ENT – O'Kula Source Term Testimony

January 3, 2011

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION,

Before the Atomic Safety and Licensing Board Panel

In the Matter of	)	
	)	
Entergy Nuclear Generation Company and	)	Docket No. 50-293-LR
Entergy Nuclear Operations, Inc.	)	ASLBP No. 06-848-02-LR
	)	
(Pilgrim Nuclear Power Station)	)	

Testimony of Dr. Kevin R. O'Kula on Source Term Used in the Pilgrim Nuclear Power Station Severe Accident Mitigation Alternatives (SAMA) Analysis

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I. WITNESS BACKGROUND

Q1: Please state your full name.

A1. (KRO) My name is Kevin R. O'Kula.

Q2: By whom are you employed and what is your position?

A2. (KRO) I am an Advisory Engineer with URS Safety Management Solutions ("URS") LLC.

Q3: Please summarize your educational and professional qualifications.

A3. (KRO) My educational and professional qualifications are provided in the "Testimony of Dr. Kevin R. O'Kula and Dr. Steven R. Hanna on Meteorological Matters Pertaining to Pilgrim Watch Contention 3," January 3, 2011 (referred to in this testimony as the "Meteorological Testimony").

II. PURPOSE OF TESTIMONY

Q4: What is the purpose of your testimony?

A4. (KRO) The purpose of my testimony is to respond to Question 2 from the Atomic Safety and Licensing Board ("Board") in Appendix A of the September

23, 2010 Board Order concerning the source term, and resultant deposition, used in the Pilgrim Severe Accident Mitigation Alternatives ("SAMA) analysis. The Board's question has three subparts and I will answer each subpart in turn.

III. RESPONSES TO BOARD QUESTION 2

Q5: Regarding the radioactive contamination to be computed from the dispersion and deposition caused by the meteorological patterns at issue, describe in sufficient detail for scientific understanding: (a) How the source term to be used for each computation of radioactivity dispersion and deposition is determined (i.e., what is the frequency distribution of source terms used in SAMA analyses for the Pilgrim Plant, and how is a particular source term selected for each dispersion/deposition computation?

A5. (KRO) The frequency distribution and associated source term for the Pilgrim SAMA analysis are based on the most recent plant-specific Probabilistic Safety Assessment (PSA) available at the time of submittal of the Environmental Report. The PSA model is referred to as the Pilgrim Nuclear Power Station (PNPS) PSA, and incorporates an updated Pilgrim individual plant examination (IPE) and a supplemental off-site consequence analysis using the Version 1.13.1 of the MACCS2 code. In general, the Pilgrim License Renewal Application (LRA)[1] as updated by subsequent LRA amendments that were made in response to Nuclear Regulatory Commission ("NRC") Requests for Additional Information[2] is the source of the information used for the SAMA analysis. The frequency distribution and source term aspects of the PSA model are described below.

Overview of PSA Analysis

As described in A18 of the Meteorological Testimony, severe accident risks are determined using plant specific PSA models to assess what can go wrong, how likely is it, and what are the resulting consequences. The models are applied in sequential phases of the PSA analysis, referred to as Level 1, Level 2 and Level 3

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1 The Pilgrim "Severe Accident Mitigation Alternatives Analysis" (Exhibit ENT000006), which is Attachment E to the Pilgrim LRA ER, describes the frequency and source term used in the Pilgrim SAMA analysis.

2 Subsequent to the filing of the LRA, there were four LRA amendments which amended the SAMA analysis, including the PSA, as it appears in the LRA ER. These are LRA Amendment 4 (Exhibit ENT000007); LRA Amendment 7, Attachment D (Exhibit ENT000008); LRA Amendment 9, Attachment E (Exhibit ENT000009); and, LRA Amendment 10, Attachment C (Exhibit ENT000010). The LRA ER itself was not updated to incorporate these amendments.

PSA. Severe accidents are postulated events that progress beyond those accounted for in design basis documentation (e.g., Final Safety Analysis Report (FSAR)) for a given plant. This type of reactor accident is more severe than design-basis accidents (DBAs), and includes those in which substantial damage is done to the reactor core, regardless of whether serious off-site consequences occur. A key function of a SAMA analysis is to identify additional potentially cost-beneficial measures to prevent or mitigate the effects of these highly unlikely, severe accidents addressed in the Level 1 PSA.

