June 27, 2011

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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)	)	

DECLARATION OF JOSEPH R. LYNCH, LORI ANN POTTS, AND DR. KEVIN R. O'KULA IN SUPPORT OF ENTERGY ANSWER OPPOSING COMMONWEALTH CLAIMS OF NEW AND SIGNIFICANT INFORMATION BASED ON FUKUSHIMA

Mr. Joseph R. Lynch ("JRL"), Ms. Lori Ann Potts ("LAP") and Dr. Kevin R. O'Kula ("KRO") state as follows under penalties of perjury:

I. INTRODUCTION

A. Entergy Declarants

1. Joseph R. Lynch

1. (JRL) I am the Manager of Licensing for the Pilgrim Nuclear Power Station ("Pilgrim"). My professional and educational experience is summarized in the Curriculum Vitae attached as Exhibit 1 to this Declaration.

2. (JRL) I have over 29 years of nuclear power experience and background in engineering, licensing/regulatory affairs, environmental compliance, complex problem solving, stakeholder communications, project management, cost control, budgeting and employee management. I obtained my Bachelor of Science in Mechanical Engineering (BSME) from the Worcester Polytechnic Institute in Worcester, Massachusetts in 1982, specializing in thermodynamics, and fluid mechanics. I have undertaken graduate-level studies in Business Management,

Communications, and Regulatory Compliance. I have also taken numerous internal and external management courses while previously working with the Yankee Atomic Electric Company and the Vermont Yankee Nuclear Power Station.

3. (JRL) In my current position as Pilgrim Licensing Manager, I am responsible for managing the Pilgrim Licensing Group, which supports the operation and regulatory compliance of the station in accordance with NRC, State and Federal regulations, permits and statutes. I am responsible for the development of all necessary letters, licensing correspondence and regulatory approvals from NRC, local, state and federal agencies required in support of plant operations. I am familiar with all Pilgrim operational systems, including the operation, and the current plant licensing basis which include the Extensive Damage Mitigation Guidelines ("EDMGs") and mitigative strategies for catastrophic events which are a series of requirements imposed by the NRC since the attacks on the World Trade Center on September 11, 2001. I am familiar with the Pilgrim License Renewal Application, and the aging management measures that Pilgrim will employ during its period of extended operation.

2. Lori Ann Potts

4. (LAP) I am a senior consulting engineer to Entergy Nuclear in the areas of Severe Accident Mitigation Alternatives Analysis and Fire Probabilistic Risk Assessment. My professional and educational experience is summarized in the Curriculum Vitae attached as Exhibit 2 to this Declaration.

5. (LAP) I have over 30 years of experience as a technical professional in the nuclear industry in the areas of safety analysis, probabilistic safety assessment, deterministic and probabilistic accident and consequence analysis, materials aging management, reactor engineering,

and systems engineering. I obtained a Bachelor of Science degree in Nuclear Engineering from The Pennsylvania State University in 1981.

6. (LAP) I have previous Probabilistic Safety Assessment ("PSA") and severe accident analysis experience in analyzing reactor, emergency system, and containment phenomena under accident conditions. I was the primary author of the industry SAMA guideline (NEI 05-01) which was endorsed by the NRC. I have participated directly in the SAMA analyses for eight nuclear plants, including the SAMA analysis for the Pilgrim plant, and have peer reviewed the SAMA analyses for three additional nuclear plants.

3. Dr. Kevin R. O'Kula

7. (KRO) I am an Advisory Engineer with URS Safety Management Solutions ("URS") LLC. My professional and educational experience is summarized in the Curriculum Vitae attached as Exhibit 2 to this Declaration.

8. (KRO) I have over 28 years of experience as a technical professional and manager in the areas of safety analysis methods and guidance development, computer code evaluation and verification, probabilistic safety assessment, deterministic and probabilistic accident and consequence analysis applications for reactor and non-reactor nuclear facilities, source term evaluation, risk management, reactor materials dosimetry, and shielding. I obtained a Bachelor of Science degree in Applied and Engineering Physics from Cornell University in 1975, a Master of Science degree in Nuclear Engineering from the University of Wisconsin in 1977, and a Ph.D. in Nuclear Engineering from the University of Wisconsin in 1984.

9. (KRO) I have over 22 years of experience in using the MELCOR Accident Consequence Code System ("MACCS") and the MACCS2 Computer Codes, and have taught MACCS2 training courses for the Department of Energy ("DOE") at Lawrence Livermore National Laboratory, Los Alamos National Laboratory, Idaho National Laboratory, and at DOE Safety Analysis Workshops. I was the lead author of a DOE guidance document on the use of MACCS2.[1] Also, I am a member of the State-of-the-Art Reactor Consequence Analysis ("SOARCA") Project Peer Review Committee that provides recommendations on applying MACCS2 in the context of accident phenomena and subsequent off-site consequences in the context of severe reactor accidents, to Sandia National Laboratories ("SNL") and the NRC.

10. In the context of the above work, I am intimately familiar with the use of probabilistic risk assessments ("PRA")[2] in the nuclear industry. In addition, as part of my responsibilities as Group Manager with Westinghouse Savannah River Company, I was responsible for the K Reactor (Savannah River Site) Probabilistic Safety Assessment group and managed the SRS K Reactor PSA program until its completion in the early 1990s. This program was a full scope PSA activity covering the phases discussed in the present contention, including core damage frequency (Level 1), plant response (Level 2), and offsite consequence analyses (Level 3). My related direct PRA experience also includes support for a new plant vendor in the preparation of the probabilistic risk analysis required for a design certification application and ongoing Quantitative Risk Analysis ("QRA") lead support with initiating events for the Waste Treatment Plant piping design to account for operational hydrogen issues.

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[1] *MACCS2 Computer Code Application Guidance for Documented Safety Analysis*, DOE-EH-4.2.1.3-Final MACCS2 Code Guidance, Final Report, U.S. Department of Energy, Washington, DC (June 2004).

[2] PRA is used interchangeably with probabilistic safety assessment ("PSA"),

11. Also, I have other direct experience working on hydrogen control issues. I was the Technical Program Chair for American Nuclear Society Topical Meeting on the Safety and Technology of Nuclear Hydrogen Production, Control and Management (Boston, Massachusetts, 2007),and am currently the Chair of the Department of Energy Facility Contractors Group (EF-COG) Safety Interest Group on Hydrogen Safety.

B. Commonwealth of Massachusetts Late-Filed Contention

12. (JRL, LAP, KRO) We have reviewed and are familiar with the Commonwealth of Massachusetts' late-filed contention[3] and the supporting report of Gordon Thompson[4] concerning alleged new and significant information arising from the March 11, 2011 accident at Japan's Fukushima Daiichi reactor complex, which was filed on June 2, 2011 in the NRC licensing proceeding for the Pilgrim license renewal.

13. (JRL, LAP, KRO) The Commonwealth's late-filed Contention states:

> [T]he environmental impact analysis and the SAMA analysis in Supp. 29 to the Generic Environmental Impact Statement (GEIS) for License Renewal (1996) are inadequate to satisfy NEPA because they fail to address new and significant information revealed by the Fukushima accident that is likely to affect the outcome of those analyses. The new and significant information shows that both core-melt accidents and spent fuel pool accidents are significantly more likely than estimated or assumed in Supp. 29 of the License Renewal GEIS or the SAMA analysis for the Pilgrim NPP. As a result, the environmental impacts of re-licensing the Pilgrim NPP have been underestimated. In addition, the SAMA analysis is deficient because it ignores or rejects mitigative measures that may now prove to be cost-effective in light of this new understanding of the risks of re-licensing Pilgrim.

[3] Commonwealth of Massachusetts' Contention Regarding New and Significant Information Revealed by the Fukushima Radiological Accident (June 2, 2011) (""Contention" or "Commonwealth Contention").

[4] New and Significant Information from the Fukushima Daiichi Accident in the Context of Future Operation of the Pilgrim Nuclear Power Plant, Gordon R. Thompson (June 1, 2011) ("Thompson Report" or "Report").

Contention at 5-6. In particular, the Commonwealth asserts six bases for its Contention based on the Thompson Report: (1) the estimate of core damage frequency relied on in Supp. 29 and the related SAMA analysis is unrealistically low by an order of magnitude as evidenced by the Fukushima accident and other reactor accidents that have occurred; (2) the NRC's assumptions about operators' capability to mitigate an accident at the Pilgrim NPP are unrealistically optimistic and that capability can be severely degraded in the accident environment; (3) the NRC's excessive secrecy regarding accident mitigation measures and the phenomena associated with spent fuel pool fires degrades the licensee's capability to mitigate an accident at Pilgrim; (4) it appears likely that hydrogen explosions similar to those experienced at Fukushima could occur at Pilgrim and therefore should be considered in the SAMA analysis; (5) there appears to be a substantial conditional probability of a spent fuel pool fire during a reactor accident at Pilgrim; and (6) it appears likely that filtered venting of the Pilgrim reactor containment could substantially reduce the atmospheric release of radioactive material from an accident at Pilgrim. Contention at 6-7.

14. (JRL, LAP, KRO) Our Declaration addresses the claims raised by the Commonwealth concerning the adequacy of the Pilgrim SAMA analysis in light of Fukushima. In summary:

- The claim that the CDF should be an order of magnitude larger based on the historical experience of five reactor core melts is fundamentally flawed in two key respects. Dr. Thompson's "direct experience" method is not an appropriate method for determining CDF at a specific plant and it is not statistically valid. Dr. Thompson's direct experience CDF violates fundamental PRA precepts underlying the Commission's application of PRAs for more than two decades.
- Dr. Thompson provides no new information from the Fukushima events that would negate the Commission's prior determination that the risk of spent fuel pool fire is "very low." Dr. Thompson provides no support for his claim of a substantial condi-

tional probability of 50% of a spent-fuel-pool fire occurring in the event of a severe reactor accident, and conveniently ignores the fact that a spent fuel pool fire did not occur in conjunction with the core melting events at the three Fukushima reactors. The mitigative measures purportedly challenged by Dr. Thompson were identified by the Commission as additional capability to "further enhance" spent fuel cooling, thereby further reducing the risk of spent fuel pool zirconium fires. In any event, Dr. Thompson focuses on only one of the many mitigative measures for spent fuel pools put in place at U.S. plants subsequent to September 11, and never addresses the other measures available.

- Dr. Thompson's claims regarding the alleged secrecy of mitigative measures does not concern either NEPA or SAMA analysis, and is therefore not pertinent here. Furthermore, Pilgrim operators and personnel responsible for implementing mitigating actions are fully trained on their implementation.

- Dr. Thompson's claims with respect to hydrogen explosions are insignificant and immaterial. Although preliminary, the Government of Japan has prepared a comprehensive report[5] that summarizes known facts concerning the accident, which provides ample evidence that the hydrogen explosions at Units 1 and 3 occurred in the reactor building structures, not the primary containments.[6] Further, both design features and procedures are in place at Pilgrim to control hydrogen generation and to prevent hydrogen explosions within the primary containment. Furthermore, the potential for hydrogen explosions within either the primary or secondary containments has been fully considered in the Pilgrim SAMA analysis and Dr. Thompson nowhere disputes the adequacy of the Pilgrim SAMA analysis consideration of hydrogen explosions. Moreover, the hydrogen-related concerns would not make any difference in the

[5] Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety – The Accident at TEPCO's Fukushima Nuclear Power Stations, Nuclear Emergency Response Headquarters, Government of Japan (June 2011) ("Fukushima Report"), which is attached to this Declaration as Exhibit 4.

[6] With respect to what is known about the operation of the DTV for Fukushima Units 1, 2, and 3, the Report prepared by the Government of Japan is consistent with a separate report prepared by the International Atomic Energy Agency entitled Mission Report: The Great East Japan Earthquake Expert Mission, IAEA International Fact Finding Expert Mission of the Fukushima Daiichi NPP Accident Following the Great East Japan Earthquake Tsunami (May 24 – June 2, 2011) ("IAEA Report"). The relevant portions of the IAEA Report summarizing the sequence of events leading to the accident at Fukushima Daiichi are attached to this Declaration as Exhibit 5.

SAMA analysis because the analysis performed for Pilgrim bounds the events at Fukushima.

- Pilgrim's SAMA analysis concluded that adding a filtered DTV would not be cost beneficial, and Dr. Thompson makes no showing, in light of the Fukushima accident, that filtered venting under accident conditions may have been inadequately considered in the Pilgrim SAMA analysis. Indeed, he nowhere references or disputes the Pilgrim SAMA analysis' consideration of the DTV.

Thus, none of Dr. Thompson's claims have merit, and none of them identify a new or significant environmental issue revealed by Fukushima that would impact the Pilgrim plant.

II. ALLEGED NEW AND SIGNIFICANT INFORMATION CONCERNING CORE DAMAGE FREQUENCY USED IN THE PILGRIM SAMA ANALYSIS

15. (KRO) Dr. Thompson claims in his Issue #1 that the core melting at the Fukushima Daiichi Units 1, 2, and 3, together with the TMI and Chernobyl core melt events, show that the estimate of core damage frequency ("CDF") used in the Pilgrim SAMA analysis is unrealistically low and should be increased by an order of magnitude. To support this claim, Dr. Thompson introduces what he calls the "direct experience" method of estimating core damage probability (which has no precedent) in which he uses five severe core damage events in a worldwide fleet of 440 plants with a cumulative total of 14,484 reactor-years ("Rx-yrs") of experience to calculate a CDF of 5 events/14,484 Rx-yrs, or 3.4E-04/Rx-yr. Thompson Report at17.

16. (KRO) Dr. Thompson's claim that the CDF should be an order of magnitude larger based on the historical experience of five reactor core melts is fundamentally flawed in two key respects. **First**, Dr. Thompson's "direct experience" method for calculating CDF has no basis in logic and has never been used to calculate a CDF for PRA applications. It would directly contradict the fundamental precepts of probabilistic risk analysis ("PRA") that are based on plant-specific characteristics. PRA has been applied in the nuclear safety community in a structured,

formal manner since the publication of WASH-1400 (Reactor Safety Study")[7] in 1975, but even then differentiated between the Pressurized Water Reactor ("PWR") and Boiling Water Reactor ("BWR") plant, as well as accounting for site differences. Fifteen years later, NUREG-1150 (1990)[8] refined the application of PRA to commercial nuclear power plants with an examination of individual plant risk. This study not only differentiated in the type of reactor (PWR or BWR), but also accounted for primary and secondary (reactor building) containments and site differences. This approach was further refined by the NRC beginning in the 1990s with the Individual Plant Examination ("IPE") and Individual Plant Examination for External Events ("IPEEE") programs, to examine risks on a plant-specific basis in order to provide a principled basis for sound risk-informed decision-making. Dr. Thompson's direct experience CDF method ignores 36 years of increased attention to reactor system or type, primary and secondary, and site details. Dr. Thompson's use of a single CDF for all plants is illogical and ignores the fact that there are great differences in plant design, operating procedures, and in threats from potentially catastrophic natural phenomena that can greatly impact a plant's core damage frequency. These considerations are unjustifiably ignored in the approach suggested by Dr. Thompson.

