s study – what is the relative risk from the two onsite spent fuel storage options. That question remains shamefully unanswered in the study.”

“…relative risks and identified strategies and tactics that enhanced the risk reductions achievable… But this draft study fails to provide that very useful service.”

“This study could have, and should have, provided useful insights to the relative risk of spent fuel pool versus dry storage. The final report must remedy this fundamental flaw”

NRC Response:

Both storage options are safe and pose low risk to the public. The objective of this study was to compare the consequences of a beyond design basis earthquake on a representative spent fuel pool in two configurations, low and high density. The study concluded that both high and low density spent fuel configurations are safe, and that expedited transfer of spent fuel from the spent fuel pool to dry cask storage is not warranted for the SFP studied. Thus, the NRC staff does not agree with the statement that this study is fundamentally flawed.

13) **Comment:**

"In March 2012, the NRC ordered plant owners (of all plants, not just Peach Bottom Unit 3) to install instrumentation to monitor conditions inside the spent fuel pools at their facilities (see http://pbadupws.nrc.gov/docs/ML1205/ML12056A044.pdf). At the same time, the NRC also ordered plants owners to develop mitigation strategies to provide assurance of adequate cooling of reactor cores and spent fuel pools when permanent electrical supplies are unavailable for indefinite periods (see http://pbadupws.nrc.gov/docs/ML1205/ML12054A735.pdf). The NRC has applied a double standard"

"If the extreme earthquake scenario examined for a single reactor in the NRC's study is sufficiently thorough and bounding, what was the basis for the NRC's March 2012 orders that owners install spent fuel pool instrumentation and provide spent fuel pool cooling capabilities during extended power outages of infinite duration?"

"This NRC study wants us to believe that damage inflicted by an extreme earthquake on a highdensity loaded spent fuel pool poses little threat to public health and safety, even if it is not successfully mitigated within 3 days"

"The NRC's March 2012 orders and its June 2013 draft study cannot both be right"

NRC Response:

Based on this study and previous studies, NRC believes that spent fuel pools protect public health and safety. Orders EA-12-051 and EA-12-049 currently being implemented by all operating U.S. nuclear power plants should serve to further reduce spent fuel pool accident risk by increasing the capability of nuclear power plant operators to monitor spent fuel pool water inventory and mitigate beyond-design-basis external events.

14) **Comment:**

"The study's assumption that the battle is won or lost within 72 hours contradicts the mission time for other scenarios and the experience at Fukushima"

NRC Response:

Please see NRC response to Comment #6

15) **Comment:**

"The 30-day mission time is not applied across the board to every structure, system, and component in every scenario. When shorter mission times are applied, they are accompanied by solid justifications. The NRC study's 3-day mission time is an unverified assumption."

"The study's imposition of the 3-day mission time serves to dismiss other plausible scenarios that could cause damage to irradiated fuel in spent fuel pools after 3 days. The study must consider longer mission times, and other scenarios that longer mission times permit. Proper consideration of longer mission times and other scenarios might show the risk is low – but that and that alone would be the proper method for dismissing these scenarios. Dismissing them via an arbitrary, unjustified assumption is poor science."

"Other credible scenarios were summarily dismissed from consideration because they took longer than 3 days to play out"

NRC Response:

The NRC does not agree that the 3-day assumption is arbitrary or unjustified. In this regard, please see the response to Comment #6.

16) **Comment:**

"The extreme earthquake considered in the draft study may represent the fastest way to place the public in harm's way from a spent fuel pool hazard. But to fixate on it and exclude other scenarios seems to replicate the tunnel-vision that factored into the March 1979 meltdown at Three Mile Island Unit 2."

NRC Response:

The study is focused on a large seismic event (beyond the design-basis for the plant) because past studies (i.e., NUREG-1353 and NUREG-1738) have suggested that this is the largest contributor to SFP risk. In addition, in Appendix D (regulatory analysis), the staff also used calculated results from previous spent fuel pool studies to extend the applicability of this evaluation to include other initiators, which could challenge spent fuel pool cooling or integrity.

17) **Commen**t:

"At a time when Americans cannot board a commercial airliner with more than 3 ounces of shampoo in a single container and without first removing footwear and all outwear to thwart terrorism, this study seems woefully deficient in summarily dismissing any and all acts of malice."

NRC Response:

This study is a safety study of a severe earthquake impact on a reference SFP. Separate security studies were performed after 9/11 and are non-public. The sensitive unclassified and classified analyses were made available to the Advisory Committee on Reactor Safeguards, the Government Accountability Office, and the National Academy of Sciences.

18) **Comment:**

"But what if a spent fuel pool only partially drains? In that case, the temperature rise is not checked (as shown in Figure 53) or turned around (as shown in Figure 55). Thus, a partially drained spent fuel pool represents a greater hazard than a fully drained one. Again, the NRC draft study opted for the better of these two choices to consider."

"Yet there are other failure modes that may be caused by these lower accelerations, and these scenarios should be considered"

"Because a refueling cavity seal has failed in the past due to reasons other than forces resulting from an earthquake's ground motion, it is unjustified to merely assume such failure cannot possibly be caused by a seismic event."

"The study needs to examine the risk from draindown events, such as the one that happened at Haddam Neck, initiated by seismic events of less severity than that likely 1 in 60,000 year before dismissing such scenarios in favor of but one "bounding" case"

NRC Response:

The study did not analyze a partial draindown, which would correspond to a leak on the walls of the SFP, because the staff's structural analysis for the reference pool studied showed that the walls are not potential leak locations. As discussed in Section 4 of the report the structural analysis predicted the earthquake would cause relative motion between the pool walls and the pool floor that would have the potential to result in cracking in the reinforced concrete structure and liner tears at the bottom of the walls. The staff found the pool structure to be strong enough to prevent liner tearing that could lead to leakage at any other location in the pool. Other components, including the refueling cavity gates and the piping attached to the SFP, whose failure could have increased the rate of coolant loss, were evaluated and found to be sufficiently strong and flexible enough to resist the ground motion without leakage.

19) **Comment:**

"The study must consider scenarios other than that involving a 1 in 60,000 year earthquake leading to complete draindown of a spent fuel pool. Just as Three Mile Island demonstrated the fallacy of the large-break loss of coolant shield, the extreme earthquake obsession does not answer all the relevant questions that must be answered"

NRC Response:

The study is a plant-specific analysis focused on the large seismic event shown by past studies to dominate risk , and is not a comprehensive search for vulnerabilities. The objective of this study was to provide a detailed accident progression analysis for a specific pool with additional sensitivity calculations to gain insights on potential radioactive releases from different loading patterns.

20) **Comment:**

"Page 11 of the draft study states

> "The conditional probability of a Zircaloy cladding fire given a complete loss of water was found to be 1.0 for PWRs [pressurized water reactors] and 0.25 for BWRs [boiling water reactors] in high-density configurations based on differences in assumed rack geometry." A 1.0 conditional probability means that there's a 100 percent chance of a PWR spent fuel pool fire if it lost water. (Conditional probability literally cannot get any higher than that, yet it's curiously been omitted from the draft study.)"

"In this, and too many other instances, when faced with multiple choices, the NRC picked a nonconservative, non-bounding option."

"In this specific instance, the NRC must either examine a PWR spent fuel pool scenario in its final report or justify excluding a 100 percent conditional probability of fire along with its disastrous consequences"

NRC Response:

The study is a best-estimate analysis that did not intentionally bias the results in a conservative or non-conservative direction. The spent fuel pool study used a BWR as a reference plant for a detailed accident progression analysis resulting in a predicted range of releases. Therefore, any conclusions in the report only apply to the SFP considered in the study for a BWR with a Mark I containment. Any reference to PWR in the report is provided in the context of past studies in Section 1.7.

21) **Comment:**

"The final study must consider full core offloads (at least parametrically) – or the NRC should ban them from happening."

NRC Response:

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 (please see Table 3 for additional information). Moreover, the full core's decay heat is actually considered during the outage when the reactor and the SFP are hydraulically connected. The additional decay heat from the reactor is provided in Table 26 for OCP 1 and 2.

22) **Comment:**

"Table 3 of the study states that "Failure of nearby dams is not explicitly addressed." The disaster at Fuksuhima has been attributed to flooding from the tsunami caused by the earthquake rather than by earthquake damage directly. This study ignores that reality by assuming that an earthquake of severe magnitude likely to occur only once every 60,000 years (and yielding geometric mean acceleration of 0.7 g) has zero chance of causing nearby dam(s) to fail."