Level 1 PSA Analysis

A Level 1 PSA models the various plant responses, or accident sequences, to an event that challenges plant operation. The challenges to plant operation are termed *initiating events*. There are numerous accident sequences for a given initiating event, each with a probability of occurrence. The various accident sequences result from whether plant systems operate properly or fail and account for operator actions. Some accident sequences will result in a safe recovery and some will result in reactor core damage. The Level 1 PSA quantifies the frequency of severe accidents that compromise mitigative and preventive engineering safety features and, ultimately, cause damage to the nuclear reactor core. Also, the Level 1 PSA estimates the core damage frequency (CDF). Appendix A defines CDF and other standard term used in PSA applications.

The Level 1 PSA for the Pilgrim SAMA analysis considered a list of internal events, developed using industry data for accident initiating events, information from the BWR Owners Group, independent assessment team reviews, the IPE, and is plant-specific. An internal event is one that originates from an upset condition or failure inside the plant (e.g. loss-of-coolant accident due to system piping failure). External events, such as earthquakes, were accounted for by applying a multiplying factor to the calculated benefits before comparing to the costs, later in the SAMA analysis.

Reliability data for various structures, systems, and components, and human operator performance were used in logic models (event trees and logic trees) to determine the likelihood of core damage given an initiating event. The outcome of the Level 1 analysis by postulated initiating events is summarized in Table 1.[3]

Table 1. Pilgrim Nuclear Power Station Core Damage Frequency Due to Postulated Initiating Events

Initiating Event	Core Damage Frequency (per year)	Contribution to CDF (%)
Loss of DC Power Buses	3.1×10^{-6}	48
Loss of Off-Site Power	1.3×10^{-6}	20
Loss of Alternating Current Power Buses	8.8×10^{-7}	14
Loss of Salt Service Water	3.9×10^{-7}	6
Transients	3.6×10^{-7}	6
Loss of Coolant Accidents	1.8×10^{-7}	3
Station Blackout	1.5×10^{-7}	2
Anticipated Transient Without Scram	5.3×10^{-8}	1
Interfacing System Loss-of-Coolant Accident	3.6×10^{-8}	<1
Internal Flooding	1.3×10^{-8}	<1
Total CDF from Internal Events	6.4×10^{-6}	100

3 Table 1 is based on the Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Pilgrim Nuclear Power Station - Final Report, Appendix G, Table G-1 (NUREG-1437, Supplement 29) (NRC 2007) (NRC000002).

Level 2 PSA Analysis

The Level 2 PSA analyzes the progression of an accident by considering how the containment structures and systems respond to the postulated accident, which varies based on the initial status of the structure or system and its ability to withstand the harsh accident environment. As part of this analysis, the Level 2 PSA considers the key phenomena that affect accident progression beginning with core damage and concluding with containment release. Once the containment response is modeled, the analysis determines the amount and type of radioactivity released from the containment. Thus, the Level 2 PSA starts with the Level 1 core damage sequences, and quantifies the frequency and amount of radioactivity released into the environment from the nuclear power plant for each accident type.

The Pilgrim Level 2 analysis examined the dominant accident sequences and the resulting categories of plant damage, or plant damage states (PDS), defined in the Level 1 PSA. Two types of analyses are performed: (1) a deterministic analysis of the physical processes for a spectrum of severe accident progressions, and (2) a probabilistic analysis component in which the likelihood of the various outcomes are assessed. The deterministic analysis examines the response of the containment to the physical processes during a severe accident. Containment response is modeled by: (1) utilization of the MAAP code to simulate severe accidents that have been identified as dominant contributors to core damage in the Level 1 analysis, and (2) reference calculation of several hydrodynamic and heat transfer phenomena that occur during the progression of severe accidents. Examples of the phenomena modeled include debris coolability, pressure spikes due to ex-vessel steam explosions, direct containment heating, molten debris behavior, containment bypass, deflagration and detonation of hydrogen, thrust forces at reactor vessel failure, liner melt-through, and thermal attack of containment penetrations.