17. (KRO) **Second**, Dr. Thompson's direct experience CDF method is inherently invalid. His basis of five core melt conditions among widely varying nuclear power plants from three accident initiating conditions is too small to calculate a valid CDF and moreover, none of the five core melt events are relevant to the Pilgrim plant. Three of the five events were caused by a combination of natural phenomena not relevant to Pilgrim and its site, the seismic event and

[7] NUREG-75?104 (WASH-1400), Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (Oct. 1975).

[8] NUREG-1150, Severe Accident Risks: An Assessment for Five Nuclear Power Plants, Final Summary Report (Dec. 1990) ("NUREG-1150").

tsunami in Japan, the fourth was at Chernobyl, a reactor of an entirely different design, and the fifth was Three Mile Island, a PWR, unlike Pilgrim, which is a BWR.

A. Dr. Thompson's Direct Experience CDF Method is not a Scientifically Acceptable Approach for Determining Plant Core Damage Frequency

18. (KRO) Dr. Thompson's direct experience CDF method of estimating the probability of core damage is not a scientifically acceptable method for determining the CDF for a specific plant. Under well-established NRC precedent, practice and regulatory guidance, PRAs are based on specific reactor and containment design features, operating procedures, and site considerations for evaluating overall vulnerabilities, establishing prioritization of potential improvements, and to aid in making risk-informed decisions. Many nuclear plants have significant differences in design and siting, but Dr. Thompson's direct experience CDF method would nevertheless establish one CDF for all plants with no distinction for design and site differences. His direct experience CDF method does not account for plant-unique site conditions, plant design, support system dependencies, plant maintenance procedures, plant operating procedures, operator training, and the dependencies between these many different factors that can influence the CDF estimate for a specific plant. Dr. Thompson totally ignores, for example, the fact that there are great differences in plant designs and their susceptibility to catastrophic natural phenomena that directly bear on and significantly affect their CDFs.

19. (KRO) The tsunami in Japan, which was the initiating event for the Fukushima core melt event, illustrates the illogical nature of Dr. Thompson's direct experience CDF method. Dr. Thompson attributes three of the five core melt events to Fukushima Daiichi Units 1, 2, and 3, which originated from this common cause. Thus, Dr. Thompson's resulting direct experience CDF is greatly influenced by three Fukushima core melts, even though most U.S. coastal plants, including Pilgrim, have small tsunami risk, and certainly not the potential for the severity of a

tsunami such as that which struck Fukushima. Similarly, the core melt event at Chernobyl has no relevance at all to Pilgrim or any other U.S. plant because it was a totally different reactor design (a graphite moderated reactor). Calculating core damage frequency based on events inapplicable to a plant, as Dr. Thompson has done, is inherently illogical and totally contradicts the purposes and precepts of PRAs for use in risk-informed decision-making.

20. (KRO) Dr. Thompson acknowledges that his "simple estimate of CDF might be criticized because the three core-damage events at Fukushima had a common cause." However, he implies that they are three different core damage events because "there are some design differences between the three affected plants at Fukushima, and it appears that there were differences in the accident sequences at these plants." Thompson Report at 17 n. 34. However, Dr. Thompson contradicts his own thesis. On one hand, Dr. Thompson asserts that we should ignore the plant-specific core damage frequency calculated from the PRA logic model using combination of possible initiating events, component failures, and operator action failures in favor of a simple direct experience CDF which ignores design differences and differences in accident sequences between plants. Then, on the other hand, he asserts that three events with a common cause in his simple direct experience CDF calculation must be considered different events because there are design differences, and differences in the accident sequences at the three affected plants.

21. (KRO) Dr. Thompson goes on to indicate that the three events with a common cause should be considered multiple events because "multiple core-damage events with a common cause could occur in the future, potentially involving plants at single-unit sites." Id. In effect, Dr. Thompson is speculating that one natural disaster which challenges normal plant operation at multiple single-unit sites could occur and that it should not be counted as one event, but

should be counted by the number of plants that don't successfully mitigate the event to prevent core damage.

22. (KRO, LAP) The illogical nature of Dr. Thompson's thesis is readily apparent from a simple analogy. Assume that a massive meteor hits North America tomorrow and that the impact from the meteor and the ensuing damage challenge normal operation at 15 individual nuclear units at different sites (including Pilgrim). In this example, five of the plants do not have adequate mitigation features in place and ultimately suffer core damage. But, Pilgrim and nine other plants successfully mitigate the event and avoid core damage. According to Dr. Thompson's direct experience thesis, this would mean that the "direct experience CDF" (3.4E-4 per year following Fukushima) must now be adjusted to account for five more core damage events, to 10 events/14,500 reactor-years (6.9E-4 per year) for <u>all</u> nuclear plants, <u>even for Pilgrim and the other nine plants that successfully prevented core damage after the same massive meteor strike</u>. Thus, under Dr. Thompson's direct experience CDF method, no matter how many severe accident mitigation features that Pilgrim or any other plant adds, the risk of core damage cannot be improved. This conclusion defies logic.

23. (KRO) Paragraphs 20-22 above summarize the illogical results of what Dr. Thompson is saying with respect to Fukushima, or Chernobyl for that matter, i.e., because three plants at Fukushima suffered core melt damage as a result of the occurrence of a large, unusual tsunami, Pilgrim and all other plants should arbitrarily increase their CDF even though they may neither be subject to such a natural phenomenon event, or if subject, be able to mitigate the event so as to suffer no core damage.

24. (KRO) Thus, the logical implications of Dr. Thompson's direct experience method are nonsensical. We are aware of no U.S. or international probabilistic risk assessments

where Dr. Thompson's direct experience method or a direct observation from a different facility is used in place of detailed design-, plant type-, and site-specific PRA analysis that identifies initiating events and their likelihood potentially leading to core damage to establish the CDF, subsequent reactor containment release, and environmental release conditions. To blindly adopt an order-of-magnitude increase in the CDF without a basis would undermine the process of systematically looking for vulnerabilities that could lead to accident sequences that may impact the plant. This process is based on the procedural steps followed in a PSA including: (1) Level 1 analysis of initiating events and determination of the core damage states and their associated frequencies; (2) Level 2 analysis of plant damage conditions and determination of the radiological release conditions; and (3) Level 3 analysis of consequences and risk indices.

25. (KRO) Furthermore, Dr. Thompson's direct experience method of establishing a generic CDF with no distinction for plant-design and plant-site differences is directly contrary to established NRC practice, in particular as reflected in NUREG-1150, NRC guidance in NRC Generic Letter 88-20[9] and Supplement 4 to Generic Letter 88-20,[10] NRC Regulatory Guide 1.200[11] and Regulatory Guide 1.174,[12] and NEI 05-01 Rev. A,[13] that have guided countless PRAs, and SAMA analyses performed for NRC licensing. Pertinent points from these studies and guidance include:

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[9] Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f) (Generic Letter No. 88-20) (Nov. 23, 1988).

[10] Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f) (June 28, 1991).

[11] Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Rev. 2, Mar. 2009) ("Reg. Guide 1.200").

[12] Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to The Licensing Basis (Rev. 2, May 2011).

[13] NEI-05-01, Severe Accident Mitigation Alternatives (SAMA) Guidance Document (Rev. A, Nov. 2005) ("NEI-05-01").

- **NUREG-1150**: NUREG-1150 was a multi-laboratory, multiyear assessment of the risks from severe accidents in five commercial nuclear power plants in the United States and was the seminal work on probabilistic risk assessment that generally preceded its use in U.S. nuclear regulation. The study, published in final in1990, pointed out in clear terms, that "NUREG-1150 is not an estimate of the risks of all commercial nuclear power plants in the United States or abroad. One of the clear perspectives from this study of severe accident risks and other such studies is that characteristics of design and operation specific to individual plants can have a substantial impact on the estimated risks." NUREG-1150, Section 1.2 at 1-3.

- **Generic Letter 88-20**: In Generic Letter 88-20, the Nuclear Regulatory Commission recognized that:

  > based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. Therefore, each existing plant should perform a systematic examination to identify any plant-specific vulnerabilities to severe accidents . . .

  Generic Letter 88-20 at 1 (emphasis added). The Generic Letter states that the general purpose of this examination, defined as an Individual Plant Examination ("IPE"), is for each utility:

  > (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant, (3) to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and (4) if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

  Id. (emphasis added). Thus, clearly, plant-specific and site-specific considerations are paramount to the use of PRAs to implement the IPE requirements.

- **Generic Letter 88-20, Supplement 4:** This supplement to Generic Letter 88-20 requested all licensees to perform an Individual Plant Examination of External Events (IPEEE) to identify "plant-specific severe accident vulnerabilities" caused by external events and report the results to NRC. Generic Letter 88-20, Supplement 4 at 2 (em-

phasis added). Seismic and fire risk evaluations are the two major aspects of the IPEEE, but a number of other external hazards are examined, as appropriate, such as high winds and tornadoes, external flooding, ice, hazardous chemical, transportation, and nearby facility incidents.

- **Regulatory Guide 1.200**: Revision 2 to Regulatory Guide 1.200 describes one acceptable approach for determining whether the technical adequacy of a PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. Approaches for building, assessing, and reviewing a high-quality PRA are described. In particular, the CDF is not defined a priori and applied, but is "defined as the sum of the frequencies of those accidents that result in uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in off-site public health effects." Reg. Guide 1.200, Section 1.1 at 7 (emphasis added).

- **Regulatory Guide 1.174**: Regulatory Guide 1.174 sets forth the requirements for using PRA to support changes in a plant's licensing basis. As set forth in NRC guidance, the "scope, level of detail, and technical acceptability" of these risk-informed analyses are to "be based on the as-built and as-operated and maintained plant," and "reflect operating experience at the plant." Regulatory Guide 1.174 at 7 (emphasis added).

- **NEI 05-01, Rev. A**: NEI 05-01, Rev. A, endorsed by the NRC,[14] provides a template for completing the severe accident mitigation alternatives ("SAMA") analysis in support of license renewal. NEI 05-01 states for the Level 1 and 2 analysis to "Use the plant-specific PSA model", and to apply Individual Plant Examination (IPE) and IPE – External Events (IPEEE) analyses and recommendations. NEI 05-01, Section 2, at 2.

[14] Final License Renewal Interim Staff Guidance LR-ISG-2006-03, Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analyses (Aug. 2, 2007) (ADAMS Accession No. ML071640133). See also 72 Fed. Reg. 45,466 (Aug. 14, 2007).

26. (KRO) Dr. Thompson incorrectly implies in his Report that the analytical techniques used in the PRA method are not based on direct experience and, therefore, the direct experience method provides a "reality check for PRA estimates." Thompson Report at 15, 16. However, PRA methods are not some mystical method of predicting the future as implied by Dr. Thompson. They are based on direct industry operating experience, as well as experience relevant to the plant and expert knowledge of the expected plant response to specific events.

27. (KRO) As indicated in Reg. Guide 1.200, the risk-informed results used to support a regulatory application must "be derived from a baseline PRA model that represents the as-built, as-operated plant." Reg. Guide 1.200 at 6. Also, "the process is performed such that the plant information identified and used in the PRA in reflecting the as-designed, as-built, as-operated plant, is as realistic as possible in assessing the risk." Id. at 29. Realistic, plant-specific information must be used in the PRA model to ensure that vulnerabilities at a plant are not masked by inappropriate assumptions. The PRA fault tree combines initiating event probabilities, component failure probabilities, component out of service probabilities, and human action failure probabilities to determine the CDF. The estimation process for each of these probabilities includes a mechanism for addressing uncertainties and has the ability to combine different sources of data in a coherent manner, including the actual operating history and experience of the plant, as well as applicable generic experience. Thus, a PRA model is based on actual experience and is therefore, realistic in its own right for the specific plant for which the PRA was established.

28. (KRO, LAP) For example, Pilgrim developed site-specific estimates of accident probabilities for use in the Pilgrim PSA model based on realistic estimates of the probability that an accident will occur. Those initiating events that challenge normal plant operation and that require successful mitigation to prevent core damage were identified using a structured, system-

atic process for identifying initiating events that accounts for plant-specific features. Generic analyses of similar plants were reviewed to ensure that the list of challenges included in the model accounts for industry operating experience. A systematic evaluation of each plant system was undertaken to assess the possibility of an initiating event occurring due to a failure of the system. Each initiating event frequency was calculated accounting for relevant generic and plant-specific data. Thus, a realistic analysis for Pilgrim was done based on both plant-specific data and relevant industry generic data.

29. (KRO) Dr. Thompson also incorrectly suggests that his direct experience method "is supported by a technical literature describing the limitations of PRA techniques." Thompson Report at 17, referencing a 1989 article by Hirsch and three other authors, "IAEA Safety Targets and Probabilistic Risk Assessment" (Aug. 1989). This article prepared for Greenpeace is, however, dated and moreover was reviewed and its conclusions rejected by the International Nuclear Safety Advisory Group ("INSAG"), an advisory group to the Director General of the International Atomic Energy Agency. In concluding its review of the 1989 Hirsch et. al article, INSAG stated as follows:

> In the field of nuclear safety, probabilistic risk assessment has graduated to a state of widespread use, with confident acceptance of its benefits and its limitations. The points raised by the Greenpeace report were taken up, debated and resolved to the satisfaction of safety experts quite a few years ago.[15]

Thus, the Hirsch et al, 1989 report provides no support for Dr. Thompson's direct experience thesis and Dr. Thompson does not refer to any other technical literature in his Report.

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[15] INSAG TECHNICAL NOTE No. 3, A Review of the Report 'IAEA Safety Targets and Probabilistic Risk Assessment' Prepared for Greenpeace International (1991) at 8.