NRC Response:

As mentioned in Table 3 of the report, failure of nearby dams is not expected to cause flooding above grade level at the reference plant. If flooding were to occur, efforts to mitigate the accident could be challenged.

23) **Comment:**

"The final study must not summarily dismiss the criticality concern associated with a high-density spent fuel pool"

NRC Response:

The NRC does not agree that this concern has been summarily dismissed. As discussed in Section 2.3, the offsite consequences of a criticality event are believed to be less severe than the consequences of a prolonged uncovering of the SFP, and are expected to be bounded by the results of this study.

24) **Comment:**

"It is not apparent from the information in Tables 8 and 9 that removing 55 percent of the mass from spent fuel pools has only positive safety margin implications."

"The NRC must, as a minimum, explain qualitatively why it only considered one configuration in its seismic evaluations."

NRC Response:

The difference in the weight of the spent fuel assemblies and mass of water between the two configurations while not negligible is not expected to lead to significant differences in the calculated probability of damage to the liner. The differences in the estimation of liner failure probabilities from differences in those quantities, which are small in the context of the total loads on the pool, are expected to be within the range of variability of the response predictions for these beyond design basis loads. The study assumed the same mass of water for the two configurations without reducing this mass to account for the volume of the assemblies in either case. This provides an upper bound on the mass of the water in the pool for both configurations.

From: Mack Sim (ADAMS Accession No. ML13211A175):

25) **Comment:**

"The exclusion of a refueling gate seal failure is unwarranted. There is no argumentation for this exclusion in the study (beyond baldly stating such an event is "unlikely") and no study or experiment to support this exclusion is cited."

NRC Response:

The description of the gates system provided by the licensee of the reference plant to the study team during a site visit as well as engineering drawings shown by that licensee to the study team during the same site visit, showed that the gate facing the spent fuel pool has a backup gate with an air gap in between the two gates. Those description and drawings also showed that both gates have mechanical seals to prevent leakage. Passive mechanical means (i.e., do not depend on air pressure, ac power, or dc power) that are unlikely to fail under the earthquake conditions provide leaks tighteness by keeping the seals pressured between the gates and walls.

26) **Comment:**

"Hydrogen production from steam radiolysis (specifically, steam radiolysis taking place in bubbles on the surface of fuel elements in a pool that has lost circulation cooling) is not considered, which leads to overly optimistic predictions [with respect to] probability and timing of hydrogen combustion events."

NRC Response:

During a severe accident, cladding oxidation modeled in MELCOR is by far the most important source of hydrogen generation, and could be about two orders of magnitude higher than radiolysis (based on the decay heat levels in the SFP used in this study). For the scenarios leading to hydrogen combustion in this study, the time required to get to ignition is only a few hours following the onset of significant heatup. In case of boiloff scenarios, the time required to generate sufficient hydrogen to cause combustion is much longer than the 72 hour truncation time considered, and the significant steam generated during boiloff events would act to temporarily inert the building.

From: EDF - France (ADAMS Accession No. ML13225A582):

27) **Comment:**

"the study is performed for a BWR nuclear reactor, can you give us the characteristics of the fuel assembly (type, pitch, enrichment, weight) and the name of the nuclear power plant?"

NRC Response:

Table 15 provides the information on the assembly type. The study used Peach Bottom as a reference and used all of the available plant specific design and operational data.

28) **Commen**t:

"we would like to know the irradiation history of the fuel assemblies (burnup, number of irradiation cycle and intercycle, lenght of irradiation and intercycle in days)"

NRC Response:

As indicated in the Response to Comment # 27, this information is proprietary. But such information was used to calculate the radionuclides inventories as explained in Section 6.1.5 of the report.

29) **Comment:**

“the study considers configuration 1x4 and 1x8 (one hot fuel surrounded by 4 or 8 cold fuel), according to you is it possible to store spent fuel like these configurations (are all the permutations of fuels practically manageable in exploitation)?”

NRC Response:

In fact, Peach Bottom uses a 1x8 loading pattern, where most plants employ a 1x4 loading pattern. In the study, the base case used Peach Bottom plant data with a 1x4 loading pattern. The 1x8 loading pattern used at Peach Bottom was analyzed as a sensitivity and had more favorable results, in terms of coolability and radiological releases than a 1x4 loading pattern, although both are safe arrangements in terms of public health and safety.

30) **Comment:**

“for the configuration 1x4 high density, could you give us the spent fuel heat (KW) for hot and cold fuel?”

NRC Response:

Table 25 of the report provides a comparison of various decay heat levels (averaged over a number of assemblies).

31) **Comment:**

“the study concludes that for BWR fuel assemblies, the latent cancer fatal risk is very low. According to you, would the conclusion be similar if the study had been performed for PWR fuel assemblies? Is BWR more conservative than [PWR] for this study?”

NRC Response:

The results of this study only apply to the reference plant (BWR Mark I) SFP. However, Section 10 provides a comparison of the consequences of this study and past spent fuel pool analyses (see Table 62 for latent cancer fatalities information).

32) **Comment:**

"Figure ES-1 shows that for small leak without mitigation measures, maximum release are 42% Cs for high density and 3.1% for low density. How can you explain that the latent cancer fatal risk is similar and very low for the two cases although there is a great release of Cs for high density? What are your assumptions for the radiological consequences study on the population?"

NRC Response:

The consequence analysis includes evacuating and relocating people. Therefore, the predominant outcome of the SFP release is land contamination, rather than public health effects, because emergency planning procedures are assumed to be effective in reducing dose to the public under both scenarios.
Section 7 of the report provides a more detailed explanation of the study for offsite consequences. Section 7.2.2 discusses the individual latent cancer fatality risk.

33) **Comment:**

"did the study take into account a propagation of zirconium fire (Sandia National Laboratory experiment)?."

NRC Response:

Yes. The model development was based on validation of MELCOR against the BWR zircornium fire experiments as documented in NUREG/CR-7143.

34) **Comment:**

"could you briefly explain what the 10 CFR 50.54(hh)2 mitigation measures are?"

NRC Response:

Each NRC reactor licensee must develop its own set of strategies to mitigate the effects of potential explosions or fires, and such strategies must include those relating to fire fighting, the mitigation of fuel damage, and minimizing radiological releases. As discussed further in Section 5.3 of the report, to meet the 10 CFR 50.54(hh)(2) requirements, licensees typically rely on equipment such as additional electrical power sources and pumps to provide water to the SFP for cooling the spent fuel (as was required by NRC orders after the terrorist attacks of September 11, 2001).

35) **Comment:**

"Finally, in your conclusion you say that 'Analysis also shows that for the scenarios and spent fuel pool studied, spent fuel is only susceptible to a radiological release within a few months after the fuel is moved from the reactor into the spent fuel pool. After that time, the spent fuel is coolable by air'. Can you explain this phenomenon and particularly the assumptions that you made to conclude that the spent fuel is coolable by air? What are the spent fuel heats after a few month of storage in the pool?"

NRC Response:

In an unlikely event of a liner failure in the reference plant SFP, the structural analysis in Section 4 predicts the leakage to occur at the bottom of the pool. Without successful deployment of mitigation, the leakage will lead to an eventual draining of all water in the pool. This leads to the natural circulation of air through the fuel assemblies. The timing of a potential zirconium fire is based on detailed accident progression analysis documented in Section 6 of the report. While releases are predicted for OCP3 (see page 142), a zirconium fire is not predicted in OCP4 (see Figures 52-57). Therefore, the spent fuel is coolable by air for at least 72 hours in OCP4. Table 25 of the report provides a comparison of various decay heat levels (averaged over a number of assemblies).

36) **Comment:**

"Is there any containment protection in the pool?"

NRC Response:

The spent fuel pool is open to the refueling room as shown in Figure 42. The refueling room is part of the reactor building adjacent to the reactor containment. This building is not designed to withstand elevated pressures. Page 107 of the report provides a description of the reactor building and the failure pressure criteria for the blowout panels and the roof. Section 4 of the report provides more details on the reactor drywell and the spent fuel pool.

From AGREE New York (ADAMS Accession No. ML13217A130):

37) **Comment:**

"The Draft study is not responsive to the Court's specific direction in June 2012 regarding what must be thoroughly evaluated."