The Level 2 probabilistic analysis is based on a containment event tree (CET) model. The CET represents an accident progression given initial plant damage states and is a logic model with functional nodes that represent sequential phenomenological events and the status of containment protection systems. Core damage sequences from the Level 1 analysis were binned into 48 plant damage states based on binning criteria that reflect the state of the reactor, containment and cooling systems as the postulated accident progresses.[4] A specific PDS defines an important combination of system states that can result in distinctly different accident progression pathways and, therefore, different containment failure and source term characteristics. In effect, the PDS are interfaces for information from the Level 1 PSA to the Level 2 PSA.

The Level 2 accident progression is evaluated for each of the PDS using a single CET to determine the appropriate release bin or category for each Level 2 sequence. Each end state associated with a Level 2 sequence is assigned to a release category based on characteristics of (1) the timing of the radioactive release into the environment and (2) the magnitude of the radioactive release into the environment.[5]

(1) Timing governs the extent of radioactive decay of short-lived radioisotopes prior to an off-site release and, therefore, has a first-order influence on the early health effects. The Pilgrim ER characterized the timing relative when the release begins, measured from the time of accident initiation, into two timing categories: early (0-24 hours) and late (>24 hours).

(2) The following four distinct radionuclide release categories were used to characterize the magnitude of a release:

- High - A radionuclide release of sufficient magnitude to have the potential to cause early fatalities. This implies a total integrated release of >10% of the initial core inventory of cesium iodide (CsI). Source term results from previous, contemporary risk studies suggest that categorization of release magnitude based on cesium iodide (CsI) release fractions alone are appropriate for the magnitude binning purpose. The CsI release fraction indicates the fraction of in-vessel radionuclides escaping to the environment.

- Medium - A radionuclide release of sufficient magnitude to cause near term health effects. This implies a total integrated release of between 1 and 10% of the initial core inventory of CsI.

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4 See ENT000006 at Section E.1.2.2.5, Mapping of Level 1 Results into the Various Release Categories, and Table E.1-8.

5 See ENT000006 at Section E.1.2.2.4, Release Bin Assessments, Table E.1-6, and Table E.1-7.

- Low - A radionuclide release with the potential for latent health effects. This implies a total integrated release of between 0.001% and 1% of the initial core inventory of CsI.
- (4) Negligible - A radionuclide release that is less than or equal to the containment design base leakage. This implies total integrated release of <0.001% of the initial core inventory of CsI.

The timing (two categories) and magnitude (four categories) were grouped together to provide eight release categories used in the Pilgrim PSA, i.e., Early/High, Early/Medium, Early/Low, Early/Negligible, Late/High, Late/Medium, Late/Low and Late/Negligible. For example, a Late/Medium release category would signify a late timing (longer than 24 hours after the start of the postulated accident) for medium severity (integrated release of between 1% and 10% of the initial core inventory of CsI) radioactivity release. In addition to the release category characteristics of timing and magnitude, the CET Level 2 model determines the release category frequency attributed to each Level 1 PDS. This is information is used in the development of the collapsed accident progression bins, discussed below.

Collapsed Accident Progression Bins

A major feature of a Level 2 PSA phase of analysis is the estimation of the source term for every possible outcome of the CET. The CET end points represent the outcomes of possible in-containment accident progression sequences. These end points represent complete severe accident sequences from initiating event to release of radionuclides to the environment.
Thus, the Level 2 PSA analysis is the basis for characterizing the release, in terms of timing, magnitude and other relevant information, i.e., the source term, for the spectrum of possible radionuclide release scenarios. The source term groups were defined in terms of similar properties, and are termed collapsed accident progression bins (CAPBs), each with a frequency-weighted mean source term. The CAPBs were generated by sorting the accident progression bins for each of the forty-eight PDS on important attributes of the accident, such as:

- the occurrence of core damage and reactor vessel breach,

- primary system pressure at reactor pressure vessel breach,
- the location of containment failure, the timing of containment failure, and
- the occurrence of core-concrete interactions.

Several hundred source terms for internal initiating events are “collapsed” into source term groups of similar characteristics for the accident progression sequences.