30. (KRO) Dr. Thompson also suggests that a 1990 Peach Bottom seismic CDF of about 1.0E-04 per year based on the Livermore seismic predictions reported in NUREG-1150 is "roughly consistent with" and supports his direct experience CDF estimate of 3.4E-04 per year. Thompson Report at 17. Dr. Thompson, however, provides no reason why this dated reference supports his thesis, and it is also deficient for the reasons already stated – that PRAs are plant-specific analyses. There is no reason to believe that an old value for seismic CDF at Peach Bottom has anything to do with the total CDF at Pilgrim. Furthermore, the 1990 value for Peach Bottom does not reflect the many plant improvements that have been made since Peach Bottom performed the IPEEE (in which plants identified and mitigated seismic vulnerabilities) and hence does not reflect Peach Bottom's current CDF.[16]

31. (KRO, LAP) Similarly, Dr. Thompson mistakenly relies upon Pilgrim IPE and IPEEE CDF and early release frequency values instead of information from License Renewal Application Environmental Report ("ER") to calculate the conditional probability of an early release given a core damage event at Pilgrim.[17] As a Result, in Section VI of his Report (pages 16-17) Dr. Thompson incorrectly claims that "the licensee's current position regarding SAMA analysis is that the …conditional probability of Early release is 0.32 (32 percent)" and that, for the time being, "it may be appropriate to use the licensee's estimate of the conditional probability of an Early release at Pilgrim, namely 0.32." Thompson Report at 16, 17. Since the performance

[16] NUREG-1437, Supplement 10, at 5-11. Dr. Thompson's Report includes and refers to other Surry and Peach Bottom CDF and release frequencies from the NUREG-1150 study (page 15 and Figures VI-1 through VI-3). But, Dr. Thompson does not use the rest of the Surry and Peach Bottom information to draw any conclusions.

[17] As Dr. Thompson states at page 15-16 of his Report, the conditional probability of early release of 32% come from Tables 6-1 and 9-1 of his 2006 Report, Risk and Risk-Reducing Options Associated with Pool Storage of Spent Nuclear Fuel at the Pilgrim and Vermont Yankee Nuclear Power Plants (May 25, 2006). In his 2006 Report, Dr. Thompson provides both the ER and IPE and IPEEE risk estimates for Pilgrim in Table 6-1 to the Report. However, as he states at page 19 of the 2006 Report that Report used the IPE and the IPEEE estimates instead of the ER estimates. Accordingly, the CDF and early release frequencies providing in Dr. Thompson's 2006 Report in Table 9-are based on the IPE and the IPEEE risk information and not the ER risk information.

of the IPE and IPEEE, the Pilgrim risk values have decreased due to several reasons as described in the ER, Section 1.4.2. These included a combination of improved plant performance, replacement of switchyard breakers, more realistic success criteria based on updated thermal hydraulic calculations, and more sophisticated data analysis. As a result, both the CDF and the early release frequency for the SAMA analysis are measurably lower. The CDF for internal events for the SAMA analysis is 6.4E-6 per year (ER Section E.1.2) and the early release frequency is 3.29% (ER Figure E.1-1). The SAMA analysis uses the same early release frequency for fire and seismic initiating events because the containment protection system status following an earthquake or fire would not be significantly different from the status defined by the internal events model which already accounts for loss of offsite power.

32. (KRO, LAP) Thus, based on the up-to-date PRA and SAMA analysis for Pilgrim, the conditional probability for an early release assuming a core damage event for the Pilgrim plant is 0.0329 early release/core damage event, or 3.29%.

B. Dr. Thompson's Direct Experience Method is Inherently Invalid

33. (KRO) In addition to violating fundamental PRA precepts as discussed above, Dr. Thompson's direct experience method is not appropriate. Indeed, Dr. Thompson acknowledges that his "five events provide a data set that is comparatively sparse and therefore does not provide a statistical basis for a high-confidence estimate of CDF." Thompson Report at 16. Yet, he uses this estimate to invalidate the plant specific CDF based on extensive site-specific data. His "sparse" data set simply cannot be used to draw such a conclusion.

34. (KRO) The invalidity of Dr. Thompson's direct experience CDF is starkly apparent given that none of his five data points used to calculate his direct experience CDF of 3.4E-04 per year are applicable to Pilgrim. The 1979 TMI core damage event occurred in a PWR with

very different design and engineering safety systems from those of BWRs such as Pilgrim. The Chernobyl event has no relevance because of the plant's entirely different design and the fact that the accident was initiated by an improper test evolution, i.e., not due to an internal or external initiating event condition arising during normal commercial power operations. And finally, the Fukushima incident was initiated by a combination of natural phenomena, a magnitude 9.0 earthquake followed by a 14-meter (over 45 feet) tsunami, sequential events that have a very low likelihood of occurring, particularly at the same severity, at the Pilgrim site. Particularly with an event such as Fukushima that was caused by sequential, but interrelated natural phenomena (earthquake and ensuing tsunami), later leading to loss of offsite power to plant systems (station blackout), it is important not to extrapolate from one location to another when evaluating the outcome. The initiating events are very region- and location-specific, and the tectonic and geological fault line locations are far different between northeast coastal Japan and the coast of Massachusetts. Thus, the population of events that Dr. Thompson uses to calculate his direct experience generic CDF are irrelevant to Pilgrim and thus inappropriate to use for a statistical determination of Pilgrim's CDF.

35. (JRL) Specifically, the following facts about the Pilgrim site reflect that a tsunami or other site flooding natural events such as those giving rise to the core melt events at Fukushima are not applicable to Pilgrim:

- *A tsunami at Pilgrim Station is not a probable event. This is based on the known geological features in the area. There are no major subduction zones in the Atlantic Ocean; except for where it borders the Caribbean Sea. As a result, there has been a low frequency of tsunamis compared to the experience in the Pacific Ocean.* ***(Evaluation of Tsunami Sources with the Potential to Impact U.S. Atlantic and Gulf Coast. USGS, August 2008).***

- *The most famous Atlantic tsunami is the 1755 Lisbon tsunami that was generated by an earthquake on a fault offshore of Portugal. Geologists today estimate that the Lisbon earthquake approached magnitude 9 on the Richter scale, with an epicenter in the Atlantic Ocean about 200 km (120 mi) west-southwest of Cape St. Vincent. Although massive destruction of Lisbon occurred, the effects on North America were unremarkable.* ***(Evaluation of Tsunami Sources with the Potential to Impact U.S. Atlantic and Gulf Coast. USGS, August 2008).***
- *The most noteworthy North American tsunami was the result of the 1929 Grand Banks earthquake near Newfoundland, Canada. The maximum tsunami run-up from this event was 6-21 feet, concentrated on the coast of Newfoundland. Wave heights near Plymouth were not noted to be affected. The U.S. tide gauge diagrams following the 1929 earthquake for Eastport (Maine), Portland (Maine), Boston (Massachusetts), and Key West (Florida) indicated no effect from the tsunami.* ***(Evaluation of Tsunami Sources with the Potential to Impact U.S. Atlantic and Gulf Coast. USGS, August 2008).***
- *Pilgrim Station is designed to withstand a flood level of 18.3 feet above mean low water (mlw) that could result from a combination of storm surge and astronomical high tide. The unlikely combination of these two events is estimated to occur once every four thousand years.* ***Pilgrim Station FSAR Section 2.4.4.***
- *Laboratory model testing conducted for the Pilgrim site has shown that the plant would survive a storm surge of 19.5 feet above mlw. These tests included simulating open ocean wave heights up to 31 feet. During the model testing, no wetting of the reactor building occurred.* ***Pilgrim Station FSAR Section 2.4.4.***
- *Pilgrim station is also designed to withstand a Category 3 hurricane (maximum wind speed of 131 mph). The storm surge caused by a hurricane of that magnitude is surpassed by the design nor'easter discussed above. Pilgrim's location on the north side of Cape Cod protects it from the full storm surge of any storm travelling up from the south.* ***Pilgrim Station FSAR Section 2.4.4.***

- *The highest astronomical tide ever recorded in the area was 10.3 feet above mlw on Feb. 24 1723. The highest storm surge (in conjunction with the high astronomical tide) recorded at Pilgrim station since original construction was 14.5 feet above mlw which occurred on February 7th 1978 (known as the "Blizzard of '78"). Pilgrim survived that event with no storm related damage to any safety systems including the emergency diesel generators ("EDGs").* ***Pilgrim Station FSAR Section 2.4.4.***
- *Pilgrim Station has three diesel generators ("DGs") at the site. Any one of the three can supply sufficient power to run the systems necessary to bring the reactor from an operating state to a cold condition.* ***Pilgrim Station FSAR Section 8.1.***
- *Two of the diesel generators, the emergency diesel generators ("EDGs"), were part of the original design of the station. They are each rated at 2600KW and can power the AC powered emergency core cooling pumps and the primary containment cooling systems. These EDGs are located on the north side of the site on the 23' elevation (above sea level) between Cape Cod Bay and the reactor building. As described in question # 4, the elevation of the EDG building was selected based on worst case historic weather related conditions that have occurred at the Pilgrim site. Both EDGs are tested monthly.* ***Pilgrim Station Procedure 8.9.1.***
- *In the late 1980's, Pilgrim added a third diesel generator, which is located on the south side of the site. It was specifically designed in response to NRC rulemaking on station blackout issues to enhance Pilgrim's capability to endure a loss of offsite power and a postulated failure of the two EDGs described above. At the location where it is sited, the reactor and turbine buildings would protect the station blackout diesel generator ("SBO DG") from any storm surge. The SBO DG is a 2000KW generator that can be started from the main control room and can power selected emergency core and containment cooling systems. The SBO DG is tested quarterly.* ***Pilgrim Station Procedure 8.9.16.1.***
- *In the event of a station blackout, operators would follow pre-established procedures to start and load the SBO DG. In parallel, operating procedures specify the minimization of battery load through the shedding of non-essential DC powered components.* ***Pilgrim Station Procedure 5.3.31.***

- *All emergency power sources at Pilgrim are air cooled and do not rely on an external cooling water source.* ***Pilgrim Station FSAR Section 8.1.***

36. In short, none of the five core melt events are applicable to the Pilgrim plant, and therefore the application of Dr. Thompson's direct experience method for determining the Pilgrim CDF is inappropriate on that basis alone.

III. ALLEGED NEW AND SIGNIFICANT INFORMATION CONCERNING SPENT FUEL POOL FIRES

37. (KRO) Dr. Thompson makes several claims of alleged new and significant information regarding spent fuel pool fires based on the events at Fukushima. He claims in his Issue #2[18] that the NRC's assumptions about operators' capability to mitigate a spent fuel pool accident are unrealistically optimistic and that there is a substantial conditional probability (50%) of a spent fuel pool fire occurring during a reactor accident that leads to an early release. Thompson Report at 18-20. In his Issue #5[19] Dr. Thompson claims that his asserted substantial conditional probability of a spent fuel pool fire occurring during a reactor accident leading to an early release is supported by the "direct experience from Fukushima." Thompson Report at 27.

38. (KRO, JRL) Dr. Thompson provides no new information from the Fukushima events that would negate the Commission's determination in its 2008 Rulemaking Denial based on a long series of studies spanning three decades, i.e., that the risk of spent fuel pool fire is "very low."[20] Mitigative measures challenged by Dr. Thompson in his Report were identified by the Commission as additional capability to "further enhance" spent fuel cooling, thereby further

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[18] "Operators' Capability to Mitigate an Accident, and its Effect on the Conditional Probability of a Spent-Fuel-Pool Fire During a Reactor Accident," Thompson Report at 18.

[19] "Probability of a Spent-Fuel-Pool Fire and Radioactive Release, Accounting for Fukushima Direct Experience," Thompson Report at 26.

[20] The Attorney General of Commonwealth of Massachusetts; the Attorney General of California; Denial of Petitions for Rulemaking, 73 Fed. Reg. 46,204, 46,207 (Aug. 8, 2008) ("Rulemaking Denial").

reducing the risk of spent fuel pool zirconium fires. 73 Fed. Reg. at 46,212. Moreover, Dr. Thompson focuses on only one of the many mitigative measures for spent fuel pools put in place after September 11 at U.S. plants, that of the use of a fire truck to spray water on the spent fuel pool, which was also eventually used at Fukushima. Pilgrim and other U.S. plants have other mitigative measures in place as well. One of the required mitigative measures for all U.S. plants is having an "independently-powered, portable, SFP coolant makeup and spray capability system that enhances spray and rapid coolant makeup to mitigate a wide range of possible scenarios that could reduce SFP water levels."[21] To satisfy this requirement, Pilgrim has a dedicated portable diesel engine-driven pump, physically on site, to provide cooling for the spent fuel pool (which apparently was not available at Fukushima).

39. (KRO) Furthermore, Dr. Thompson provides no support for his claim of a substantial conditional probability of 50% that a spent fuel pool fire will occur in the event of a severe reactor accident. The Commission rejected this identical claim in its Rulemaking Denial (73 Fed. Reg. at 46,209-10) and Dr. Thompson has made no attempt to address the many deficiencies identified by the Commission in his approach. Moreover, since none of the Fukushima spent fuel pools ultimately failed, the "direct experience from Fukushima" provides no basis for the large 50% conditional probability of a spent fuel pool fire that Thompson, ipso facto, assumes in the event of a severe reactor accident causing an early release. To the contrary, Fukushima shows that spent fuel pools are very rugged and are capable of withstanding extreme conditions, such as severe earthquakes and hydrogen explosions.

[21] Rulemaking Denial, 73 Fed. Reg. at 46,209.

A. The 2008 Commission Rulemaking Denial

40. (KRO) In its 2008 Rulemaking Denial, the Commission denied the Commonwealth of Massachusetts' 2006 Petition for Rulemaking. The Commonwealth's 2006 Petition for Rulemaking requested that the Commission to revoke the provisions in 10 C.F.R. Part 51 that exclude consideration of spent fuel pool issues in individual licensing proceedings based on claims of purported "new and significant information" developed since the 1996 issuance of NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants (1996) ("GEIS"), which the Commonwealth asserted showed that the NRC's determination in the GEIS of "insignificant" environmental impacts for high-density spent fuel storage is incorrect.[22] The new information asserted by the Commonwealth included a 2006 Report by Dr. Thompson.[23]

41. (KRO) In the 1996 GEIS, the Commission expressly considered severe spent fuel pool accidents and concluded that "even under the worst probable cause of a loss of spent-fuel pool coolant (a severe seismic-generated accident causing a catastrophic failure of the pool), the likelihood of a fuel-cladding fire is highly remote." GEIS at 6-72 to 6-75 (citation omitted). As support for this conclusion, the GEIS referred to the Commission's 1990 Review and Revision of the Waste Confidence Decision,[24] which in turn, was based on a series of technical studies dating back to 1979 and before. Based on those studies, the Commission concluded in the 1990 Waste Confidence Decision that:

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[22] Massachusetts Attorney General's Petition for Rulemaking to Amend 10 C.F.R. Part 51 (Aug. 25, 2006).

[23] The Commonwealth provided as an attachment to its 2006 Petition a copy of its hearing request in the Pilgrim license renewal proceeding which included the 2006 report by Gordon R. Thompson, Risks and Risk-Reducing Options Associated with Pool Storage of Spent Nuclear Fuel at the Pilgrim and Vermont Yankee Nuclear Power Plants (Cambridge, Massachusetts: Institute for Resource and Security Studies, 25 May 2006) ("2006 Thompson Report" or "2006 Report").