"However, the Court decision in June 2012, US Court of Appeals for the District of Columbia Circuit, concerning NRC's waste confidence decision, required NRC to thoroughly evaluate the following:

1. the environmental effects of failing to secure permanent disposal
2. the risks of spent fuel pool leaks
3. the consequences of spent fuel pool fires."

NRC Response:

The purpose of the SFPS is to determine whether accelerated transfer of older, colder spent fuel from the spent fuel pool at a reference plant to dry cask storage significantly reduces risks to public health and safety.

A separate NRC activity, the ongoing Waste Confidence rulemaking, will respond to the June 2012 decision by the U.S. Court of Appeals for the District of Columbia Circuit that vacated and remanded the NRC's Waste Confidence Rule. The Waste Confidence rulemaking will address the three deficiencies the Court—and the commenter—identified.

For more information about the Waste Confidence rulemaking, visit http://www.nrc.gov/waste/spent-fuel-storage/wcd.html.

38) **Comment:**

"A number of scientific studies have been completed by the Nuclear Regulatory Commission and the National Academy of Sciences which document the severity of the problems of overcrowded spent fuel pools and the potential catastrophic risks. This body of scientific work, which includes that of Allison Macfarlane, the current Commission Chairperson, cannot now simply be ignored by the NRC."

NRC Response:

The SFPS builds upon numerous past studies of spent fuel pool risk as discussed in Section 1.7 of the report.

From: Charles Pennington (ADAMS Accession No.ML13217A132):

39) **Comment:**

"I question the use of such highly conservative "safety analysis" codes (thermal, structural, source and release calculations, especially in the dispersion, inhalation, ingestion, direct dose rates, population dose, etc. determinations)"

"It appears there has been no "sanity check" of the individual case results with an actual event such as Chernobyl"

"Such inconsistencies and significant modeling conservatisms should be carefully reviewed as to whether the displayed results are consistent with the modeling (a good QA review is probably in order) and whether they should appear in this report as has been said by others, these results are not hyper-conservative; they are wrong."

"It would seem logical, if modeling is the issue here (such safety analysis models classically being unable to project reasonable outcomes, which is especially troubling when used to discuss BDB events with the public), that for at least the dose consequence portion of the analyses, simple use of Chernobyl correlations would result in still conservative dose projections without the need to overstate impacts on the public; I have developed correlation models myself for that purpose, and that is the course I would recommend."

NRC Response:

The accident progression analysis contains no intentional conservatisms. MELCOR and MACCS2 represent state of the art codes for evaluation of accident progression and accident consequences, respectively.

MELCOR is the NRC's best estimate tool for severe accidents analysis, and has been validated against experimental data. MELCOR has the capability to mechanistically model deposition of aerosols on the structures within the reactor building. This is factored into the building decontamination factor as shown in Figure 86. Section 6.1.1 of the report details the modeling approach used for this analysis including the experimental basis for the breakaway oxidation kinetics model as well as other required models when modeling SFPs. The zirconium fire experiments were used to help validate MELCOR (see NUREG/CR-7143). The code's predictions showed good agreement with the experimental data for the initiation and propagation of zirconium fire.

The MACCS2 code has been compared to a number of alternate atmospheric transport and deposition codes. These include: Gaussian puff models and a state-of-the-art Lagrangian particle tracking code, for estimating concentrations and deposition out to distances as great as 100 miles from the point of release. The study was documented in NUREG-6853. Generally, MACCS2 performed as well as either of the Gaussian puff models when compared with the state-of-the-art Lagrangian code for calculating mean consequence results.

Extrapolation from the results of historical releases of radioactivity to the environment, such as the Chernobyl accident, to yield estimates of the exposures and risks of fuel pool accidents would require great care in order to yield valid insights. This is due to major differences in source term characteristics (e.g., heat content, aerosol characteristics), transport characteristics, and exposure factors (such as, shielding factors, dietary habits, and protective action strategies). However, the NRC is participating in a number of international and domestic efforts underway to benchmark codes against available Fukushima data, although releases from spent fuel pools were not experienced and therefore not available for comparison.

40) **Comment:**

"The fact that the most highly regarded nuclear regulatory body in the world still uses LCFs is simply beyond credulity."

NRC Response:

Individual latent cancer fatality (LCF) risk is computed to provide a perspective on how the accident evaluated in the report could affect societal risks. Such use of LCF risk data is consistent with the NRC safety goal policy statement.

41) **Commen**t:

"Finally, I would suggest that some part of the report provide more discussion of the doses generated by the events and the risk issue, as well as offering the public a comparison of the collective doses from this report with the collective doses the public receives every year from a selection of non-nuclear industries (out of the 15 to 20 that are most impactful on U.S. population doses)."

NRC Response:

The NRC Safety Goal regarding LCF risk from nuclear power plant operation (i.e., 2x10-6 or 2 in 1 million per year) is set 1,000 times lower than the sum of cancer fatality risks resulting from all other causes (i.e., ~2x10-3 or 2 in 1 thousand per year). As discussed further in the report's Executive Summary, this study estimated that the likelihood of a radiological release from the spent fuel pool resulting from the selected severe seismic event the study analyzed is on the order of one time in 10 million years or lower. This in part is a reason why the LCF risks within 10 miles in this study are low, in the range of one in a trillion (10-12) to 1 in 10 billion (10-10) per year for the analyzed scenario. Comparisons of the individual LCF risk within 10 miles calculated in this study to the NRC Safety Goal are provided in Figure ES-3. Such comparisons provide context that may help the reader to understand the contribution to cancer risks from the accident scenarios that were studied. The results of this study are scenario-specific and related to a single spent fuel pool. However, staff concludes that since these risks are several orders of magnitude smaller than the 2x10-6 (2 in 1 million) individual LCF risk – which corresponds to the safety goal for latent cancer fatalities – it is unlikely that the results here would contribute significantly to a risk that would challenge the Commission's safety goal policy.

From: Electric Power Research Institute (ADAMS Accession No. ML13217A133):

42) **Comment:**

"The SFPS is a consequences analysis as opposed to a full risk assessment. This is highlighted and acknowledged in the title as well as several times throughout the report. A more comprehensive probabilistic evaluation, while more resource intensive, would provide greater refinement and insight and increase the usefulness of the study, such as in risk-informed regulatory activities."

NRC Response:

This study is intended to provide a detailed assessment of the consequences associated with spent fuel pool accidents, and to compare the consequences of high and low density pool configurations. Although the study did not examine all scenarios typically considered in a probabilistic risk assessment (PRA), it focused on a large seismic event (beyond the design-basis for the plant) because past studies have suggested that this is the largest contributor to SFP risk. In fact, any analytical technique, including PRA, will have inherent limitations of scope and method. Nevertheless, the staff is currently performing a comprehensive site Level 3 PRA for a US PWR as discussed in SECY-11-0089.

43) **Comment:**

"The SFPS acknowledges the potential for interactions between the reactor and the spent fuel pool, and identifies such interactions in Section 2.2. These interactions are not, however, evaluated within the scope of the study. This decoupling of the SFP from the reactor affects several parameters including the potential for hydrogen from a reactor accident to collect in the SFP area, with the possibility of combustion; the timing of assumed operator actions; and the timing of the general emergency order (potential to affect evacuation). Additional discussion or basis would be helpful."

NRC Response:

Although such interactions between the reactor and SFP may be significant, they are outside the scope of the current study. The possibility that a concurrent reactor event may preclude operator actions is part of the motivation for the "unmitigated" cases analyzed in the study. Further, Section 9.4 presents sensitivity calculations to show the importance of the reactor building in the progression of accidents in the SFP and the source term with a concurrent reactor accident. Moreover, the full core's decay heat is actually considered during the outage when the reactor and the SFP are hydraulically connected.

Additionally, as noted above, the staff is currently performing a comprehensive site Level 3 PRA for a US PWR as discussed in SECY-11-0089.

44) **Comment:**

"The report assumes a uniform spray flux of water over the top of the SFP. Depending on timing and locations at which operator actions would need to be taken (e.g., proximity to pool and radiation fields), it may be difficult to align the spray. It could also be difficult to achieve 100% coverage if debris has blocked the top of some fuel assemblies."