From the Level 2 PSA, a total of 19 CAPBs resulted to adequately represent various categories of release timing and magnitude.[6] The release characteristics for each CAPB are determined by frequency-weighting the release characteristics for each PDS contributing to the CAPB. The source term release fractions for the PDS accident progression CET endpoints are estimated using a source term algorithm that separately accounts for in-vessel and ex-vessel fission product releases, and fission product removal mechanisms appropriate for the release pathways. Inputs to the source term algorithm are based on the results of Pilgrim plant-specific analyses of the dominant CET scenarios using the MAAP code, and fission product decontamination factors from the Peach Bottom BWR plant reported in the earlier NUREG-1150 study.[7]

Table 2 provides a description of the qualitative accident sequence characteristics for each CAPB source term. Table 3 provides the distribution of frequency, release timing, height, and energy content for each CAPB. Table 4 shows the fraction released from the core inventory by fission product group comprising the source term for each CAPB. There are a total of nine fission product groups considered for each CAPB. Each radionuclide group in the reactor contains radionuclides that have similar physical-chemical properties, and are shown in Table 5. For example, there are six noble gas (NG) radionuclides of xenon (Xe) and krypton (Kr) considered in the inventory. In total, sixty radionuclides are evaluated in the MACCS2 reactor core inventory, as represented in nine fission product groups.

[6] See ENT000006 at Section E.1.2.2.6 Collapsed Accident Progression Bins Source Terms, (2006).

[7] U.S. Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, (NRC, 1990).

Summary of Source Term Development for SAMA Analysis

In summary, the distribution of frequencies and the set of source terms for application in the SAMA analysis for the Pilgrim Plant are developed from the plant-specific PSA. Fault tree and event tree logic models, plant data, and mechanistic models of severe accident phenomena are used as part of the Level 1 and Level 2 Pilgrim PSA analysis and are the bases for defining the source terms and their respective frequency used in the Pilgrim SAMA analysis. The Level 1 PSA covers initiating event analysis and core damage sequence analysis, leading to a set of 48 plant damage states and associated frequencies. The Level 2 PSA uses Containment Event Tree and deterministic source term models to provide a set of 19 collapsed accident progression bins (CAPBs), each with a characteristic frequency, and unique timing and fission product magnitude characteristics. The 19 CAPBs are the same accident scenarios used in the MACCS2 analysis to determine the Pilgrim Plant off-site population dose risk (PDR) and off-site economic cost risk (OECR) that is described in the Meteorological Testimony.

Table 2. Collapsed Accident Progression Bins (CAPB) Descriptions (Based on ENT000006 at Table E.1-9)

CAPB	Description of Accident Sequence Characteristics (<u>See</u> ENT000006 at Section 1.2.2.6)
1	Core damage (CD) occurs, but timely recovery of RPV injection prevents vessel breach (No VB). Therefore, containment integrity is not challenged (No CF) and core-concrete interactions are precluded (No CCI). However, the potential exists for in-vessel release to the environment due to containment design leakage.
2	Core damage (CD) occurs followed by vessel breach (VB). Containment does not fail structurally and is not vented (No CF). Ex-vessel releases are recovered, precluding core-concrete interactions (No CCI). Although containment does not fail, vessel breach does occur, therefore the potential exists for in- and ex-vessel releases to the environment due to containment design leakage. RPV pressure is not important because, even though high pressure induced severe accident phenomena (such as direct containment heating [DCH]) occurs, containment does not fail.
3	Core damage (CD) occurs followed by vessel breach (VB). Containment does not fail structurally and is not vented (No CF). However, ex-vessel releases are not recovered in time, and therefore core-concrete interactions occur (CCI). RPV pressure is not important because, even though high pressure induced severe accident phenomena (such as direct containment heating [DCH]) occurs, containment does not fail, nor is the vent limit reached.
4	Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is greater than 200 psig at time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] are possible). There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
5	Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is less than 200 psig at time of vessel breach; precluding high pressure induced severe accident phenomena. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
6	Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is greater than 200 psig at time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] are possible). Following containment failure, core-concrete interactions occur (CCI).
7	Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is less than 200 psig at time of vessel breach; precluding high pressure induced severe accident phenomena. Following containment failure, core-concrete interactions occur (CCI).
8	Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is greater than 200 psig at time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] are possible). There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
9	Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is less than 200 psig at time of vessel breach; precluding high pressure induced severe accident phenomena. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
10	Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before