[24] Review and Final Revision of Waste Confidence Decision, 55 Fed. Reg. 38,474 (Sept. 18, 1990).

> [E]ven if the timing of a spent fuel pool failure were conducive to fire, a fire could occur only with a relatively sudden and substantial loss of coolant – a loss great enough to uncover all or most of the fuel, damaging enough to admit enough air to keep a large fire going, and sudden enough to deny operators the time to restore the pool to a safe condition. Such a severe loss of cooling water is likely to result only from an earthquake well beyond the conservatively estimated earthquake for which reactors are designed. Earthquakes of that magnitude are extremely rare.

55 Fed. Reg. at 38,481. The probability of a seismically induced major spent fuel pool failure, which the Commission considered "extremely rare," was calculated in the technical studies relied upon for the 1990 Waste Confidence Decision as 2E–6 per year, or two chances in a million per reactor year.[25]

42. (KRO, LAP) The occurrence of a highly unlikely event does not make that event more likely to occur in the future. (A person's chances of winning the lottery are not increased each time someone wins the lottery.) Thus, if a seismically induced major spent fuel pool failure had occurred in a seismically active region such as Japan, the probability of a major spent fuel pool failure in the U.S. would be unchanged.

43. (KRO) In its 2008 Rulemaking Denial, the Commission evaluated the purportedly new and significant information proffered by the Commonwealth and found that the information was neither new nor significant. To the extent that any information proffered by the Commonwealth was not considered in the GEIS, the Commission found the information to be insignificant because it "would not lead to 'an impact finding different from that codified in 10 CFR Part

[25] These studies calculated other probabilities of spent fuel pool fire risks due to other accident scenarios – such as structural failure of the pool due to high energy tornado or other missiles, aircraft crashes, and heavy load drops, inadvertent drainage of the pool, and boil-down of the pool due to loss of spent fuel cooling or make-up water – which were "at least an order of magnitude smaller." 55 Fed. Reg. at 38,481. The technical study primarily relied upon in the 1990 Waste Confidence Decision that calculated these probabilities was NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue, Beyond Design Basis Accidents in Spent Fuel Pools (April 1989) ("NUREG-1353").

51,' or as set forth in [the GEIS]." 73 Fed. Reg. at 46, 208. The Rulemaking Denial reviewed and evaluated each piece of purportedly new and significant information presented by the Commonwealth, including the 2006 Thompson Report, and found that none of them presented new and significant information. The Commission specifically rejected the different spent fuel pool fire phenomenological claims presented by the Commonwealth and Dr. Thompson, as well as Dr. Thompson's claim of a 50% conditional probability of a spent fuel pool fire occurring in the event of a severe reactor accident leading to an early release. 73 Fed. Reg. at 46,208-12.

44. (KRO) In its Rulemaking Denial, the Commission concluded that the risk of a zirconium fire initiating in a spent fuel pool is "very low." The Commission based its conclusion on a series of "[s]tudies conducted over the last three decades [that] have consistently shown that the probability of an accident causing a zirconium fire in an SFP to be lower than that for severe reactor accidents." 73 Fed. Reg. at 46, 207. These included:

- NUREG-1353, which was the primary basis for the 1990 Waste Confidence Decision underlying the 1996 GEIS.
- NUREG-1738,[26] which was a conservative bounding study of spent fuel pool fires in decommissioned plants. For example, the study conservatively assumed that the spent fuel would burn if the "water level reached 3 feet from the top of the spent fuel."[27] Even with these conservative assumptions, NUREG-1738 found a "very low

[26] NUREG-1738, Technical Study of Spent Fuel Pool Accident Risk and Decommissioning Nuclear Power Plants (Jan. 2001) ("NUREG-1738"). Although for decommissioned plants, the Commission considers NUREG-1738 to be equally applicable to operating plants, as reflected in the Rulemaking Denial, discussed above, as well as the Commission's Waste Confidence Decision Update, 75 Fed. Reg. 81,037, 81,069-70, 81,073 (Dec. 23, 2010). In this respect, NUREG-1738 conducted analyses for plants that had only recently been shut down (starting at 30 or 60 days after final shutdown depending on the analyses) and moreover, assumed that, because the plant was permanently shutting down, the full core would be unloaded into the spent fuel pool. NUREG-1738 at 2-1, 3-28, A1A-3 – A1A-4, A4-2. Because of the assumption that the full core had just been off-loaded to the spent fuel pool, the analysis in NUREG-1738 would generally be conservative compared to Pilgrim where typically only one-third of the core is off-loaded at each refueling outage.

[27] NUREG-1738 at 3-1; see also, e.g., id. at 2-1.

likelihood of a zirconium fire"[28] which was "conservatively estimated to be in the range of 5.8E–7 per year to 2.4E–6 per year."[29] The Commission concluded from NUREG-1738 that, "[e]ven with its numerous conservatisms," the risk of an SFP fire is "low and well within the Commission's Safety Goals" and is "substantially lower than reactor risk."[30] NUREG-1738 is referenced in the Draft Revision 1 of the GEIS as the primary basis for the conclusion that spent fuel pool accidents are a Category 1 issue.[31]

- Sandia tests conducted subsequent to the events of September 11. Unlike NUREG-1738, these studies evaluated the potential for spent fuel pool fires under more realistic assumptions that accounted for relevant heat flow and fluid flow mechanisms. The Sandia studies assuming these more realistic conditions showed that spent fuel pool fires would not occur under many of the circumstances conservatively assumed to cause a fire in NUREG-1738, thereby "reducing the likelihood" of a spent fuel pool zirconium fire to further below that calculated in NUREG-1738.[32]

45. (KRO, JRL) In addition to these studies, which consistently showed that the risk of a spent fuel pool fire is lower than that for severe reactor accidents, the Commission found that mitigative measures put in place by all plant licensees subsequent to September 11, 2001 to cope with losses of large areas of a plant due to fire or explosions provided enhanced capability to cool the spent fuel.[33] These mitigative measures included both an internal strategy to implement a diverse SFP makeup system and an SFP spray to remove decay heat and an external strategy "using an independently-powered, portable, SFP coolant makeup and spray capability system

[28] NUREG-1738 at ix, xi, 5-1, 5-3.

[29] 73 Fed. Reg. at 46,212.

[30] 73 Fed. Reg. at 46,207, 46,209.

[31] Draft Report for Comment, NUREG-1437, Volume 2, Rev. 1, Generic Environmental Impact Statement for License Renewal of Nuclear Plants (July 2009) Appendix E, Environmental Impact of Postulated Accidents, E.3.7 Impact from Accidents at Spent Fuel Pools, at E-32 to E-37.

[32] 73 Fed. Reg. at 46,207-208, 46,212.

[33] 73 Fed. Reg. at 46,208.

that enhances spray and rapid coolant makeup to mitigate a wide range of possible scenarios that could reduce SFP water levels."[34] As the Commission stated, "[t]hese mitigative measures further reduce the risk from SFP zirconium fires, and make it even more unlikely that additional SFP safety enhancements could substantially reduce risk or be cost- beneficial" in a SAMA analysis.[35]

B. The Thompson Report Provides No New or Significant Information Based on Fukushima to Negate the Commission's Rulemaking Denial Determinations

1. Dr. Thompson's claims regarding operator capability to mitigate a spent fuel pool accident do not negate the Commission's Determinations

46. (KRO) Dr. Thompson provides no new information in his Report to show that the studies relied upon by the Commission in its Rulemaking Denial are in error based on information from Fukushima. His Report neither mentions these extensive studies nor makes any attempt to say whether these studies are affected by Fukushima in any way.

47. (KRO) Nor does Dr. Thompson's Report reference or address the Commission's 2008 Rulemaking Denial, which rejected the claims made in his 2006 Report. Indeed, Dr. Thompson asserts that, because no evidence from Fukushima shows that his 2006 Report is wrong, his 2006 Report "describes the general state of knowledge about pool fires." Thompson Report at 27. In making this assertion, Dr. Thompson completely ignores the fact that the Commission 2008 Rulemaking Denial rejected the claims made in his 2006 Report that the Commonwealth relied upon in its 2006 Petition for Rulemaking as not describing the current state of knowledge about spent fuel pool fires. 73 Fed. Reg. at 46,208-10.

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[34] Id.

[35] Id. at 46,212.

48. (KRO) Rather, Dr. Thompson focuses his Report on the mitigative measures put in place by licensees subsequent to September 11, which the Commission relied upon in its Rulemaking Denial as providing enhanced capability to cool spent fuel pools. This focus on mitigative measures ignores the fact that the Commission had already determined, based solely on the technical studies, that the potential for a zirconium fire was "extremely rare," as concluded by the Commission in the 1990 Waste Confidence Decision, or of "very low likelihood" as concluded in NUREG-1738, even with its conservative assumptions. From this "very low likelihood" of a zirconium pool fire, the Commission concluded that the risk from spent fuel pool fires is "substantially lower than reactor risk." As explained by the Commission, because "the SFP risk level is less than that for a reactor accident, a SAMA that addresses SFP accidents would not be expected to have a significant impact on total risk for the site." Therefore, as relied upon by the Commission in its Rulemaking Denial, the mitigative measures only served to "further reduce" an already "low level of risk from fuel stored in SFPs" that would not "have a significant impact on the total risk for a site."[36]

49. (KRO, JRL) Moreover, as discussed below, Dr. Thompson's claim that Fukushima shows that the mitigative measures employed at U.S. plants are inadequate to mitigate spent fuel accidents rests on a series of faulty premises that invalidate his claim.

50. (KRO, JRL) **First,** the discussion in the Thompson Report essentially equates the mitigative measures used at Fukushima with the extensive damage mitigation guidelines (EDMGs") in place at plants in the U.S. This claim of equivalence is incorrect. Fukushima did not have in place prior to the earthquake and tsunami the range of mitigative measures that U.S. plants have in place. The official report prepared by the Government of Japan on what is known

[36] 73 Fed. Reg. at 46,212.

about Fukushima as of May 31, 2011[37] reflects that the mitigative measures employed at Fukushima were developed on the spot during the course of the accident. According to the Government of Japan's Fukushima Report, alternative measures to mitigate a spent fuel pool accident "such as alternative water injection into a spent fuel pool, etc. were not considered" at Fukushima. Fukushima Report at 30. Unlike in the U.S. there "are no requirements for the capability to perform alternative water injection" to the spent fuel pool, and as such there were no dedicated equipment, no procedures, and no personnel trained to implement such alternative measures at Fukushima prior to the earthquake and tsunami events. Id. at IV-134 to IV-135.

51. (JRL) In contrast, in the U.S., all nuclear utilities have been required by Commission Order[38] following September 11, 2001 to establish a range of mitigative strategies to respond to events that could lead to a serious accident, including the simultaneous loss of offsite and onsite power, and the loss of the ultimate heat sink. These measures include making provision for the physical and personnel resources and the necessary training to implement these different mitigative strategies depending on the plant's status and circumstances surrounding a potential accident. In developing the different mitigative measures, utilities are required to address and to include the methodology, logic, and the prioritization of multiple challenges from, security, fire, and natural events, while mitigating challenges to the reactor, primary containment, and the spent fuel pool in order to safeguard the public and plant personnel. To that end, and subject to past and ongoing NRC inspection, the equipment, procedures and trained personnel to implement this range of mitigative measures were in place at U.S. utilities before Fukushima. As ref-

[37] The Fukushima Report states that it is a "preliminary accident report" that "represents a summary of the evaluation of the accident and the lessons learned to date based on the facts gleaned about the situation so far" concerning the March 2011 accident at Fukushima Daiichi. Fukushima Report. at 2. The intent of the Fukushima Report is "to provide as accurately as possible an exact description of the facts of the situation," while "providing a clear distinction between known and unknown matters" as of May 31, 2011. Id. at 3.

[38] Mitigating Strategies Requirements from Order EA-02-026, Section B.5.b.

erenced in the Rulemaking Denial and documented in NEI 06-12, Rev. 2 (Dec. 2006) at 2-19,[39] the range of mitigative measures implemented at U.S. plants includes providing an independent portable power supply, external to the plant, to pump makeup water from different external sources to the spent fuel pool and the hoses and connections necessary to supply makeup water to the pool. These measures, now in place at Pilgrim and other plants, are required by each plant's license and by Commission regulation in 10 C.F.R. § 50.54(hh), and are subject to NRC inspection and enforcement.

52. (JRL) Pilgrim has developed and implemented the required mitigative measures. It has in place the capability to initiate the different spent fuel mitigative strategies outlined in NEI 06-12 within two hours of an event. Pilgrim's spent fuel mitigative measures include a portable diesel-driven pump capable of drawing water from on-site sources, municipal sources, or from the ocean. Should station blackout render power to operate the spent fuel pool make-up unavailable, the portable diesel-driven pump has the capability to pump water to the spent fuel pool, where it may be used in a fill or spray mode. Multiple pathways to the spent fuel pool have been identified and procedures have been developed to implement this mitigative strategy. Moreover, Pilgrim operators (licensed and unlicensed) and designated operations staff have been trained, and continue to be trained, in an ongoing program on the implementation of these mitigative procedures, and the use and capability of designated equipment.

53. (JRL) Thus, Dr. Thompson is wrong to compare and equate the mitigative measures employed at Fukushima to those in place at U.S. plants. In contrast to the delayed, ad hoc mitigative measures employed at Fukushima, U.S. plants have established pre-existing physical and

[39] NEI 06-12 was recently made publicly available by the Commission and is available at ADAMS Accession ML070090060.

personnel capabilities to implement mitigative measures shortly following a station blackout or other serious accident.

54. (JRL) Furthermore, Dr. Thompson's suggestion that the mitigation measures have been "separately devised"[40] for Pilgrim is incorrect. The EDMGs in NEI 06-12 provide the generic requirements that are to be met by each plant's mitigative measures, e.g., "Establish a flexible means of SFP makeup of at least 500 gpm using a portable, power-independent pumping capability," for which NEI 06-12 provides a list of the required attributes. NEI 06-12 at 6-8. Each plant is therefore required to provide a flexible means for spent fuel makeup of 500 gpm using a portable, power-independent pumping capability. Only the method for providing this capability, e.g., on-site fire pumper truck or an onsite, external portable diesel-driven pump, is left for the individual utilities to determine in their implementation of the generic NEI 06-12 requirements.