NRC Response:

The study analyzed scenarios with and without successful deployment of mitigation and thus reasonably characterizes the range of possibilities. Some aspects of the mitigation success are provided in Section 8 in the HRA analysis. As stated in Table 3 of the report, no significant debris generated by the seismic event is expected to enter the SFP based on the structural response of the building and overhead crane. In addition, as stated in Table 3, some debris could be generated and could fall into the pool as a result of hydrogen combustion. However, the occurrence of a hydrogen combustion event in this study denotes that the fuel in the SFP has already become uncovered and is undergoing a fission product release.

45) **Comment:**

"The option to spray into the SFP from outside the Reactor Building is possible if the refueling floor is open to the environment. Depending on the size of the assumed opening, this option could limit the ability to achieve full spray coverage of the pool, but might be more likely to result in atmospheric scrubbing of the radionuclides"

"Spraying into the refueling floor and spent fuel pool from an external position creates the potential for direct release to the environment that is not currently included. Any operator actions to ventilate the building are considered should be considered in this context as well."

"The report indicates that "for PBAPS, the capacities of the available equipment are somewhat higher. The use of … 200 gpm here attempts to account for uncertainties in the … spray that goes outside the boundary of the pool". However, it is not clear whether this particular conservatism could balance the potential lack of full spray effectiveness noted above."

NRC Response:

The study analyzed scenarios with and without successful deployment of mitigation which reasonably characterizes the range of possibilities. Section 8 in the HRA analysis provides some aspects of the mitigation success.

Regarding the comment on considering operator action to ventilate the building, and the need to capture the associated direct radiological release to the environment that might accompany this, the study did not consider negative or positive impacts of operator actions associated with ad hoc measures to ventilate the building. This is in keeping with the approach of focusing on mitigation measures covered by 10 CFR 50.54(hh)(2) (and more specifically those in the NRC-endorsed guidance document NEI-06-12, Revision 2). When modeled actions or phenomenologically-driven events did affect radiological release characterization (e.g., spray scrubbing, hydrogen combustion-induced building damage), these effects were mechanistically modeled.

46) **Comment:**

“Air cooling of the assemblies is dependent upon having clear channels for air flow. Debris from the Reactor Building superstructure or miscellaneous equipment on the refueling floor could conceivably fall into the pool, potentially damaging the liner or impeding local air flow in the SFP channels. The analysis does not consider debris interaction with the SFP resulting either from a large seismic event or from a hydrogen combustion event (see Section 1, Table 3 for technical basis). Additional technical justification or a sensitivity analysis may be desirable. We do note that experience from recent large seismic events, including Kashiwazaki-Kariwa 2007, Fukushima Daiichi, Fukushima Daini, Onagawa, and North Anna, indicates little or no debris generation and deposition into SFPs due to earthquake itself.”

NRC Response:

The occurrence of a hydrogen combustion event associated with the spent fuel itself in this study denotes that the fuel in the SFP has already become uncovered and is undergoing a fission product release. The reduction in flow area and losses associated with debris generated from a hydrogen combustion resulting from a reactor accident is explicitly considered in Section 9.4 of the report.

No significant debris generated by the seismic event is expected to enter the SFP based on the structural response of the building. In addition, as stated on page 25, some debris could be generated and could fall into the pool as a result of hydrogen combustion.

47) **Comment:**

“The reports states that “The cool gases leaving the lower regions of the building are not brought into thermal equilibrium with gases above the SFP.” (Page 107). This assumption allows for cooler air at the inlet of the fuel. In order to achieve the lower inlet temperatures other actions may be necessary such as opening doors or other access ways to enhance airflow. It may be useful to provide additional justification or sensitivity analysis.”

“Only the refueling floor is modeled in MELCOR (pp. 105, 107, 108). The equipment shaft from lower Reactor Building elevations is modeled as supplying the air inlet to the SFP. This leads to air temperatures lower than what might be expected. In practice, air ingress into the SFP racks may be limited by the flow area available into the bottom of the pool and the air temperature may be higher than assumed. A sensitivity evaluation may be useful to address this issue.”

NRC Response:

The analysis has shown that during the heatup of the fuel and especially during natural circulation once the baseplate clears, the atmosphere of the refueling room becomes very hot and the flow is out through the normal leakages and the open hatch region to maintain the pressure. If the reactor building is intact, oxygen is eventually depleted that would limit the oxidation of remaining cladding. If ventilation is established in time before significant heatup of the fuel, it is possible to prevent a zirconium fire; however, if ventilation comes late, then air ingress would supply additional oxygen that would lead to additional oxidation. Some of these insights can be inferred from Figures 130 and 131 in Section 9.4 of the report.

48) **Comment:**

"The crack in the concrete and the tearing of the liner are assumed to occur at the bottom of the SFP. The complete draining of the pool opens the possibility for air cooling, which could be reduced if the leak occurred above the bottom of the fuel. The impact of this assumption can be estimated where fuel assembly inlet flow is delayed, which results in higher fuel temperatures. Further discussion of this particular assumption would be useful."

NRC Response:

Please see the response to Comment #18.

49) **Comment:**

"The SFPS acknowledges the large uncertainties for several assumptions associated with location and size of the breach (e.g., finite element modeling of the concrete structure, modeling of the liner strain, modeling of the liner tear and the concrete cracking that would allow leakage flow, and calculation of the friction factor associated with the leakage pathway). Sensitivity calculations could be used to investigate the possible impact of changes to failure location or size. A more probabilistic treatment could produce more refined results."

NRC Response:

The size and location of the leak were not assumptions made by the staff in the study. As discussed in Section 4 of the report, the structural analysis of the reference spent fuel pool design predicted the liner tear to be at the bottom of the pool for two leak sizes with a range of draindown times. For the SFP studied the uncertainty on the location of the concrete cracking and liner tearing is small. The study acknowledges that uncertainties on the water leakage rate are high. Consideration of two significantly different leakage rates and associated draindown times addresses the consequential range of those uncertainties. Please also see the response to Comment # 18.

50) **Commen**t:

"The study predicts cesium releases following hydrogen combustion (approximately 18 hours into the accident) in high density loading but not in low density loading. The difference in outcome is associated with the higher hydrogen concentration in the base case (high density loading)."

NRC Response:

The NRC staff agrees. None of the low density cases produced enough hydrogen to initiate a hydrogen deflagration even though sufficient oxygen was present to sustain combustion.

51) **Comment:**

“The MELCOR calculation does not predict that hydrogen combustion will take place on the refueling floor. This result may be influenced by not including the entire reactor building in the model. A sensitivity study that accounts for sufficient oxygen from other parts of the building may produce additional insights.”

NRC Response:

For high density cases, MELCOR predicts hydrogen deflagration in the refueling bay (see the sensitivity to hydrogen combustion in Section 9.1 on the report).

52) **Comment:**

“The SFPS “did not consider hydrogen events originating from a concurrent reactor accident”.”

NRC Response:

Hydrogen combustion is considered as a part of the sensitivity analysis in Section 9.4 of the report.

53) **Comment:**

“Molten core concrete interactions (MCCI) at the bottom of the pool can result in higher release fractions for radionuclides. MCCI can also lead to hydrogen generation. While the sequence is very unlikely, some discussion in the analysis may be warranted.”

NRC Response:

The sensitivity with MCCI in Section 9.5 of the report was focused on the releases and chemical form of the radionuclides. This sensitivity assumed that the reactor building has already failed as a result of a concurrent reactor accident. The limitations of representing MCCI in SFPs using MELCOR are also discussed on page 222 of the report. Nevertheless, the large scale fuel damage and relocation occurs mainly for small leak scenarios, and in most cases, the reactor building is expected to fail as a result of the hydrogen generated from the spent fuel before the start of MCCI.

54) **Comment:**

“If the conditions of the reactor do not warrant prompt declaration of a General Emergency (GE) based on the emergency action limits (EALs), then the time to initiate public evacuation could be delayed. Generally, EALs do not include guidelines for declaring a GE for events involving the SFP. For irradiated fuel or SFP events the highest EAL is alert, although a GE is declared by the emergency director.”

NRC Response:

In general, staff acknowledges that uncertainty can be associated with the timing of protective actions (such as sheltering and evacuation). However, staff expects the specific site conditions assumed (those being the loss of all off-site and on-site ac power that is not expected to be restored for a prolonged period of time) would promptly warrant declaration of a GE, and that this would occur before other potential conditions arose. Also, staff notes that the largest releases from the baseline results in this study arise from slow leaks, and these releases offer relatively long periods (40 hours or more) prior to the onset of release. Because of these reasons and others, the results of this analysis are not expected to be very sensitive to the exact time of the GE declaration.