CAPB	Description of Accident Sequence Characteristics (<u>See</u> ENT000006 at Section 1.2.2.6)
	core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is greater than 200 psig at time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] are possible). Following containment failure, core-concrete interactions occur (CCI).
11	Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is less than 200 psig at time of vessel breach; precluding high pressure induced severe accident phenomena. Following containment failure, core-concrete interactions occur (CCI).
12	Core damage (CD) occurs followed by vessel breach (VB). Containment fails late due to loss of containment heat removal (Late CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is not important because high-pressure severe accident phenomena (such as DCH) did not fail containment. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
13	Core damage (CD) occurs followed by vessel breach (VB). Containment fails late (late CF) due to core-concrete interactions (CCI) after vessel breach. Containment failure occurs in the torus (WW), above the water level. RPV pressure is not important because high-pressure severe accident phenomena (such as DCH) did not fail containment.
14	Core damage (CD) occurs followed by vessel breach (VB). Containment fails late due to loss of containment heat removal (Late CF). Containment failure occurs in the drywell or below the torus water level (DW). RPV pressure is not important because high-pressure severe accident phenomena did not fail containment. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
15	Core damage (CD) occurs followed by vessel breach (VB). Containment fails late (late CF) due to core-concrete interactions (CCI) after vessel breach. Containment failure occurs in the drywell or below the torus water level (DW). RPV pressure is not important because high-pressure severe accident phenomena did not fail containment.
16	Small break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at high RPV pressure with a bypassed containment. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
17	Large break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at low RPV pressure with a bypassed containment. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
18	Small break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at high RPV pressure with a bypassed containment. Following vessel breach, core-concrete interaction occurs (CCI).
19	Large break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at low RPV pressure with a bypassed containment. Following vessel breach, core-concrete interaction occurs (CCI).

Table 3. Release Characteristics of the PNPS Collapsed Accident Progression Bin Releases for the SAMA Analysis (Based on ENT000010 at Table E.1-15)

Release Mode	Frequency (per year)	Time of Release After Shutdown, (seconds)	Release Duration, (seconds)	Release Height, (m)	Energy Release Rate* in Plume, (W)
CAPB-1	9.51E-08	2.20E+04	9.00E+03	30.	2.61E+05
CAPB-2	1.27E-08	2.20E+04	9.00E+03	30.	2.50E+05
CAPB-3	2.39E-09	2.20E+04	9.00E+03	30.	2.50E+05
CAPB-4	3.29E-09	1.83E+04	3.56E+03	30.	1.10E+07
CAPB-5	2.73E-09	2.53E+04	7.93E+03	30.	8.34E+06
CAPB-6	7.95E-09	2.56E+04	8.11E+03	30.	8.23E+06
CAPB-7	7.93E-09	2.61E+04	8.46E+03	30.	8.03E+06
CAPB-8	2.06E-08	2.00E+04	4.59E+03	30.	1.04E+07
CAPB-9	9.25E-09	2.44E+04	8.87E+03	30.	4.18E+06
CAPB-10	8.53E-08	2.60E+04	8.40E+03	30.	8.06E+06
CAPB-11	4.35E-08	2.60E+04	8.40E+03	30.	8.06E+06
CAPB-12	1.70E-06	4.64E+04	9.00E+03	30.	7.59E+06
CAPB-13	2.30E-09	2.71E+04	9.00E+03	30.	1.80E+06
CAPB-14	2.26E-06	4.46E+04	9.00E+03	30.	7.08E+06
CAPB-15	2.12E-06	4.62E+04	9.00E+03	30.	7.60E+06
CAPB-16	1.18E-09	2.12E+04	9.00E+03	30.	2.50E+05
CAPB-17	6.91E-09	2.14E+04	9.00E+03	30.	2.50E+05
CAPB-18	4.61E-10	2.12E+04	9.00E+03	30.	2.50E+05
CAPB-19	2.43E-08	2.18E+04	9.00E+03	30.	2.50E+05

* The energy release rate is often described as the sensible heat rate, and is energy released per unit time (in units of Watts) due to the plume being at a higher temperature from radioactive decay than the relatively cooler temperature of the ambient atmosphere. This thermal energy will normally make the release more buoyant, effectively increasing the height of release.