55. (JRL) **Second**, Dr. Thompson's discussion of mitigative measures focuses on only one of the several mitigative measures in place at U.S. plants implemented since September 11, 2001. He focuses on the use of a fire truck to spray water on the spent fuel pool, as was done at Fukushima, and assumes, with no basis, that this sole mitigative measure is equivalent to those in place at U.S. plants. Thompson Report at 19. He totally ignores the fact that U.S. utilities are required to provide and have in place a range of mitigative measures to draw upon as needed. For example, a wholly independent portable power supply, such as the portable diesel-driven pump, is physically in place at Pilgrim while agreements with the Town of Plymouth to provide additional or alternative pumping capability also exist. No one measure is relied upon to the ex-

[40] Declaration of Dr. Gordon R. Thompson in Support of Commonwealth of Massachusetts' Contention and Related Petitions and Motions (June 1, 2011) ¶ 17.

clusion of others, but all are required to be available in order to respond to an event as plant circumstances may dictate. In this respect, Pilgrim does have, as the Thompson Report suggests at 19-20, an arrangement with local fire companies for a ladder truck, but this mitigative measure is a secondary means of providing spent fuel pool flooding and spray. The primary means relied upon is the dedicated diesel-driven portable pump located on site.

56. (KRO, JRL) **Third**, Fukushima illustrates the extended, available time during which mitigative measures may be implemented to prevent a spent fuel pool zirconium fire. The Fukushima Report and IAEA Report show that the initiation of SFP cooling and spraying of water did not begin on Unit 3 until March 17, 2011, approximately six days after the accident, on Unit 2 and Unit 4 until March 20, 2011, nine days after the accident, and on Unit 1 until March 31, 2011. Fukushima Report at IV-82, IV-65, IV-92, IV-50; IAEA Report at 36. As Dr. Thompson acknowledges, "no full scale" zirconium pool fire occurred at Fukushima even though mitigative measures did not begin until after six days, in some cases, well after six days. Thompson Report at 26. Unit 4, with the highest heat load with its recently off-loaded full core, and SFP being 97% full, was the subject of concern. Although the Unit 4 pool went without cooling for nine days, the Fukushima Report states that "the results of analyzing nuclides from the spent fuel pool and visual inspections have revealed that Unit 4's spent fuel pool remains nearly undamaged." Fukushima Report at IV-91. The IAEA similarly stated the following in its Report.

> To determine the status of the SFPs, images were taken of the Unit 3 and 4 SFPs remotely. The images verified the presence of a water level and showed that the fuel appeared to be intact. An extensive amount of debris generated by the explosion in Unit 3 had fallen into the Unit 3 SFP, so that the structural integrity of the racks could not be confirmed. There was some debris in the Unit 4 SFP, likely due to the explosion at Unit 4, but the status of the racks and the fuel is reported to be near normal on the basis of present information.

IAEA Report at 36 (emphasis added). Thus, there is no evidence that a spent fuel pool zirconium fire occurred at Fukushima.[41]

57. (KRO, JRL) Thus, even with the prolonged delay in implementing mitigative measures at Fukushima, they have been adequate to prevent a zirconium spent fuel pool fire. The events at Fukushima thus corroborate the Commission's determination in the Rulemaking Denial that there would be "a significant amount of time" from the initiating event to the possible onset of a zirconium fire, "thereby providing a substantial opportunity for both operator and system event mitigation." 73 Fed. Reg. at 46,208.

2. Dr. Thompson provides no basis for his 50% conditional probability

58. (KRO) Dr. Thompson asserts, based on the direct experience from Fukushima (i.e., the operators' difficulty in implementing mitigative measures and pool-fire precursor events occurring at Fukushima) as well as the alleged inadequacies of the mitigative measures in place for U.S. plants, that there is a substantial conditional probability of a spent fuel pool fire occurring following the event of a severe reactor accident that leads to an early release. He contends that the 50% conditional probability assumed in his 2006 Report "continues to be a reasonable assumption for the purposes of SAMA analysis." Thompson Report at 20, 27. This claim of Dr. Thompson's is, however, invalid for the reasons discussed below.

59. (KRO) At the outset, Dr. Thompson's claim that the 50% conditional probability from his 2006 Report "continues to be a reasonable assumption for the purposes of SAMA analysis" ignores the Commission's rejection of that very same claim in its 2008 Rulemaking Denial. The Commission found Dr. Thompson's claim that 50% of all severe reactor accidents

[41] In this respect, it is now believed that the reports that a hydrogen explosion occurred in Unit 4 may have been caused by hydrogen that was generated by the core damage at Unit 3 back-flowing from the Unit 3 standby gas system lines through the vent lines of Unit 4. Fukushima Report at IV-90; IAEA Report at 33.

that result in an early release will also lead to a spent fuel pool zirconium fire to be an "unsubstantiated assumption" that lacked technical or analytical support. The Commission stated:

> The Thompson Report does not identify the necessary sequence of events by which such scenarios might lead to SFP zirconium fires, or discuss the probability of their occurrence.

73 Fed. Reg. 46,209. The Commission referred to litigation for the Shearon Harris plant where similar claims by Dr. Thompson were rejected based on more complete and mechanistic quantitative assessments performed by the Staff and the licensee which showed the actual probabilities of a spent fuel pool zirconium fire to be much lower. Id. at 46,209-10.

60. (KRO) In his current Report, Dr. Thompson ignores the Commission's rejection of his 50% conditional probability claim and makes no defense of it in light of the many criticisms leveled by the Commission. Similar to his 2006 Report, his current Report does not identify the necessary sequence of events that might lead to SFP zirconium fires under the different possible initiating events or discuss the probability of their occurrence. Rather, like his 2006 Report, it continues to rest on his unsubstantiated assumptions that there is a 50% conditional probability of a spent fuel fire assuming an early release.

61. (KRO) Rather than addressing the deficiencies identified by the Commission in its Rulemaking Denial, Dr. Thompson now solely relies upon his claimed inadequacy of the mitigative measures in place at U.S. plants. However, as discussed in paragraph 48 above, his reliance on mitigative measures ignores the fact that the Commission has determined based on extensive technical studies without relying on mitigative measure that the potential for a zirconium is "extremely rare," or of "very low likelihood."

62. (JRL) Moreover, as discussed in paragraphs 49-55 above, Dr. Thompson's claim that mitigative measures in place at U.S. plants are inadequate rests by and large on his mistaken

assumption of the equivalence of the mitigative measures used in Japan and those put in place at all U.S. plants following September 11. While he asserts that his conclusion is also based on his review of EDMGs in NEI 06-12, his Report only discusses, briefly, one of the suite of mitigative measures provided for in those guidelines for spent fuel pools.[42] Indeed, as discussed above, he ignores the primary mitigative measure relied upon for spent fuel pools - the use of an independently powered pump to provide makeup water to the spent fuel pool. Dr. Thompson's cursory review of one of the NEI EDMGs in his Report provides no basis for his sweeping conclusions.

63. (KRO, JRL) In short, Dr. Thompson's claim of a 50% conditional probability of a spent fuel pool fire assuming an early release continues to be merely Dr. Thompson's "unsubstantiated assumption" lacking technical or analytical support.

64. (KRO) Moreover, the recent direct experience at Fukushima relied upon by Dr. Thompson provides no support for, and in fact negates, his 50% conditional probability claim. As Dr. Thompson acknowledges, the mitigative measures employed at Fukushima "ultimately succeeded in covering the fuel with water" (Report at 27), and none of the spent fuel pools at Fukushima experienced a zirconium fire, even though there were meltdowns (i.e., severe accidents) in three of the Fukushima Daiichi reactors. Therefore, following Dr. Thompson's own line of reasoning regarding direct experience as applied in his Issue #1, the conditional probability of a severe reactor accident also resulting in a spent fuel pool zirconium fire is zero. Indeed, none of the five plants included in Dr. Thompson's Issue #1 assessment of direct experience of core damage frequency experienced a spent fuel pool zirconium fire. This fact further affirms

[42] Dr Thompson refers to a UCS press release for "additional information on the limitations of EDMGs." Thompson Report at 20 n. 41. A press release cannot, however, supply the technical or analytical support that the Commission found lacking in Dr. Thompson's 2006 Report. Moreover, the particular press release cited by Dr. Thompson concerns mitigative measures for reactor accidents, and not for spent fuel pools, which is the subject of Dr. Thompson's claims in his Report.

the lack of any significant tie between the occurrence of a severe reactor accident and the occurrence of a spent fuel pool zirconium fire.

65. (KRO) Moreover, as a final point, even accepting Dr. Thompson's premise of a 50% conditional probability of a spent fuel pool fire assuming an early release (0.5 spent fuel pool fire/early release), the likelihood of such an event at Pilgrim would be considered remote and speculative. As stated above, the conditional probability of an early release assuming a core damage event at Pilgrim is 3.29% (0.0329 early release/core damage event). Multiplying the core damage frequency from the Pilgrim ER, taking into account external and fire events of 3.2E-5 referred to in the Thompson Report (at page 15), and discussed in supplement 29 of the GEIS[43], by the Pilgrim conditional probability of an early release of 0.0329, and by Dr. Thompson's conditional probability of 0.5 spent fuel pool fire per early release, yields a spent fuel pool fire frequency for Pilgrim of:

SFP fire frequency = (3.2E-5 core damage event per year)·(0.0329 early release/core damage event)·(0.5 spent fuel pool fire/early release)

= 5.3E-7 spent fuel fire per year.

This frequency of approximately 5 occurrences in 10 million years falls within the category of what the Commission has traditionally considered as "remote and speculative matters."

66. (KRO) Thus, in summary, events at Fukushima provide no basis for Dr. Thompson's spent fuel pool claims. In fact, the actual lessons learned from Fukushima are that the spent fuel pools are very rugged and are capable of withstanding extreme conditions (such as se-

[43] Page G-10 of Supplement 29 to the GEIS indicates, "Based on the aforementioned results, the external events CDF is approximately 3.5 times the internal events CDF (based on a seismic CDF of 1.61×10^{-5} per year, a fire CDF of 6.37×10^{-6} per year, and an internal event CDF of 6.4×10^{-6} per year). Accordingly, the total CDF from internal and external events would be approximately 4.5 times the internal events CDF.

vere earthquakes and hydrogen explosions which occurred at the Fukushima Daiichi Units) and are also capable of maintaining spent fuel for extended periods of time before makeup water is required. Fukushima shows that the likelihood of a spent fuel pool zirconium fire is "very low" as previously determined by the NRC based on its numerous studies of the potential for fires in spent fuel pools.

IV. DR. THOMPSON'S CLAIMS OF SECRECY ARE NOT NEPA ISSUES

67. (KRO) Dr. Thompson claims in his Issue #3 (Secrecy Regarding Accident Mitigation Measures) that Fukushima shows that secrecy degrades mitigation capability and that the generic guidance on mitigative measures should be made public. This issue, however, does not concern either NEPA or SAMA analysis, and is therefore not pertinent here.

68. (JRL) Furthermore, Dr. Thompson's claims concerning sharing of relevant information with persons involved with implementing mitigation measures (Thompson Report at 22) is misplaced. As discussed above, Pilgrim operators and personnel responsible for implementing mitigating actions are trained and qualified on their implementation. Furthermore, Pilgrim has memoranda of understanding ("MOUs") with local governmental agencies to provide specific support, as requested, under the direction of responsible Pilgrim personnel.

69. (JRL) Finally, Dr. Thompson's suggestion that secrecy has led to an incomplete understanding on the part of the NRC and other responsible entities of the phenomena associated with spent fuel pool fires (Thompson Report at 22-23) is not true, at least insofar as Pilgrim is concerned. Dr. Thompson provides no basis for this assertion other than his unsubstantiated assumption. As part of their training, the responsible Pilgrim personnel are provided and trained on the information necessary to implement different mitigative strategies developed for Pilgrim.

V. DR. THOMPSON'S HYDROGEN CONTROL CLAIMS ARE UNFOUNDED AND DO NOT MATERIALLY DISPUTE PILGRIM'S SAMA ANALYSIS

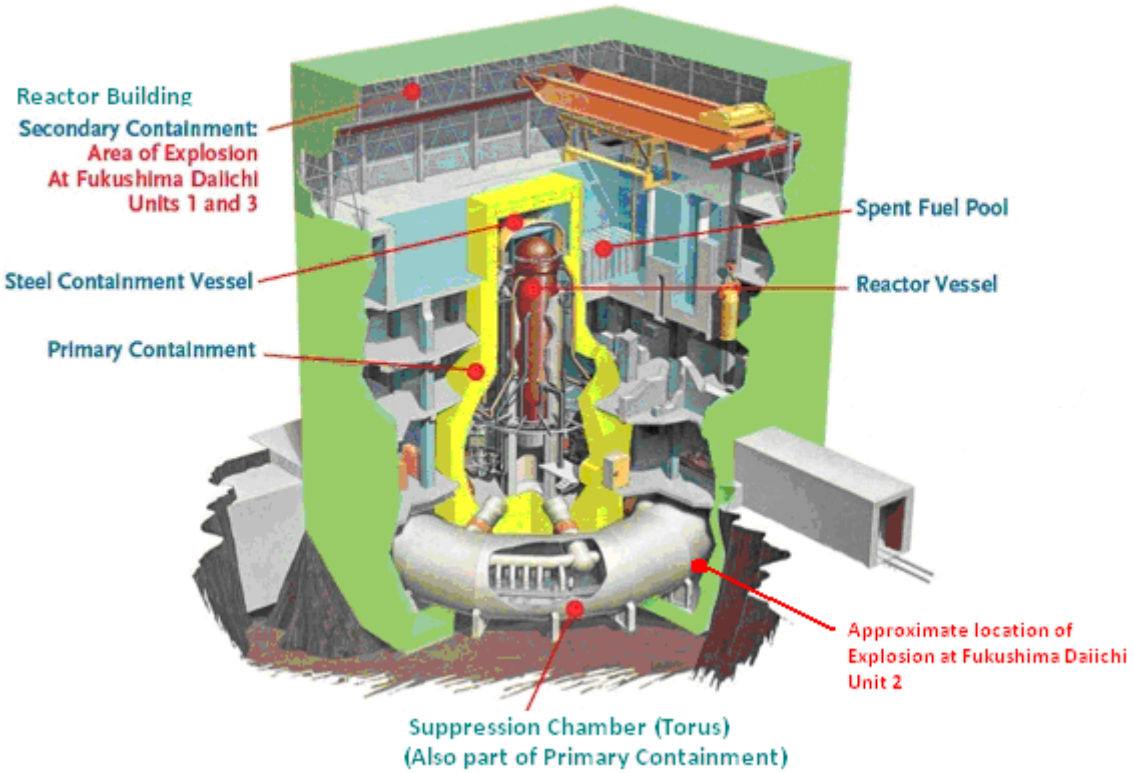
70. (JRL, KRO, LAP) Dr. Thompson claims that hydrogen explosions experienced at Fukushima could be replicated at the Pilgrim plant, and that the potential for such explosions has not been adequately considered in the Pilgrim license renewal proceeding. Thompson Report at 24-25. He further concludes (at least "provisionally") that containment venting and other hydrogen control systems at the Pilgrim plant should be upgraded, and should use passive mechanisms as much as possible. Id. at 25-26.