55) **Comment:**

"The conclusion that there are no calculated early fatalities would benefit from further discussion"

NRC Response:

Staff has identified some of the factors that, in its professional judgment, contribute to the modeled result of no early fatalities. Staff believes that this is due in part to the efficacy of protective actions such as sheltering, evacuation, and relocation, although staff acknowledges that other factors, such as the isotopic composition and duration of the release, may play an important role as well.

Staff believes that the results of the study are consistent with the NUREG-1738. For example, Appendix A4B of NUREG-1738 provides computed early fatalities as a function of time since shutdown for both high and low ruthenium source terms and for early vs. late evacuation. The largest numbers of early fatalities reported in Appendix A4B of NUREG-1738 are associated with releases shortly after shutdown, a 75% Ru release, and late evacuation of 95% of the population. The NUREG-1738 results for the lower Ru release and early evacuation all show substantially less than one early fatality.

Moreover, with respect to the potential effect of evacuation uncertainty on computed early fatalities, staff notes that detailed outputs of SFPS's consequence simulations include a cohort representing the 0.5 percent of the population that is assumed to not evacuate, although they are assumed to be subject to dose-dependent relocation following plume passage. These detailed results show that even for those who are not assumed to evacuate, no early fatalities are computed for any of the releases evaluated. Staff believes this result supports the view that uncertainties in the evacuation model are not likely to significantly affect the conclusion that the risk of early fatalities from spent fuel pool accidents is very low.

In summary, although a detailed analysis of the factors leading to the result of no early fatalities could be instructive, the staff believes that the results reported are reasonably robust and are adequate to support the conclusions of the report.

56) **Comment:**

"The portable equipment for mitigation is assumed to be available and able to be aligned to the SFP despite the potential priority use on the reactor."

NRC Response:

The study analyzed scenarios with and without successful deployment of mitigation, and thus it reasonably characterizes the range of possibilities.

57) **Comment:**

"The work area to align the equipment is assumed to be accessible despite the draining of the SFP, potential high radiation, and potential loss of building structural integrity."

NRC Response:

The study analyzed scenarios with and without successful deployment of mitigation that reasonably characterizes the range of possibilities.

58) **Comment:**

"The applicability of the HRA approach to the conditions of interest in the main report and appendices may merit some additional justification."

NRC Response:

The combination of SPAR-H and expert judgment approach to the HRA is appropriate for the study. Further refinement to the analysis would require performing a probabilistic risk assessment which is outside the scope of this analysis.

59) **Comment:**

"The main report and the associated detailed analysis are a very useful compendium of methods and associated results for the chosen scenario. However, extrapolating the analysis of this single scenario to the risk conclusions drawn in Appendix D introduces large uncertainties and additional assumptions. Explaining and quantifying these uncertainties should be considered."

NRC Response:

The objective of this report is to study the consequences of a beyond design basis earthquake on a representative spent fuel pool in two configurations, low and high density at the reference plant. This analysis uses information contained within the study, to evaluate whether the reference plant in the study will benefit from the change from high- to low-density storage configurations in the spent fuel pool.

This analysis calculates the potential benefit per reactor year resulting from expedited fuel transfer by comparing the safety of high-density fuel pool storage relative to low density fuel pool storage. The comparison uses the initiating frequency and consequences from the Spent Fuel Pool Study to indicate whether any changes in the NRC's understanding of safe storage of spent fuel are warranted. Given the uncertainties involved in performing this plant-specific regulatory analysis, low and high estimates were used to supplement the best estimate. For areas in which there was large uncertainties, a bounding approach was used. The staff also used calculated results from previous spent fuel pool studies (i.e., NUREG

1353 and NUREG-1738) to extend the applicability of this evaluation to include other initiators, which could challenge spent fuel pool cooling or integrity.

Appendix D discusses the methodology, inputs, and assumptions used.

60) **Comment:**

"With respect to insight #3 on p. 246, the "informed expectation" has not been compared against an accepted seismic hazard curve. While the analysis still provides insights, a more traceable seismic hazard would improve the study."

NRC Response:

The seismic hazard used in the study was derived using the USGS 2008 seismic hazard model. The hazard information is provided in Chapter 3 of the report, specifically in Figure 2 and Table 4. The seismic hazard curve provides hazard information for peak ground accelerations up to 1.0 g while Table 4 also provides information on the expected hazard for all events with peak ground accelerations greater than 1.0 g. The information in Figure 2 shows that the seismic hazard curve used in the study predicts higher seismic hazards for events with peak ground accelerations greater than about 0.4 g than the hazard curves used in past NRC studies. These higher hazards informed the selection of the seismic event for the study.

Additional seismic hazard information is expected to be developed by licensees and submitted to the NRC for review as part of licensees' responses to the NRC information request for resolution of the Fukushima Near-Term Task Force Recommendation 2.1: Seismic. When completed, reviewed and published that new hazard in conjunction with the results of the Spent Fuel Pool Study, is expected to provide further insights on seismic hazard issues.

61) **Commen**t:

"With respect to insight #5 on p. 246, the case is made that the hottest fuel should be stored in 1x8 patterns to minimize the time air cooling would not be effective. This insight was not the focus of the analysis and could benefit from further review or additional discussion to indicate the expected risk impact."

NRC Response:

The 1x8 sensitivity calculations were performed based on the reference plant's practice of fuel loading in this configuration. Conclusion 5 summarizes the advantages associated with dispersed fuel loading patterns (both 1x4 and 1x8). As the commenter points out, detailed analysis of the risk impact of the 1x8 loading pattern was not the focus of the study and is beyond its scope.

62) **Comment:**

"With respect to insight #17 on p. 249, the statement is made that "The risk due to beyond design basis accidents in the spent fuel pool analyzed in this study is sufficiently low that the added costs involved with expediting the movement of spent fuel from the pool to achieve the low-density fuel pool storage alternative are not warranted." This insight is not well supported by the main report because a comprehensive risk assessment has not been performed in the main report. Referencing Appendix D would be needed along with all of the assumptions and inputs used in Appendix D."

NRC Response:

As the comment indicates, Conclusion #17 is supported by the regulatory analysis and backfitting discussion in Appendix D of the report. Conclusion #17 is subject to the analysis model, data, inputs and assumptions in Section D.3 of Appendix D . This clarification has been made in Conclusion #17 the report.

63) **Comment:**

"To enhance the existing study some additional sensitivity cases and additional investigations may be useful, including:

i. Sensitivity of hydrogen combustion (MELCOR)
ii. Sensitivity of 1x fuel assembly pattern (MELCOR)
iii. Sensitivity to a contiguous(uniform) fuel pattern during an outage (MELCOR/MACCS2)
iv. Sensitivity to concurrent reactor accident (MELCOR) (loss of Reactor Building)
v. Sensitivity to SFP leakage location and magnitude (MELCOR/MACCS2)
vi. Sensitivity to molten core-concrete interaction (MELCOR/MACCS2)
vii. Sensitivity to radiative heat transfer (MELCOR) (low density case)
viii. Sensitivity to land contamination (MACCS2)
ix. Sensitivity to time truncation (MELCOR/MACCS2)
x. Sensitivity to Reactor Building leakage (MELCOR)
xi. Sensitivity to 50-mile radius assumption (MACCS2)
xii. Sensitivity to scenario duration (>72 hrs)

It seems appropriate that the sensitivities should also address whether these could affect the cost-benefit assessment (e.g., would they affect both configurations in a manner that the differences in consequences apply equally?)."

NRC Response:

These sensitivity calculations have been performed and documented in the report. The purpose was to show the range of responses that would be expected, and in some cases, the results were bounded by the results for the base case.

From: State of New York, Office of the Attorney General (ADAMS Accession No. ML13217A134):

64) **Comment:**

"As an initial matter, the State reiterates its request for more time to provide comments on this long-term Draft Study"

NRC Response:

Because this is a research study we did not extend the comment period. This research study does not authorize any licensee action or set regulatory requirements. This study also does not establish any Commission policy. The comment period was appropriate for a research study and provided sufficient opportunity to receive comments..

65) **Comment:**

"The State is concerned with the study's failure to perform a benchmarking or bounding analysis."