Table 4. Radionuclide Release Fractions of the PNPS Collapsed Accident Progression Bins (Based on ENT000006 Table E.1-11)

	Release Fractions								
	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	1.99E-07	1.85E-07	1.85E-07	0.00E+00	1.24E-09	8.00E-09	5.01E-11	8.43E-11	1.70E-08
2	9.97E-05	4.81E-05	4.66E-05	1.76E-07	3.97E-07	4.00E-06	1.65E-08	5.15E-08	4.87E-06
3	9.97E-05	5.37E-05	4.97E-05	1.76E-06	5.80E-07	4.00E-06	2.37E-08	1.57E-07	4.95E-06
4	1.00E+00	4.90E-02	2.62E-02	4.18E-05	2.46E-05	3.66E-04	8.97E-07	3.04E-06	1.92E-04
5	9.85E-01	7.86E-02	3.68E-02	4.28E-05	4.10E-05	3.66E-04	1.56E-06	6.79E-06	3.44E-04
6	1.00E+00	4.02E-02	2.32E-02	1.48E-03	3.19E-04	3.66E-04	6.50E-06	7.17E-05	3.23E-04
7	9.76E-01	6.11E-02	2.94E-02	1.26E-03	2.30E-04	3.66E-04	9.10E-06	1.06E-04	4.52E-04
8	1.00E+00	2.98E-01	2.72E-01	3.07E-05	9.89E-04	2.23E-02	4.49E-05	6.57E-05	1.15E-02
9	5.97E-01	7.61E-02	7.07E-02	1.41E-05	9.72E-04	1.09E-02	3.69E-05	7.63E-05	1.02E-02
10	1.00E+00	2.80E-01	2.49E-01	1.11E-02	3.07E-03	1.81E-02	7.95E-05	5.81E-04	1.03E-02
11	9.79E-01	1.73E-01	1.41E-01	9.97E-03	3.13E-03	1.78E-02	1.22E-04	9.39E-04	1.72E-02
12	2.01E-01	5.84E-05	4.37E-05	1.25E-07	2.36E-07	1.72E-06	8.04E-09	2.56E-08	2.99E-06
13	9.97E-01	7.99E-03	5.99E-03	1.76E-04	3.63E-05	3.66E-04	2.15E-06	1.41E-05	4.52E-04
14	7.75E-01	2.88E-02	2.67E-02	2.47E-05	2.05E-04	2.13E-03	8.49E-06	2.27E-05	2.61E-03
15	9.97E-01	2.76E-01	2.68E-01	1.27E-03	2.27E-03	2.25E-02	9.33E-05	3.00E-04	2.74E-02
16	1.00E+00	6.71E-02	3.26E-02	4.06E-04	9.11E-05	2.21E-02	1.45E-06	1.65E-05	4.27E-05
17	9.72E-01	3.62E-01	3.37E-01	1.34E-03	2.37E-03	2.20E-02	9.90E-05	1.62E-04	8.57E-03
18	1.00E+00	9.76E-02	6.25E-02	2.09E-02	4.67E-03	2.27E-02	7.45E-05	8.50E-04	2.12E-03
19	9.72E-01	4.03E-01	3.77E-01	6.87E-02	9.58E-03	2.26E-02	3.00E-04	2.33E-03	1.20E-02

Table 5. Radionuclide Group Composition Used in MACCS2, See Code Manual for MACCS2: Volume 1, User's Guide, NUREG/CR-6613 (SAND97-0594) (1998).		
Radionuclide Group	Number of nuclides	Nuclides
NG	6	Kr-85, Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135
I	5	I-131, I-132, I-133, I-134, I-135
Cs	4	Rb-86, Cs-134, Cs-136, Cs-137
Te	8	Sb-127, Sb-129, Te-127, Te-127m, Te-129, Te-129m, Te-131m, Te-132
Sr	4	Sr-89, Sr-90, Sr-91, Sr-92
Ru	8	Co-58, Co-60, Mo-99, Tc-99m, Ru-103, Ru-105, Ru-106, Rh-105
La	15	Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97, Nb-95, La-140, La-141, La-142, Pr-143, Nd-147, Am-241, Cm-242, Cm-244
Ce	8	Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241
Ba	2	Ba-139, Ba-140
Total	60	

Q6: **Regarding the radioactive contamination to be computed from the dispersion and deposition caused by the meteorological patterns at issue, describe in sufficient detail for scientific understanding: (b) The degree of conservatism imbedded in the methodology for determining the source term to be used for each computation of radioactivity dispersion and deposition, its sources, and the rationale for each source of conservatism.**

A6. (KRO) The probabilistic safety assessment frequency and source term analyses have several sources of conservatism in frequency and inventory, as described below. These are discussed by topic:

Frequency for Loss of Offsite Power (LOOP) Initiating Event: The 2003 PSA model used one single frequency for loss of offsite power from the 345kV ring bus. Loss of the 23kV feed from the Manomet Station to the shutdown transformer was modeled as a split fraction (i.e. conditional probability) of this frequency.[8] It was conservatively assumed that 50% of the losses of offsite power resulted in a complete loss of all incoming AC power, despite the independence of the 23kV line. Thus, for this initiating event, a higher frequency is assumed than if independence of the 23kV line is credited.