71. (JRL, KRO, LAP) Dr. Thompson's claims related to hydrogen control are unfounded and otherwise immaterial for several reasons. As discussed in the following paragraphs, Dr. Thompson does not show how the hydrogen explosions at Fukushima were ultimately of radiological significance, as they did not occur in the primary containment. In any event, the potential for hydrogen explosions within either the primary or secondary containments has been considered at Pilgrim. Both design features and procedures are in place at Pilgrim to control hydrogen generation and to prevent hydrogen explosions within the primary containment. Furthermore, the SAMA analysis has fully analyzed and taken into account potential hydrogen generation during the course of a reactor accident. Indeed, Dr. Thompson neither references nor discusses, let alone disputes, the Pilgrim SAMA analysis' consideration of hydrogen explosions. Moreover, Dr. Thompson could not show, based on events at Fukushima, that his hydrogen-related concerns would make any difference in the SAMA analysis because the analysis performed for Pilgrim bounds the events at Fukushima.

A. The Fukushima Hydrogen Explosions Did Not Occur in the Primary Containments

72. (JRL, KRO, LAP) Dr. Thompson overstates the significance of the hydrogen explosions that occurred at Fukushima Units 1 and 3, which did not occur in the primary containment. For Units 1 and 3, the Fukushima Report provides ample evidence that the hydrogen explosions at Units 1 and 3 occurred in the reactor building structure, sometimes referred to as the "secondary" containment, and not in the primary containment. Fukushima Report at 9, 32-33, IV-6. See also IAEA Report at 32 ("the first hydrogen explosion occurred at the site in the Unit 1 reactor building"), 33 ("a hydrogen explosion occurred in the Unit 3 reactor building"). Further evidence that the hydrogen explosions did not occur in the primary containments for Units 1 and 3 is that data summarized in the Fukushima Report show no indication of rapid pressure increase, followed by rapid pressure decrease, for these units. Figure 1 below depicts our understanding of where the explosions occurred, outside the primary containment, for Units 1 and 3.

Figure 1. Cross-sectional view of Fukushima Primary and Secondary Containments



73. (JRL, KRO, LAP) The distinction between the primary containment and reactor building is important because the primary containment is the structure designed to contain radioactive releases from the reactor coolant pressure boundary. The reactor building houses the primary containment, which is the robust steel structure that houses the reactor vessel containing the nuclear fuel. Although the leakage pathways have not been clearly identified, hydrogen (and radioactive material) was leaked to the Fukushima Units 1 and 3 reactor buildings, ultimately resulting in hydrogen explosions. The result is that some gases that were intended to vent directly to the atmosphere first collected in the reactor buildings and then were emitted into the atmosphere after the explosions.

74. (JRL, KRO, LAP) Because the Fukushima Units 1 and 3 hydrogen explosions occurred outside of the primary containments, there was no large release of radioactive materials that one would otherwise expect had the primary containments suffered catastrophic failures. The radioactivity that was released was essentially the same that would have been released had all of the gases vented directly to the atmosphere. In other words, some radioactive gases that were intended to vent directly to the atmosphere first collected in the reactor buildings and then were emitted into the atmosphere when the explosions occurred.

75. (JRL, KRO, LAP) The information on the status of Unit 2 is not as clear as it is for Units 1 and 3. With respect to Unit 2, it is suspected that a hydrogen explosion occurred in the torus room, i.e. the room housing the torus at the lower elevations of the reactor building. Fukushima Report at 11, IV-35, IV-63. The location of the suspected explosion is depicted in Figure 1 above. Although the torus is part of the primary containment, the torus room is not. It is possible, however, that the torus itself, which is also known as the suppression chamber, was damaged. Id. at V-33.

B. Pilgrim Has Procedures in Place to Prevent Hydrogen Explosions in Primary Containment

76. (JRL, KRO, LAP) As Dr. Thompson recognizes, Thompson Report at 24, the potential for hydrogen explosions is not new information. This potential has been recognized by the industry at least since the accident at Three Mile Island, and regulations are in place to ensure that combustible gases are controlled to minimize this potential (see 10 C.F.R. § 50.44). These regulations focus on preventing hydrogen explosions that would damage the primary containment because it is the structure that is designed to contain radioactive releases. Mark I containments, such as at Pilgrim, have an inert atmosphere (consisting of non-combustible nitrogen gas) to preclude the possibility of a hydrogen combustion event within the containment. Indeed, the

Fukushima Report notes that, whereas measures were in place to prevent hydrogen explosion within the primary containments, such measures were not in place to prevent hydrogen explosion in the reactor building. Fukushima Report at IV-6.

77. (JRL) Further, Pilgrim's venting procedures work to assure that sufficient hydrogen will not accumulate within the primary containment. Entergy's operational and severe accident procedures clearly identify the actions that are to be undertaken by plant personnel under different plant circumstances. These procedures require Entergy to vent the primary containment using the direct torus vent ("DTV") long before the Fukushima operators attempted that same operation. Pilgrim Emergency Operating Procedures EOP-03 and 5.4.6 detail the steps that operators are to follow, starting at a containment pressures of 2.2 psig, for venting using the standby gas treatment system ("SGTS") to restore containment pressure to less than 2.2 psig. Multiple piping pathways are available to reduce containment pressure below 2.2 psig. The first is a two-inch (2") low capacity path, second is a twenty-inch (20") high capacity path, and third is the one-inch (1") Containment Atmospheric Dilution ("CAD") system path. These three paths are directed through the SGTS. Should use of these three paths fail to restore pressure to less than 2.2 psig, the procedures direct that the eight-inch (8") DTV is to be used before reaching a primary containment pressure of 56 psig. The DTV bypasses the bulk of the SGTS, but connects to the SGTS discharge piping that leads to the main stack, an elevated release point. Based on the data provided in the Fukushima Report, it appears that pressure inside Unit 1's primary containment reached 101.5 psig before venting was performed, approaching twice the level (56 psig) at which Pilgrim's procedures would <u>require</u> opening the DTV.

78. (JRL) In addition, Pilgrim's Control Room Shift Manager has the authority to direct operation of the DTV in accordance with Pilgrim's specific procedures long before reaching

a level that could challenge the primary containment, so that authorization from someone outside the plant is not needed. Based on multiple references in the Fukushima Report, the level of authority required to allow use of the DTVs at Fukushima was a "Minister" level in the government. With multiple nuclear units involved, and infrastructure unavailable because of the earthquake, tsunami, and nuclear emergency, the delays in operating the DTV are therefore explainable, but would not be analogous to Pilgrim, where the decision and authority to operate the DTV in order to vent the primary containment rests with the Shift Manager.

C. Hydrogen Explosions are Accounted for in the Pilgrim SAMA

79. (KRO, LAP) As a result of the fact that the potential for hydrogen explosion has long been known, hydrogen explosions are fully considered in the Pilgrim SAMA analysis. As described in the license renewal application environmental report ("ER"), the SAMA analysis is based on a three-level, plant- and site-specific probabilistic safety analysis ("PSA") (also referred to as probabilistic risk analysis, or "PRA"). The first two levels of the PSA are used to determine the radioactive releases from the plant used to calculate off-site consequences. The Level 1 PSA assesses the scenarios that could lead to core damage. The Level 2 PSA examines the response of the containment and its protection systems to the accident phenomena and related physical processes from core damage accident sequences that can lead to a radiological release.

80. (KRO, LAP) Plant damage states ("PDS") provide the interface between the Level 1 and Level 2 analyses (i.e., between core damage accident sequences and fission product release categories). In the PDS analysis, Level 1 results are grouped ("binned") according to plant characteristics that define the status of the reactor, containment, and core cooling systems at the time of core damage. This ensures that systems important to core damage in the Level 1 event trees, and the dependencies between containment and other systems are handled consistently in the

Level 2 analysis. A PDS therefore represents a grouping of Level 1 sequences that defines a unique set of initial conditions that are likely to yield a similar accident progression through the Level 2 containment event trees ("CETs") and the attendant challenges to containment integrity. Table E.1-3 of the ER provides a list of the risk-significant basic events (component failures, operator actions, and initiating events) from the Level 1 PSA model and indicates the SAMA(s) evaluated in the cost-benefit analysis for each of them. Table E.1-8 of the ER summarizes the core damage accident sequence PDSs.

81. (KRO, LAP) The Pilgrim Level 2 model includes two types of considerations: (1) a deterministic analysis of the physical processes for a spectrum of severe accident progressions, and (2) a probabilistic analysis 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 and analyzes on a deterministic basis under what conditions the containment will fail. It includes consideration for several hydrodynamic and heat transfer phenomena that occur during the progression of severe accidents, including debris coolability, pressure spikes due to ex-vessel steam explosions, direct containment heating, molten debris filling the pedestal sump and flowing over the drywell floor, containment bypass, explosion of hydrogen, thrust forces at reactor vessel failure, and liner melt-through.

82. (KRO, LAP) The probabilistic element of the Level 2 analysis consists of a containment event tree ("CET") with functional nodes that represent different phenomenological events and containment protection system status and determines probabilistically for each functional node the likelihood of containment failure or containment bypass. Table E.1-5 of the ER lists the functional nodes considered in the Level 2 analysis and describes the failure mechanisms that contribute to each. As discussed later, these nodes consider hydrogen explosions.

83. (KRO, LAP) Evaluation of the forty-eight PDSs through the CET resulted in hundreds of Level 2 accident progression sequences and thus hundreds of source terms for internal initiators, making calculation with MACCS2 cumbersome. Thus, following standard PRA methodology practice, the source terms were grouped into a smaller number of source term groups defined in terms of similar properties, with a frequency weighted mean source term for each group. The consequence analysis source term groups are represented by 19 collapsed accident progression bins ("CAPBs"). The CAPBs represent a range of severe accident releases from small to very large. For each of the 19 CAPBs, a series of simulations were run using the MACCS2 Code to evaluate postulated consequences. The 19 CAPBs are presented in ER Table E.1-9. The 19 CAPBs account for postulated system, structure, and component failures, the status of the reactor pressure vessel, the status of the containment, and accident sequence timing. Each CAPB represents a different combination of plant feature status and release mechanism and has a characteristic frequency and source term release based on attributes of the accident. The CAPBs have different characteristics to describe the occurrence of core damage, the occurrence of vessel breach, primary system pressure at vessel breach, the location of containment failure, the timing of containment failure, and the occurrence of core-concrete interactions.

84. (KRO, LAP) The source term for each CAPB comes from the Level 2 deterministic analysis, which considers the hydrodynamic and heat transfer phenomena that occur during the progression of each CAPB severe accident. To simplify, for each CAPB, the deterministic analysis considers what plant features have failed and what plant features have succeeded in that accident sequence. It then considers the hydrodynamic and heat transfer phenomena that would apply with the plant in that configuration and determines (1) the fraction of the core inventory of each of nine radionuclide groups present in the reactor core that would be released; (2) the heat

energy in the plume associated with the release (which will cause the plume to rise); (3) the height of the release; (4) the timing of release; and (5) the release duration. These parameters are the source terms for each of the CAPBs reported in ER Table E.1-11 and input to the MACCS2 analysis.

85. (KRO, LAP) The potential for hydrogen explosion within the primary containment (which, as previously discussed, did not occur at Fukushima Units 1 and 3) is considered in the Level 2 PRA model for Pilgrim. The probability of failure of each functional node is determined by a subordinate fault tree analysis. Hydrogen explosion is considered a credible mechanism for early primary containment failure and therefore contributes to the CFE ("Containment Failure, Early") functional event node, which considers the potential loss of containment integrity at, or before, reactor vessel failure. As can be seen in ER Table E.1-5, the CFE node considers that the containment may fail soon after failure of the reactor pressure vessel due to overpressure or hydrogen explosion. The CFE node states:

> This top event node considers the potential loss of containment integrity at, or before, vessel failure. Several phenomena are considered credible mechanisms for early containment failure. They may occur alone or in combination. The phenomena are containment isolation failure; containment bypass; <u>containment overpressure failure at vessel breach; hydrogen deflagration or detonation</u>; fuel-coolant interactions (steam explosions); high pressure melt ejection and subsequent direct containment heating; and drywell steel shell melt-through.

ER Table E.1-5 (emphasis added). Thus, the subordinate fault tree for CFE includes component and operator action failures that could result in a buildup of hydrogen in the primary containment. Following a core damage event with reactor pressure vessel failure, the Level 2 model assumes that failing to vent the containment could result in overpressure of the containment or in hydrogen explosion within the containment.

86. (KRO, LAP) CAPB 4 through CAPB 11 include accident sequences in which early containment failure occurs. Therefore, primary containment hydrogen explosion events have been appropriately considered in the Level 2 PRA model used in the Pilgrim SAMA analysis.

87. (KRO, LAP) The potential for hydrogen explosion within the reactor building is also appropriately considered in the Level 2 PRA model for Pilgrim, which was used in the SAMA analysis. The adverse impact of an explosion of hydrogen that had accumulated in the reactor building instead of being emitted through the plant stack would be a ground level release of radiation to the environment, and no more radioactive material than had the release occurred through the plant stack. The CET in the Pilgrim Level 2 PRA model includes functional node Reactor Building ("RB"), which is used to assess the ability of the reactor building to retain fission products released from containment. Success of event RB is defined to be a reduction of the containment release magnitude. Failure of event RB means that containment failures or bypasses are released directly to the environment without being held up in the reactor building.

88. (KRO, LAP) In summary, although it may be possible for the hydrogen explosions experienced at Fukushima to be replicated at the Pilgrim plant, the potential for such explosions has been considered in the Pilgrim PRA model used for the SAMA analysis in the license renewal proceeding. While Dr. Thompson suggests that the potential for hydrogen explosions has not been adequately considered in the Pilgrim license renewal proceeding, Thompson Report at 25, nowhere in his Report does Dr. Thompson reference, discuss, or otherwise dispute the means by which hydrogen explosions are in fact considered in the Pilgrim SAMA analysis. Thus, Dr. Thompson's Report raises no material dispute concerning the adequacy of the Pilgrim SAMA analysis consideration of hydrogen explosions.

D. The Pilgrim SAMA Analysis Bounds the Fukushima Radioactive Releases

89. (KRO) Finally, Dr. Thompson could not show, based on events at Fukushima, that his hydrogen-related concerns would make any difference in the Pilgrim SAMA analysis because the analysis performed for Pilgrim bounds the events at Fukushima. As discussed in a prior Declaration previously submitted to the Board in this proceeding,[44] comparison of the radiological releases assumed in the single-unit Pilgrim SAMA analysis shows that the Pilgrim SAMA analysis accounts for severe accident releases that more than bound the reported releases from all of the Fukushima units. Sowdon/O'Kula Declaration at ¶ 41.