NRC Response:

The study is a best-estimate analysis that did not intentionally bias the results in a conservative or non-conservative direction. This study is intended to provide an estimate of the consequences from a postulated SFP accident in part to inform a regulatory analysis comparing the costs and benefits of expediting the transfer of fuel from the pool to dry cask storage. A bounding consequence analysis would have been less useful for this purpose.

66) **Comment:**

"The Draft Study relies upon assumptions that underestimate the likelihood of a spent fuel pool accident. For example, the Draft Study assumes that liner damage is the only way to cause a radiological release. Thus, the Draft Study fails to address other ways cooling water could be lost, such as water boil-off. Instead of examining partial boil-off and partial drain-down, the study focuses solely on rapid drain-down."

NRC Response:

The study did not analyze a partial draindown, which would correspond to a leak on the walls of the SFP, because the staff's structural analysis for the reference pool studied showed that the walls are not potential leak locations. As discussed in Section 4 of the report the structural analysis predicted the earthquake would cause relative motion between the pool walls and the pool floor that would have the potential to result in cracking in the reinforced concrete structure and a liner tear at the bottom of the walls. The staff found the pool structure to be strong enough to prevent liner tearing that could lead to leakage at any other location in the pool. Other components, including the refueling cavity gates and the piping attached to the SFP, whose failure could have increased the rate of coolant loss, were evaluated and found to be sufficiently strong and flexible enough to resist the ground motion without leakage.

The study investigated loss of water from boil-off, which did not lead to any releases in 72 hours after the earthquake. For the complete drain down, the study analyzed two cases, the moderate leak case, which leads to drain down in a few hours, and the small leak case, which leads to drain down in about 2 days. Based on the structural analysis performed, partial drain downs are not considered credible mechanisms for the reference SFP used in this study under severe earthquake conditions.

67) **Comment:**

"the Draft Study fails to address the impact of a prolonged loss of power. As the events at the Fukushima facilities demonstrated, natural disasters often cause prolonged power loss and equipment failures, and such events at multi-unit sites can have synergistic and cascading consequences; therefore, it is important for NRC to conduct an in-depth study of the consequences of such occurrences."

NRC Response:

The study analyzed scenarios with and without successful deployment of mitigation. All the scenarios without successful deployment of mitigation assumed a prolonged loss of power.

68) **Comment:**

"Additionally, the offsite consequence analysis in section 7 of the Draft Study is flawed. In particular, the Draft Study fails to use realistic input values for its MACCS2 analysis. For example, the Draft Study unreasonably relies upon "Sample Problem A" generic values developed decades ago for the Surry site in rural Virginia. Instead, Staff, in drafting the study, should have developed site-specific MACCS2 input values. The Draft Study underestimates land contamination, land interdiction, and displaced individuals."

NRC Response:

This study used input values specific for the analyzed reference plant.. The input parameters for the consequence analyses are based on those developed for Peach Bottom for the recently completed "State-of-the-Art Reactor Consequence Analyses" research project (NUREG-1935), as discussed in Chapter 7. Examples of site specific input parameters used include site specific population distributions, meteorology, and evacuation timing parameters including the effects of seismic events on local road networks.

69) **Commen**t:

"The State recommends that the NRC clarify the differences between Peach Bottom and other U.S. nuclear reactors, and that it make clear that the Draft Study is limited to the reference plant and not necessarily indicative of conditions at other plants."

NRC Response:

Section 1.3 of the SFPS report is devoted to explaining the use of the reference plant. In addition, the report makes clear in the Executive Summary and Conclusions that "The regulatory analysis for this study indicates that expediting movement of spent fuel from the pool does not provide a substantial safety enhancement for the reference plant. The NRC plans to use the insights from this analysis to inform a broader regulatory analysis of the spent fuel pools at all US operating nuclear reactors as part of its Japan Lessons-learned Tier 3 plan."

This study is intentionally based on plant-specific information for a particular site, as opposed to attempting to define a generic site that might bound a set of plants. This approach was taken because it provides the best context for examining SFP accident progression and release phenomenology in a realistic fashion, for the purpose of providing a better understanding of the factors that affect the characterization of SFP beyond-design basis accidents.

70) **Comment:**

"Finally, the State submits that the Draft Study's conclusion-that "expediting movement of spent fuel from the pool does not provide a substantial safety enhancement for the reference plant" is not adequately supported or explained in the Draft Study."

"The Draft Study does not give proper weight to the substantial benefits associated with low density spent fuel pool storage."

NRC Response:

Section D.5 of Appendix D, "Decision Rationale," explains that the low-density spent fuel storage alternative evaluated does not achieve a cost-beneficial increase in public health and safety for the reference plant using the current regulatory framework when all event initiators, which may challenge spent fuel cooling or pool integrity, are considered. Moreover, the three sensitivity studies provided in Section D.4.1.4 also showed that the low-density spent fuel storage alternative was not cost-justified for any of the discounted sensitivity cases.

The NRC also conducted a backfit analysis for the reference plant relative to the backfit requirements in 10 CFR 50.109. The NRC does not conclude that this alternative results in a cost-justified substantial safety enhancement for the reference plant. First, the risk of a spent fuel pool accident at the reference plant appears to meet the Safety Goal Policy Statement public health objectives. The estimated spent fuel pool accident release frequency also is less than the 1×10^{-5} per reactor-year Large Early Release Frequency guideline. Second, the cost-justified criteria are not met when evaluating the averted accident consequences within 50 miles of the site consistent with the regulatory framework. Sensitivity analyses that extend the analyses beyond 50 miles also show that the low-density spent fuel storage alternative was not cost-justified for any of the discounted sensitivity cases. Therefore, the Regulatory Baseline is justified for the alternative described in Section D.2.2 as evaluated for the reference plant.

In light of the findings above, the NRC concludes that the quantified safety benefits of the potential regulatory action provisions that qualify as backfits, considered in the aggregate, would not constitute a substantial increase in protection to public health or safety or the common defense and security. In addition the costs of this potential action would not be justified in view of the increase in protection to safety and security provided by the backfits embodied in the potential regulatory action.

From: Nuclear Energy Institute (ADAMS Accession No. ML13217A129):

71) **Comment:**

August 1, 2013 letter from Steven Kraft on behalf of NEI.

NRC Response:

This letter reflects on the study from NEI's perspective, but does not contain any request for changes or clarifications. Thus, no response is provided here.

From: State of Vermont, Department of Public Service (ADAMS Accession No. ML13217A131):

72) **Comment:**

"...the comment period be extended to allow for an adequate review of a complicated study."

NRC Response:

Becuase this is a research study we did not extend the comment period. This research study does not authorize any licensee action or set regulatory requirements. This study also does not establish any Commission policy. The comment period was appropriate for a research study and provided sufficient opportunity to receive comments..

73) **Comment:**

"First, the study finds that there is an increased risk of a radiological release from a spent fuel pool within the first few months after fuel is moved from the reactor into the spent fuel pool. The Department thus requests that the NRC research all measures that can be taken by nuclear power plants to reduce the risks of a radiological release during refueling when "hot" fuel is present in the spent fuel pool. The goal of the NRC should be to prevent any radiological release."

"Second, the study explicitly found that there are beneficial effects in the form of a reduction in offsite consequences, such as land contamination when a nuclear power plant moves from a high-density spent fuel pool loading scenario to a low-density scenario. Because measures can be taken to reduce the potential of land contamination which would also result in a reduction of radiation dose to the public and subsequent health effects, the NRC should consider immediately ordering reductions of spent fuel assemblies stored in a pool."

NRC Response:

NRC's regulatory framework is based on providing adequate protection of public health and safety. Land contamination is considered in the consequences and cost-benefit analysis. The results of this study and previous studies show that high density SFPs protect public health and safety in that the risk of an accident is low, and in the unlikely event of a SFP accident, no early fatalities are predicted and latent cancer fatality risk is low. The regulatory analysis in Appendix D of the report indicates that expediting movement of spent fuel from the pool does not provide a substantial safety enhancement for the reference plant. However, the NRC plans to use the insights from this analysis to inform a broader regulatory analysis of the spent fuel pools at all US operating nuclear reactors as part of its Japan Lessons-learned Tier 3 plan. In addition, Orders EA-12-051 and EA-12-049 currently being implemented by all operating U.S. nuclear power plants should serve to further reduce spent fuel pool accident risk by increasing the capability of nuclear power plants to mitigate beyond-design-basis external events.