Inventory: An initial estimate to the radionuclide inventory for the SAMA analysis was originally based on expected power level alone, as provided for by

[8] ENT000007 at Response to RAI 1a.

industry guidance.[9] This default inventory was revised per a NRC Request for Additional Information in consideration of an increased level of long-lived radionuclides such as Sr-90, Cs-134, and Cs-137. The inventory was recalculated above that expected based on power level alone from an ORIGEN calculation assuming 4.65% enrichment and average burn-up according to the expected fuel management practice over the twenty-year extended (license renewal) operation period as provided in ENT000007. The inventory obtained with this approach differed from the power-scaled reference inventory for long-lived radionuclides by approximately a 25% increase. The revised baseline benefits in the SAMA analysis include the impact of the 25% increase in the inventory values for Sr-90, Cs-134, and Cs-137 for each analysis case. The inventory change in the base case led to a 7.4 % increase to the mean off-site population dose risk (PDR) and a 14.6% increase in the mean off-site economic cost risk (OECR).

Q7: Regarding the radioactive contamination to be computed from the dispersion and deposition caused by the meteorological patterns at issue, describe in sufficient detail for scientific understanding: (c) The extent to which the conservatism imbedded in the methodology for determining the source term cause the resultant deposition to be conservative, being as quantitative as is practicable (note that qualitative discussions are acceptable where quantitative analysis is not practicable).

A7. (KRO) The conservatism of initiating event frequency for the LOOP implies that a greater frequency of occurrence of the LOOP event is assumed in the Level 1 PSA, and increases both the PDR and OECR. The source term and consequence conservatisms of inventory increase impacts the resulting isopleths (footprint of the plume as it travels downwind and contaminates land through the dry deposition phenomenon), and result in larger land contamination than would be the case if the smaller inventory of radionuclides was used. Larger land contamination (resultant deposition) in turn will result in larger population dose and off-site economic risks.

Table 6 summarizes each source of conservatism and assesses the magnitude of the conservatism.

[9] See Nuclear Energy Institute, NEI-05-01 Rev. A, Nuclear Energy Institute Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document.

Table 6. Conservatism in Source Term and Deposition Analysis

Item	Conservatism in Analysis	Change to PDR	Change to OECR	Basis
1.	Initiating event frequency for Loss of Offsite Power	Increase	Increase	ENT000007 at RAI 1a.
2.	25% Increase in long-lived radionuclides in the core inventory	7.4% increase	14.6% increase	ENT000007 at RAI 4c.

Appendix A. Definitions Based on NUREG-1150 and other PSA References:

Collapsed Accident Progression Bin: A group of postulated accidents that has similar characteristics with respect to the timing of containment building failure and other factors that determining the amount of radioactive material released. Sometimes referred to as containment failure modes in older PRAs.

Core Damage Frequency: The frequency of combinations of initiating events, hardware failures, and human errors leading to core uncovery with reflooding of the core not imminently expected.

External Initiating Events: Events occurring away from the reactor site that result in initiating events in the plant. In keeping with PRA tradition, some events occurring within the plant during normal power plant operation, e.g., fires and floods initiated within the plant, are included in this category.

Internal Initiating Events: Initiating events (e.g., transient events requiring reactor shutdown, pipe breaks) occurring during the normal power generation of a nuclear power plant. In keeping with PRA standard practice, loss of offsite power is considered an internal initiating event.

Plant Damage State: A group of accident sequences that has similar characteristics with respect to accident progression and containment engineered safety feature operability.

Release Fraction: The fraction defining the portion of the radionuclide inventory by radionuclide group in the reactor at the start of an accident that is released to the environment.

Source Term: The fractions defining the portion of the radionuclide inventory in the reactor at the start of an accident that is released to the environment. Also included in the source term are the initial elevation, heat or energy content of the plume, and timing of the release (time after accident initiation or shutdown, and duration of release).