90. (KRO) Subsequent to the development of the comparisons in the Sowdon/O'Kula Declaration, the Japanese authorities increased their estimate of the radioactive release from Fukushima by about 22% above the estimates used in the Sowdon/O'Kula Declaration.[45] This increase has no effect on the conclusions drawn from the comparisons made in Table 5 of the Sowdon/O'Kula Declaration. As noted there, "even if Fukushima radionuclide release estimates were to double, CAPB-15 (which contributes over 80% of the PDR and OECR to the Pilgrim SAMA analysis) would still bound the estimated I-131 releases from all of the Fukushima facilities by about a factor of two (1.78) and the estimated Cs-137 releases by about a factor of three (2.66)." Sowdon/O'Kula Declaration at 24 n.16. Thus, the radionuclide releases assumed in the Pilgrim SAMA analysis far exceed actual releases at Fukushima.

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[44] Declaration of Dr. Thomas L. Sowdon and Dr. Kevin R. O'Kula in Support of Entergy's Answer Opposing Pilgrim Watch Request for Hearing on Post-Fukushima SAMA Contention (June 6, 2011) ("Sowdon/O'Kula Declaration").

[45] The comparisons made in Sowdon/O'Kula Declaration were based on release estimates for cesium and iodine made by the Japan Nuclear Safety Commission ("NSC"), which estimated that 630 petabecquerels ("PBq") or 6.3E+17 Bq had been released from Fukushima. In June, the Nuclear and Industrial Safety Agency ("NISA"), another Japanese regulatory authority, increased its original estimate of 370 PBq (which had been much lower than the NSC's estimate used in the Sowdon/O'Kula Declaration) to 770 PBq, which is approximately 22% higher than the NSC estimate. The iodine equivalent radiological hazard for both the NSC and the NISA are determined by multiplying the estimated Cs-137 release by a factor of 40 to account for its higher radiological hazard and adding this to the estimated iodine activity.

91. (KRO) The Pilgrim SAMA analysis has considered larger radioactive releases than that which occurred at Fukushima, and it is thus not necessary to redo the SAMA analysis to take account of the radiological releases from the Fukushima accident.

VI. DR. THOMPSON'S CLAIMS CONCERNING A FILTERED DTV DO NOT MATERIALLY DISPUTE PILGRIM'S SAMA ANALYSIS

92. (KRO, LAP) Dr. Thompson reaches a "Provisional" finding concerning the need for a filtered DTV based on the reported release of radioactive material to the atmosphere from the Fukushima Daiichi plants. The finding is that filtered venting of the Pilgrim reactor containment could substantially reduce the atmospheric release of radioactive material from an accident at Pilgrim and that filtered venting of containment should be considered in a revised SAMA analysis for Pilgrim. Thompson Report at 29.

93. (KRO, LAP) Dr. Thompson's concerns fail to materially dispute the SAMA analysis. The Pilgrim SAMA analysis comprehensively analyzed a host of potential mitigation measures, and concluded that adding a filtered DTV would not be cost beneficial. Dr. Thompson's brief analysis of the issue based on events at Fukushima makes no showing that venting under accident conditions may have been inadequately considered in the Pilgrim SAMA analysis. Indeed, Dr. Thompson nowhere references or discusses, let alone disputes, the Pilgrim SAMA analysis' consideration of the DTV.

94. (KRO, LAP) The SAMA analysis is a detailed and systematic assessment of potentially cost beneficial enhancements that could further reduce accident risk. In other words, the SAMA assessment is a cost-benefit analysis and not a safety analysis, as Dr. Thompson suggests in his Report. It compares the benefits associated with a mitigation measure with the cost to im-

plement. The Pilgrim SAMA analysis was performed using site specific hazard considerations such as seismic and flooding events that could challenge Pilgrim.

95. (KRO, LAP) The installation of a filtered DTV claimed to be necessary by Dr. Thompson was considered in Pilgrim's SAMA analysis (SAMA Number 2) in the Pilgrim License Renewal Application (LRA) and a subsequent RAI response.[46] This SAMA would essentially provide an alternate decay heat removal method with the fission products being scrubbed through the installed filter. This is not a new concept. Prior to Pilgrim's evaluation of the mitigation measure, other U.S. plants have considered it and screened it from consideration in their SAMA analyses because it was not found to be cost-beneficial.

96. (KRO, LAP) For the Pilgrim SAMA analysis, the filtered DTV was likewise not found to be potentially cost beneficial. The maximum benefit each SAMA could have – the present value of the complete averted cost of an accident – was compared to the implementation cost of the SAMA. The cost-benefit approach outlined in NUREG/BR-0184, Regulatory Analysis Technical Evaluation Handbook,[47] was used to calculate the baseline benefit (averted cost).

97. (KRO, LAP) It is important to note that any radioactive releases that would pass through the DTV would be scrubbed prior to release because the DTV vents from the air space in the torus (a pool of water). Iodine products in the drywell atmosphere are scrubbed by bubbling through the torus water when the direct torus vent is used. The condensation of steam and increase of available volume from the reactor pressure vessel to the torus also provides delay with

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[46] Entergy Response to Request for Additional Information Regarding Severe Accident Mitigation Alternatives for Pilgrim Nuclear Power Station (TAC No. MC9676) (July 5, 2006) (Exhibit No. ENT000007), Table RAI-3-2, Revised Summary of Phase II SAMA Analysis. SAMA Number 19 in the LRA is a duplicate of SAMA Number 2, which was inadvertently not screened during Phase I.

[47] NUREG/BR-0184, Regulatory Analysis Technical Evaluation Handbook, Final Report (Jan. 1997) ("NUREG/BR-0184").

resulting benefit of decay. In short, adding a DTV filter would mean filtering a radioactive release that has already been scrubbed, which reduces the benefit that would be obtained by installing the filter.

98. (KRO, LAP) The cost of installing a filtered DTV is approximately $3 million. The benefit of additional filtering capability was estimated for SAMA Number 2 using the Pilgrim PRA model assuming that the amount of radioactive material released through the DTV was reduced by half. The use of a filtered DTV resulted in an 18% reduction in off-site population dose risk and a benefit of approximately $872,000. As a result, the filtered DTV in the SAMA analysis fails the implementation criteria by more that a factor of 3.[48]

99. (KRO, LAP) As stated, nowhere in his Report does Dr. Thompson discuss or take issue with this SAMA cost-benefit analysis for the filtered DTV. Moreover, as with the Dr. Thompson's hydrogen control claims, this claim is not based on new information because the lack of a filtered DTV at Pilgrim was reflected in the SAMA analysis filed with the license renewal application.

VII. CONCLUSION

100. (JRL, LAP, KRO) We have thoroughly evaluated the claims in Dr. Thompson's Report against information in the Pilgrim ER, the applicable accepted standards for performing PRAs and available relevant studies and reports. Based on our evaluation we conclude that none of Dr. Thompson's claims have merit, and none of them identify a new or significant environmental issue revealed by Fukushima that would impact the Pilgrim plant.

[48] See Exhibit No. ENT000007, Table RAI-3-2, Revised Summary of Phase II SAMA Analysis

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

Executed in Accord with 10 C.F.R. § 2.304(d) Joseph R. Lynch Manager, Licensing Pilgrim Nuclear Power Station 600 Rocky Hill Rd. Plymouth, MA 02360 Phone: 508-830-8403 E-mail: jlynch4@entergy.com	Executed in Accord with 10 C.F.R. § 2.304(d) Lori Ann Potts Entergy License Renewal Services & ANO NFPA-805 Transition Project 1448 SR 333 Russellville, AR 72802 ANO-GSB-45 Phone: 479-858-3529 Email: lpott90@entergy.com
Executed in Accord with 10 C.F.R. § 2.304(d) Kevin O'Kula, Advisory Engineer URS Safety Management Solutions LLC 2131 South Centennial Avenue Aiken, South Carolina 29803-7680 Phone: 803.502.9620; Email: kevin.okula@wsms.com	

EXHIBIT 1

600 Rocky Hill Road
Plymouth, MA 02360

(508) 830-8403 (Work)
(508) 728-1421 (Cell)
E-mail: jlynch4@entergy.com

Joseph R. Lynch Jr.

Objective

Senior Manager/Regulatory Affairs with 29 years of nuclear power experience and background in engineering, licensing/regulatory affairs, environmental compliance, creative problem solving, stakeholder communications, complex project management, cost control, budgeting and employee management. Strong strategic thinker and team builder.

Areas of Expertise

- Regulatory Affairs/Licensing
- Project Management
- Design and Systems Engineering
- Environmental Health & Safety
- Compliance
- Oral and Written Communications

Education

Worcester Polytechnic Institute – Worcester, Massachusetts
Bachelor of Science in Mechanical Engineering (BSME)
Specialized in Thermo/Fluids/Nuclear

Graduate Studies in Business Management, Communications and Regulatory Compliance

Numerous Internal and External Management Courses – Yankee Atomic Electric Company; Vermont Yankee Nuclear Power Corporation

Professional experience

2007-Present Entergy Nuclear Operations

Licensing Manager

Manages the Pilgrim Nuclear Power Station Licensing Group supporting the operation and regulatory compliance of the station in accordance with NRC, State and Federal regulations, permits and statutes.

Development of all necessary letters, licensing correspondence and regulatory approvals from NRC, local, state and federal agencies required in support of plant operations.

- Responsible for communicating with regulators, governmental representatives and media on plant status, regulatory issues and emergent events.
- Management of a $ 6-8 million dollar annual department budget.

2006-2007 Environmental Resources Management (ERM)

Senior Nuclear Consultant

Directs ERM's national nuclear team, and leads, coordinates, or supports nearly all of the nuclear-related work on behalf of ERM nationally.

Responsible for the oversight of all technical and policy-related activities of the nuclear staff experts within ERM. Maintains a close awareness of current and emergent regulatory issues within the nuclear sector, developing trends, and industry initiatives. Recent projects include;

- Support of Groundwater Program development at several nuclear power plants in accordance with EPRI/NEI Guidance Documents and plant specific attributes.
- Regulatory affairs, licensing and permitting responsibility for nuclear client.
- Environmental, Health & Safety (EHS)/Due Diligence Assessments for several nuclear clients supporting Merger & Acquisition (M&A) efforts.

2003-2006 Yankee Atomic Electric Company

Director, Regulatory Affairs, Licensing and Site Closure

Directed and managed the Site Closure Project activities for the clean-up and decommissioning of the Yankee Nuclear Power Station in strict compliance with NRC, State and Federal regulations, permits and statutes.

Development of strategies to obtain all necessary permitting, licensing and regulatory approvals from NRC, local, state and federal agencies required to remediate all environmental hazards from the site in support of unrestricted re-use of the property.

- Responsible for communicating with community members, regulators, governmental representatives and media on project status, key company decisions and emergent issues.
- Authored the Site Closure Project Plan (SCPP), an industry first, comprehensive plan that integrated stakeholder input, corporate goals and regulatory compliance by working with local, state and federal stakeholders to solicit input and acceptance.
- Extensive outreach to stakeholders via written and verbal communication including site personnel, executive management, community advisory boards, local/state government leaders, town meetings, community and media events.
- Control and management of a $ 6-8 million dollar annual project budget.

2000-2003 Connecticut Yankee Nuclear Power Company

Decommissioning Oversight Manager

Directed and coordinated oversight of the construction and plant demolition activities at the Connecticut Yankee Nuclear Power Station in accordance with applicable regulatory standards.

Position responsibilities included, but were not limited to the following;

- Implemented department activities in accordance with established and newly developed station programs, policies and procedures.
- Assured a safety conscious work environment, including implementation of the Standards of Conduct and the Site Corrective Action Program.
- Communicated site performance to stakeholders including site personnel, executive management, Independent Assessment groups, community members, regulators, governmental representatives and the media.
- Assured decommissioning activities did not affect safe storage of spent nuclear fuel.
- Assisted site executive management in establishing and implementing strategic plans.

1997-2000 Vermont Yankee Nuclear Power Corporation

Department Manager – Design Engineering

Responsible for management of plant design modifications, thermal/hydraulic design analyses and plant support functional areas for a twenty (20) engineer staff, including consultants, contractors and administrative support. Control and oversight of a $ 3-4 million dollar annual budget.

Position responsibilities included;

- Interface with U.S. Nuclear Regulatory Commission (USNRC) through Senior Resident and /or Region I Project Manager.
- Reviewed and approved all departmental design work products.
- Preparation, review and approval of Bases for Maintaining Operation (BMO).
- Preparation and maintenance of plant Design Basis Documentation (DBD).
- Preparation, review and approval of 10CFR50.59 Safety Evaluations.
- Employee performance appraisals, bonus/compensation determination and goal setting.
- Qualified adjunct instructor for providing training to engineering staff.
- Maintained the Department budget by tracking expenditures on capital and O&M Projects, contractor costs and employee salaries/benefits.
- Identified and developed Continuous Process Improvements (CPI) initiatives with plant management, supervision, and staff to improve overall performance of engineering work products.

1982-1997 Yankee Atomic Electric Company

Manager – Design Engineering Fluid Systems (1995-1997)

Supervised the Design Engineering Fluid Systems staff supporting the Vermont Yankee Nuclear Power Station.

- Responsible for the review and approval of design change packages, calculations and analyses, 10CFR50.59 Safety Evaluations, Operability Determinations, Bases for Maintaining Operability, Design Basis Documents (DBD) and Department assigned commitments/corrective actions.

- Responsible for oversight of the VY MOV, and Safety Classification Programs.
- Responsible for planning and scheduling of all assigned work, outage preparation/implementation activities, project budget accountability and direction of contractor staff.

Motor-Operated Valve (MOV) Program Manager (1994-1995)

Directed Connecticut Yankee's efforts for planning and implementation of the Generic Letter 89-10 MOV Program testing and overhaul activities through the 1995 Refueling Outage (RFO).

- Coordinated the completion of all documentation in support of NRC inspection and closure of the GL 89-10 imposed requirements for safety-related MOVs.

Lead Systems Engineer (1992-1994)

Project Manager for the Millstone Unit 1 Hardened Wetwell Vent System design and implementation.

- Instrumental in engineering and development of all design change documents.
- Responsible for the management of all work projects for Northeast Utilities, within the Systems Engineering discipline.

Senior Project Engineer (1990-1992)

Served as Project Manager for the Yankee Nuclear Power Station (YNPS) high-pressure turbine retrofit and Main Condenser replacement projects.