From: Mark Kelly (ADAMS Accession No. ML13219B156):

74) **Comment:**

"The NRC draft study depends on information that was generated using MELCOR in inappropriate ways. Even with input of reliable data, MELCOR produces accurate projections for systems only under circumstances and conditions that are understood and modeled well by the program. The NRC descriptions of its use of MELCOR indicate that the NRC attempted to use MELCOR to predict outcomes under circumstances and conditions that are beyond its current capabilities. MELCOR may be the best currently-available program for this type of study, but MELCOR capabilities don't appear to be up to performance standards required for this type of NRC study of accidents."

"NRC relies on technical information from the nuclear industry, which sometimes fails to meet scientific standards necessary for studies to reliably inform decision makers. The NRC relied heavily upon industry data for the study. The NRC has failed to require the nuclear industry to maintain technical information on nuclear fuel rod cladding alloys and other materials."

"The draft NRC report relies excessively on NUREGs and other information sources which themselves rely on information that may not have been adequately maintained."

NRC Response:

MELCOR is the NRC's best estimate tool for severe accident analysis and has been validated against experimental data. Section 6.1.1 of the report details the modeling approach used for this analysis. The study relied on specific SFP models that have been integrated into MELCOR over the past 10 years. These models are supported by experimental data (e.g. new air oxidation kinetics as documented in NUREG/CR-6846). The comment states that NRC has relied only on technical information from the nuclear industry. This statement in not correct. For example MELCOR was validated against NRC-sponsored zirconium fire experiments conducted at Sandia National Laboratory (see NUREG/CR-7143).   Please see the response to Comment #83 for additional discussion on the use of MELCOR in the study.

Plant specific design information for the study was provided by the licensee. In addition, NRC staff visited the reference plant to gather and verify plant information including interviews with plant staff regarding plant equipment and procedures.

NUREGs are NRC or NRC contractor reports that have been reviewed and approved for publication. Only credible information sources were used in the study.

<u>From: Harmon, Curran, Spielberg + Eisenberg LLC (ADAMS Accession No. ML13225A397)</u>

75) **Comment:**

"Prior to publication of the Draft Consequence Study, NRC's technical credibility on the pool-fire issue was low. Over a period exceeding three decades, NRC had published bad analysis and hidden other analysis behind a veil of secrecy. Moreover, NRC failed to conduct sophisticated modeling and supporting experiments that could have resolved technical issues central to pool-fire risk, despite having an appropriate capability prior to 1990."

NRC Response:

This study provides updated, publicly available consequence estimates of a representative, postulated spent fuel pool severe accident from a beyond-design-basis earthquake under high-density and low-density loading conditions. MELCOR is the NRC's best estimate tool for severe accident analysis, and it has been validated against experimental data. Section 6.1.1 of the report details the modeling approach used for this analysis. The study relied on specific SFP models that have been integrated into MELCOR over the past 10 years. These models are supported by experimental data (e.g. new air oxidation kinetics as documented in NUREG/CR-6846). In addition, MELCOR was validated against NRC-sponsored zirconium fire experiments conducted at Sandia National Laboratory (see NUREG/CR-7143). Please see the response to Comment #83 for additional discussion on the use of MELCOR in the study.

The NRC conducted the SFP security assessments in a manner consistent with their sensitivity. The sensitive unclassified and classified analyses were made available to the Advisory Committee on Reactor Safeguards, the Government Accountability Office, and the National Academy of Sciences. None of those entities expressed concern with the classification of the SFP security assessments.

76) **Comment:**

"NRC's Draft Consequence Study seeks to create the appearance of being a comprehensive assessment of the risk of a pool fire. That image is conveyed by the structure of the Study, by the way the Study is described in its Foreword, Abstract, and Executive Summary, and by unequivocal statements that high-density spent-fuel pools protect public health and safety. In fact, the Study's scope is narrow. As a result, the Study's examination of pool-fire risk is incomplete, and cannot support the broad, unequivocal findings that the Study presents. This disjunction between the apparent and actual scope of the Study is misleading. Moreover, in specific instances, the Study is misleading and is designed to support pre-determined conclusions."

NRC Response:

The staff made clear throughout the report, (including in its Foreword, Abstract and Executive Summary), that the analysis and results are specific to a beyond-design-basis seismic event at a reference plant spent fuel pool connected to a boiling water reactor with a Mark I containment. For example, Chapter 2 of the report lists, in detail, the key limitations and assumptions of the study, and states that the study is a limited-scope consequence study rather than a full PRA.

77) **Comment:**

"Pretence of considering low-density storage: The Study does not consider the risk implications of reverting to low-density, open-frame racks. Instead, NRC misuses the phrase "low density" in order to create a false impression of the Study's scope. This pretence reflects pre-determined conclusions based on a "feeling"."

NRC Response:

The comment does not specify how the phrase "low density" was misused, and the staff had no intention of creating a false impression of the study's scope. Section 6.2 of the report describes in text and diagrams the staff's modeling of high density and low density racking. The comment correctly states that the study did not look at open-frame racks. This was made clear in the report at page 23, that "to get the full benefit of low-density racking, BWR fuel would likely need to have the channel boxes removed." Note that, for the reference plant, the BWR fuel assemblies channel boxes would impede crossflow even with open-frame racks.

78) **Comment:**

"Limited consideration of water-loss scenarios: The Study focuses its analysis exclusively on water-loss scenarios involving total drainage. By so doing, the Study ignores a substantial part of the pool-fire risk. For example, the Study makes no effort to determine how the presence of residual water could affect fuel ignition. Extrapolation of Study findings indicates that consideration of this issue would substantially increase the estimated risk."

NRC Response:

The study considered the consequences of a very large earthquake at the reference plant. As discussed in Section 4 of the report the structural analysis for this event predicted the liner tear to be at the bottom of the pool for two leak sizes with a range of draindown times. This provided the boundary condition for the accident progression analysis.
Please also see response to Comment #66.

79) **Comment:**

"Limited consideration of initiating events: The Study considers only one type of initiating event – an earthquake. That narrow focus reflects a pre-determined conclusion that earthquake is the dominant contributor to the risk of a pool fire."

NRC Response:

The study is focused on an earthquake because it is the event shown by past studies to dominate spent fuel pool risk. (See NUREG-1353 and NUREG-1738)

80) **Comment:**

"No consideration of attack: The Study ignores the potential for an attack on a pool and/or adjacent reactor to initiate a pool fire. Yet, the probability of an attack with a substantial likelihood of success is at least equal to the probability of the severe earthquake that the Study does consider. Thus, the Study significantly under-estimates pool-fire risk."

NRC Response:

Please see NRC response to Comment #17

81) **Comment:**

"No analysis of risk linkages among pools and reactors: The Study identifies the potential for risk linkages, but does not properly analyze them. For example, the Study does not analyze a situation in which onsite radioactive contamination and other impacts of a reactor core melt would preclude mitigating actions that might prevent a pool fire. Yet, the probability of a core melt at an adjacent reactor is at least equal to the probability of the severe earthquake that the Study does consider. Thus, the Study significantly under-estimates pool-fire risk."

NRC Response:

The possibility that a concurrent reactor event may preclude operator actions is part of the motivation for the "unmitigated" cases analyzed in the study. Section 9.4 presents sensitivity calculations to show the importance of the reactor building in the progression of accidents in the SFP and the source term from a concurrent reactor accident. More explicit treatment of multi-unit issues between SFPs and reactors is being pursued under the NRC's comprehensive site Level 3 PRA project discussed in SECY-11-0089. The NRC staff does not agree that the study significantly under-estimates pool-fire risk.

82) **Comment:**

"Misleading statements regarding mitigating actions: The Study concedes that its analysis of the feasibility of mitigating actions is very limited. Yet, the Study makes unequivocal statements about this feasibility. Some of those statements are misleading, and reflect pre-determined conclusions."

NRC Response:

The NRC staff had no intention of making misleading statements regarding the feasibility of mitigating actions. The uncertainty with the feasibility of mitigating actions is part of the motivation for the “unmitigated” cases analyzed in the study. The study reports in Section 6 both "mitigated" and "unmitigated" scenarios to show the effectiveness of mitigation. The assumptions regarding mitigation are discussed in Section 5.3 and Section 5.3.2 details the rationale for reporting the unmitigated results. The study treats situations where deployment of mitigation is either successful or not successful, considering only portable pumping equipment since the study assumes that the normal ac-powered systems are likely to be unavailable. As Chapter 8 explains, the Human Reliability Analysis (HRA) studies the human performance elements of mitigation along with binning scenarios based on ac and dc power assumptions. But the HRA does not include a system failure analysis for the equipment (such as water tanks or the fire header). These types of failures are subsumed by the cases without successful deployment of mitigation.

83) **Comment:**

“In the Study, NRC employs the MELCOR code to model phenomena related to a pool fire – including heat transfer, cladding ignition, and fire dynamics. MELCOR findings are significant to NRC’s estimation of pool-fire risk. Yet, the validity of MELCOR in this context, and the appropriateness of NRC’s input assumptions, have not been tested through a process of open scientific inquiry. There are significant issues that should be addressed through such a process, including MELCOR’s simplified treatment of radiative heat transfer.”

NRC Response:

This response discusses both the validity of the MELCOR code for SFP analysis, and some of the input assumptions and results of the accident progression in the reviewer detailed comments (see ADAMS Accession No. ML13225A397) .

MELCOR is the most widely-used regulatory severe accident code in the world. It represents the current state of the art in severe accident analysis, and has been developed through NRC and international research performed over the past three decades. In addition to domestic use, the code is provided by NRC to international organizations through bilateral agreements under the Cooperative Severe Accident Research Program (CSARP).

MELCOR has been benchmarked against many experiments including separate and integral effects tests for a wide range of phenomena. Any new application of MELCOR requires targeted assessment of the code. The SFP models in MELCOR have been developed over the past 10 years, and are supported by experimental validation. The air oxidation kinetics models in MELCOR for zirconium-based alloys (including Zirlo and M5) are based on the research sponsored by NRC and documented in NUREG/CR-6846. The MELCOR model does not make assumptions about the temperature at which a zirconium fire is initiated, but rather relies on experimental data (see Section 6.1.1) to correlate the timing of breakaway oxidation. The generalized thermal radiation model (between any groups of assemblies modeled as individual rings) was developed to have flexibility in the modeling of the spent fuel assemblies, so no need exists for the assumption of a “cylindrical” geometry. MELCOR provides a best estimate prediction of fuel heat up considering all the heat transfer mechanisms such as radial radiation between assemblies, convective cooling by steam and air, radiation and convective heat transfer to the pool liner, and heat transfer from the exposed fuel assemblies to an underlying water pool.

MELCOR was used in the zirconium fire experiments (see NUREG/CR-7143) and the predictions showed good agreement with data for the initiation and propagation of zirconium fire. The publication of experimental results in NUREG/CR-7143 (including code to code comparisons) as well as the present study and review by the Advisory Committee on Reactor Safeguards supports the adequacy of MELCOR's use for this purpose.

The main focus of the present report was on progression of severe accidents following a loss of coolant inventory in a SFP, resulting in high temperature oxidation of cladding (and in some instances large scale melting and relocation of the assemblies) and associated release of fission products. Uncertainties in modeling of the accident progression (e.g., hydrogen combustion, radiative heat transfer) have been considered as part of sensitivity studies showing a range of fission product release. The model limitations (e.g., lack of a cladding deformation model) are documented in the report. In some instances, these limitations are not expected to change the results significantly. For example, in OCP3, the code calculates large scale zirconium fire and fuel degradation, and release from the fuel. However, in OCP4, temperatures remain below 700 C (see Figures 52-55 in the report reproduced as Figures VII-2 and VII-3 by the reviewer).

For the layout of the assemblies in the pool (see Figure 44 in the report reproduced as Figure VII-1 by the reviewer), this is a stylized assumption regarding the 1x4 loading pattern for both the high density and low density pool to compare the consequences. In fact, as stated in the report, the reference plant employs a 1x8 arrangement. This is a valid assumption since there is relatively less variation in the decay heat of the older assemblies and it was appropriate to group all the older assemblies in a few computational rings since the risk from the zirconium fire is dominated by the hot fuel. For the study, individual assembly data were provided by the licensee that were subsequently used for an inventory calculation using the computer code SCALE. The breaking up of the freshly discharged fuel (284 assemblies) into two rings was to account for the variations in decay heat of the individual assemblies with different burnups based on the actual assembly data as discussed in Section 6.1.5. The calculations in the present report were for a specific SFP with pre-arranged fuel loading pattern. This fact is acknowledged in the report in Table 15 and in Section 9.3. Even though the specific time requirement for achieving the 1x4 arrangement is not publicly available information, the sensitivity calculations presented in Section 9.3 show that somewhat higher releases are expected in OCP1 and OCP2 for a contiguous fuel pattern, and the large releases for 1x4 in OCP3 are comparable to OCP2. The objective of this study was not to provide a detailed analyses for spent fuel pools, but to provide a detailed accident progression analysis for a specific pool with additional sensitivity calculations to offer insights on releases from different loading patterns.

84) **Comment:**

"In the Study, NRC has erected an elaborate superstructure of analysis on a weak foundation of basic knowledge about pool-fire phenomena. This superstructure culminates in a regulatory analysis. As discussed in paragraph VIII-2, above, the findings emanating from this superstructure lack scientific credibility and are misleading. Thus, the design of the Study is fundamentally and irredeemably flawed."

NRC Response:

As summarized in Chapter 1 of the report, the NRC has been studying SFP risk for many years, including the performance of zirconium fire experiments and MELCOR benchmarking (NUREG/CR-7143). The NRC team conducting the study included senior engineers in each of the key technical areas (seismic, structural, severe accident, consequence analysis, and probabilistic considerations).

The statements above that the study lacks "scientific credibility" and that its design is "fundamentally and irredeemably flawed: are not supported by the Advisory Committee on Reactor Safeguards (ACRS), which the NRC views as an independent body. In the ACRS letter recommending that the study be published (ML13198A433), the ACRS stated in part:

- The SFPS has been performed in a thorough and systematic manner, and provides a state-of-the-practice assessment of the consequences of a beyond-design-basis seismic event on the spent fuel pool in a reference boiling water reactor containing either high-density or low-density fuel loading.
- The SFPS provides sound approaches, tools, and insights for a broader evaluation of the consequences of severe seismic events on spent fuel pools of different design and will be valuable in determining whether expedited transfers to dry cask storage systems (DCSSs) produce substantial safety benefit for U.S. BWRs and PWRs.

85) **Comment:**

"The Study addresses an issue that is significant in terms of public health and safety. This significance is illustrated by the Study's finding that a pool fire could lead to long-term displacement from their homes of more than 4 million people. Thus, citizens deserve a much better analysis of pool-fire risk than the incomplete, misleading work presented in NRC's Draft Consequence Study."

NRC Response:

The possibility of a zirconium fire does not contradict a finding of low risk. Risk is the product of likelihood and consequence, and can only be evaluated by considering both of these components. Moreover, the potential that beyond design-basis events might lead to a radiological release does not contradict a finding of reasonable assurance of adequate protection.

Five Japanese nuclear power plant sites with a combined total of 20 reactors and 20 SFPs were subjected to severe ground motions from two major earthquakes in the past 6 years (i.e., March 11, 2011 Tohoku and July 16, 2007 Niigataken Chuetsu-Oki earthquakes). The operators of these 20 SFPs reported no water leakage from the pools after these earthquakes.

86) **Comment:**

NRC's Draft Consequence Study should be scrapped.

NRC Response:

See response to Comment 84.

87) **Comment:**

"In addressing the pool-fire issue, NRC should focus its initial attention exclusively on establishing a solid technical understanding of phenomena directly related to a potential pool fire. To do this, NRC would start with a clean slate and use the best available modeling capability backed up by experiment. This modeling and experimental work would be done according to scientific principles."

NRC Response:

Starting with a "clean slate" would ignore 35 years of accrued knowledge as summarized in Chapter 1 of the report. Please also see the response to Comment #83.

88) **Comment:**

"the Study proceeds to make unequivocal statements about the feasibility of mitigation. For example, in addressing the potential for a boil-off scenario of water loss, the Study says that the probability of mitigation failure extending for 7 days is "negligible".[43] That statement is based on no analysis, and reflects a pre-determined conclusion. NRC ignores, for example, the possibility that radiation fields and other onsite impacts of a reactor core melt could preclude mitigation for an extended period."

Footnote 43: Barto et al, 2013, page 175.

NRC Response:

The study analyzed scenarios with and without successful deployment of mitigation that reasonably characterizes the range of possibilities. Please also see the NRC response to Comment #6