- Provided project management, engineering and scheduling oversight.
- Worked extensively on condition assessment, performance monitoring and replacement justification for the YNPS Main Condenser.
- Negotiated the Contracts for acquisition of the HP Turbine and Main Condenser.

Systems/Senior Systems Engineer (1982-1990)

Designed, specified, and analyzed nuclear power plant fluid/air systems and equipment at the Yankee Nuclear Power Station.

- Provided technical assistance and installation supervision on primary and secondary plant equipment and systems.
- Designed/installed the Safe Shutdown System (Appendix R requirement for remote shutdown of YNPS), Emergency Diesel Generator (EDG) and Safety Injection Building ventilation upgrades, EDG replacement and commercial grade dedication of the EDGs.
- Shift outage coordinator for the 1990 summer Refueling Outage at YNPS.
- Worked closely with the plant staff in planning, prioritizing and craft labor oversight/support.

EXHIBIT 2

Entergy

RESUME
Lori Ann Potts

EDUCATION

B.S., Nuclear Engineering, The Pennsylvania State University, 1981

EXPERIENCE

February 2008 - Present

Consultant to Entergy Nuclear – NFPA-805 Transition Project Team

Development of products and documents associated with the ANO-1 and ANO-2 Fire PRAs.

- Plant Boundary Definition, Plant Partitioning, and Fire Ignition Frequencies
- Fire PRA Component Selection and Fire-Induced PRA Model
- Post-Fire Human Reliability Analysis
- Fire Risk Quantification
- Fire PRA Peer Reviews Preparation, Execution, and Responses
- Risk assessments of Fire Protection non-compliances and issues

February 2002 - Present

Consultant to Entergy Nuclear – License Renewal Project Team

- Mechanical Aging Management Reviews and Development of Aging Management Programs for DC Cook, Pilgrim, Vermont Yankee, Palisades, ANO-2, J.A. Fitzpatrick, Indian Point, and Cooper License Renewal Projects
- Responding to NRC questions on submitted applications
- Coordinating and reviewing evaluation of severe accident mitigation alternatives (SAMA) for ANO-1, ANO-2, Pilgrim, Vermont Yankee, J.A. Fitzpatrick, Indian Point, Cooper, and Grand Gulf Environmental Reports
- Peer reviewed SAMA analyses for Beaver Valley, Columbia, and Palo Verde
- Developed industry SAMA guideline (NEI 05-01)

01/1994 - 08/2001

Consultant to Entergy Operations - Arkansas Nuclear One (ANO) Nuclear Safety Analysis

- Project Manager on ANO-2 Probabilistic Safety Assessment (PSA) Model Update
- Risk Analysis of the ANO-2 Power Uprate
- Risk Sensitivity Analysis of alternate repair criteria for ANO-1 Steam Generator tubes containing Intergranular Attack
- Power Uprate/Steam Generator Replacement modification of ANO-2
- Fortran code to calculate time to boil and time to core uncovery upon

loss of shutdown cooling

- Reviewed Safety Analyses for ANO-2 Steam Generator Replacement and Power Uprate
- Created ANO PSA Analysts' Deskguide
- Updated ANO-1 and ANO-2 PSA models and associated analyses

11/1989 - 06/1993 Arkansas Nuclear One (ANO), Entergy Operations, Inc.

Senior Engineer, Nuclear Engineering Design (05/91-06/93)

- Lead Engineer for Unit 1 PSA
- Responsible for documentation of Design Basis for reactivity related design basis accidents on Unit 2

Reactor Engineer III, System Engineering (11/89 - 05/91)

- Responsible for Unit 2 Core Protection Calculators, Core Operating Limits Supervisory System and Excore Nuclear Instrumentation
- Defined core offload, shuffle and reload sequence for Unit 2 Cycle 9
- Startup Physics Testing and Core Monitoring Surveillances

02/1988 - 11/1989 Plant A. W. Vogtle, Georgia Power Company

Senior Plant Engineer, Outage Management

- Project-2 scheduling for Refueling and mid-cycle Outages
- Performed Critical Path Analyses, Plots and Reports

01/1987 - 08/1987 Consultant to Pilgrim Power Station

Senior Systems Specialist, I&C (03/87 - 08/87)

- Ensured Neutron Monitoring, Radiation Monitoring, Reactor Water Level, Turbine Generator Protection and Controls, Recirculation System Controls and Communications systems were operational and ready for start-up of the plant from its extended outage
- Acted for Lead Systems Specialist, I&C and Electrical in his absence; supervising six engineers

NPRDS Administrator (01/87 - 03/87)

- Reviewed plant Maintenance Requests, Malfunction Reports and Design Changes for reportable failures
- Generated failure and out-of-service reports

01/1985 - 12/1985 Consultant to Arkansas Nuclear One

Principal Engineer

- Document Research and Engineering Evaluation for all components in both units to generate a computerized Component Database
- Supervised four technicians and five engineers

04/1984 - 12/1984 Clinton Power Station, Illinois Power Company

Plant Maintenance Engineer

- Resolved abnormal condition reports, commitments and audit findings
- Walk-downs of systems being turned over from Start-Up to Operations
- Developed maintenance program for Environmental Qualification of equipment
- Assisted in development of Computerized Maintenance Management System

05/1981 – 03/1984 EG&G Idaho, Inc.

Three Mile Island, I&C and Electrical Program Engineer

- Performed failure and survivability testing of electrical components and instrumentation within the damaged Unit 2 reactor building
- Developed test procedures, instructed technicians in use of equipment and directed performance of tests
- Reviewed resultant data and test plans for off-site examinations
- Presented technical reports describing the program status to DOE, NRC and the industry

05/1980 – 08/1980 EG&G Idaho, Inc.

Three Mile Island, Intern

- Participated in and graphed data from survivability testing of electrical components and instrumentation within the damaged Unit 2 reactor building

EXHIBIT 3

KEVIN R. O'KULA

Advisory Engineer

URS Safety Management Solutions LLC

2131 South Centennial Avenue
Aiken, South Carolina 29803-7680
Telephone: 803.502.9620 – Email: kevin.okula@wsms.com

KEY AREAS:

- **Computer Model Verification and Validation**
- **Accident and Consequence Analysis for Design Basis Accident Support**
- **Regulatory Standard & Guidance Development**
- **New Reactor Design Accident Analysis and PRA Support**
- **Severe Accident and Quantitative Risk Analysis**
- **Level 2/3 Probabilistic Risk Assessment**
- **MACCS2 Code Applications**
- **Level 3 PRA Standard Development**

PROFESSIONAL SUMMARY:

Dr. O'Kula has over 28 years experience as a manager and technical professional in the areas of accident and consequence analysis, source term evaluation, commercial and production reactor probabilistic risk assessment (PRA) and severe accident analysis, safety software quality assurance (SQA), safety analysis standard and guidance development, computer code evaluation and verification, risk management, hydrogen safety, reactor materials dosimetry, shielding, and tritium safety applications. He is a member of the American Nuclear Society (ANS) Standard working group ANS 58.25 on Level 3 Probabilistic Safety Assessment, and is a member of the Peer Review Committee for the Nuclear Regulatory Commission's (NRC's) State-of-the-Art Reactor Consequence Analysis (SOARCA) Program. Kevin was part of the Department of Energy (DOE) team writing DOE G 414.1-4, *Safety Software Guide*. He coordinated technical support for the DOE Office of Environment, Safety, and Health (EH) in addressing Defense Nuclear Facilities Safety Board (DNFSB) Recommendation 2002-1 on Software Quality Assurance (SQA), and was a consultant to DOE/EH-31 Office of Quality Assurance for disposition of SQA issues. Dr. O'Kula was a member of the Partner, Assess, Innovate, and Sustain (PAIS) Safety Case team for the Sellafield Site in the United Kingdom in the early 2009 period. The PAIS team identified and began implementation of improvement opportunities in nuclear safety and related areas. Recommendations were documented in comprehensive reports to the Site's Nuclear Management Partners consortium in March 2009.

He is, or has supported, Atomic Safety Licensing Board (ASLB) relicensing issue resolution for several commercial plants including Indian Point, Prairie Island, and Pilgrim Nuclear Power Station, on severe accident mitigation alternatives (SAMA) analysis. He was also part of the accident analysis and PRA/severe accident teams supporting the Design Certification Document for the U.S. Advanced Pressure Water Reactor (US-APWR) a joint effort with URS Washington Division and Mitsubishi Heavy Industries (MHI). He has provided similar support for an alternative reactor technology, the Pebble Bed Modular Reactor (PBMR).

Kevin is coordinating URS support to the Quantitative Risk Analysis (QRA) for evaluation of hydrogen events in a waste vitrification plant design, including fault tree and human factors areas. He is also a contributor to the DOE response on the use of risk assessment methodologies as part of the DNFSB Recommendation 2009-1 implementation action for Risk Assessment. He led work in reviewing EIS food pathway consequence analysis performed on assumed accident conditions from the Mixed Oxide Fuel Fabrication Facility (MFFF), sited at the Savannah River Site. This project compared and evaluated the impacts calculated from three computer models, including MACCS2, GENII, and UFOTRI.
He is past chair of the American Nuclear Society (ANS) Nuclear Installations Safety Division (NISD), and the Energy Facility Contractors Group (EFCOG) Accident Analysis Subgroup. He is a member of the Nuclear Hydrogen Production Technical Group under the ANS's Environmental Sciences Division, and is chair for the EFOCG Hydrogen Safety Interest Group. He was the Technical Program Chair for two ANS embedded topical meetings on Operating Nuclear Facility Safety (Washington, D.C., 2004) and the Safety and Technology of Nuclear Hydrogen Production, Control and Management (Boston, MA, 2007).

Dr. O'Kula was PRA group manager for K Reactor at the time of restart in the early 1990s. He led a successful effort demonstrating Savannah River Site (SRS) K-Reactor siting compliance to 10 CFR 100, and tritium facility compliance with SEN-35-91.
He was the project leader for independent Verification and Validation (V&V) of urban dispersion software for the Defense Threat Reduction Agency (DTRA) and is the current V&V project manager for the evaluation of several chemical/biological software tools for the U.S. Army Test and Evaluation Command (ATEC) and Chemical-Biological Program (Dugway Proving Ground (Utah) and Edgewood Chemical/Biological Center in Maryland.

EDUCATION:

Ph.D., Nuclear Engineering, University of Wisconsin, 1984
M.S., Nuclear Engineering, University of Wisconsin, 1977
B.S., Applied and Engineering Physics, Cornell University, 1975

TRAINING:

Conduct of Operations (CONOPS), 1994
Harvard School of Public Health, Atmospheric Science and Radioactivity Releases, 1995
Consequence Assessment, (Savannah River Site, 1995)
U.S. DOE Risk Assessment Workshop (Augusta, GA, 1996)
MELCOR Accident Computer Code System (MACCS) 2 Computer Code, 1997, 2005
MCNPX Training Class (ANS Meeting, 1999)

CLEARANCE:

Active DOE "Q"

PROFESSIONAL EXPERIENCE:

Washington Safety Management Solutions **1997 to Present**
Advisory Engineer and Senior Fellow Advisor

Dr. O'Kula is a member of the State-of-the-Art Reactor Consequence Analysis (SOARCA) Project Peer Review Committee that provides recommendations on applying MACCS2 in the context of accident phe-

nomena and subsequent off-site consequences in the context of severe reactor accidents. This activity supports the efforts of Sandia National Laboratories (SNL) and the Nuclear Regulatory Commission (NRC) to provide more realistic assessment of severe accidents.

Dr. O'Kula is also part of the Level 3 PRA Standard working group charged with developing an ANSI/ANS standard for Level 3 PRA analysis. He participated in a team that conducted an SQA gap analysis on the bioassay code [Integrated Modules for Bioassay Analysis (IMBA)] based on DOE G 414.1-4 requirements. He identified safety analysis codes that were designated as DOE "toolbox" codes, and oversaw production of the first documents (QA criteria and application plan, code guidance reports, and gap analysis) for six accident analysis codes designated for the DOE Safety Software Toolbox. He provided support to DOE/EH-31 (now DOE/HSS) for addressing SQA issues for safety analysis software. He was a contributor to DOE G 414.1-4, *Safety Software Guide* on SQA practices, procedures, and programs.

Kevin has provided technical input for work packages on several recent commercial projects. In the first, he teamed with Entergy on MACCS2 code applications issues in the Severe Accident Mitigation Alternatives (SAMA) analysis area for the Pilgrim Nuclear Power Station. In the second, he was part of tritium environmental release analysis team that supported evaluation of tritium control and management areas for the Braidwood plant. A third effort developed an initial SAMDA document for the Mitsubishi Heavy Industries (MHI) US-APWR (1610 MW_e evolutionary PWR), as well as complete a control room habitability study for postulated toxic chemical gas releases.

Kevin was part of a Washington Group team that developed a Design Control Document (DCD) for the MHI US-APWR using input information from MHI. He was Chapter lead on Chapter 15 (Transient and Accident Analysis), and later transitioned to severe accident evaluation and documentation support to Chapter 19 (PRA and Severe Accidents). He currently is the Chapter 19 lead for PRA and Severe Accident for COLA development for the Pebble Bed Modular Reactor (PBMR).
Dr. O'Kula developed the outline, coordinated contributors, and assembled the first draft of the DOE *Accident Analysis Guidebook*, a reference guide for hazard, accident, and risk analysis of nuclear and chemical facilities operated in the DOE Complex. He is also the primary author and coordinator for the *Accident Analysis Application Guide* for the Oak Ridge contractor. Dr. O'Kula also developed a one-day course and exam for the guide, which he later presented to the Oak Ridge, Paducah, and Portsmouth staff. Dr. O'Kula also led an independent V&V review for the DTRA of the U.K.-developed Urban Dispersion Model (UDM) software for predicting chemical and biological plume dispersion in city environments, and is leading projects to verify and validate chemical/biological simulation suite software applications for the Dugway Proving Ground (Utah), and the Edgewood Chemical Biological Center (ECBC) in Maryland.

Managing Member, Consequence Analysis

Dr. O'Kula was responsible for the consequence analysis associated with accident analysis sections of Documented Safety Analysis (DSA) reports and other safety basis documents for SRS, Oak Ridge, and other DOE nuclear facilities. He also developed the methodology and identified appropriate computer models for this purpose. Additionally, Dr. O'Kula developed training to enhance consistency and standardize analyses in the consequence analysis area. He was project manager for environmental assessment support to SRS on a transportation safety analysis using the RADTRAN code.
Dr. O'Kula coordinated development of a DOE Accident Analysis Guidebook involving over 10 sites and organizations. He also led the effort to produce Computer Model Recommendations for source term (fire, spill, and explosion), in-facility transport, and dispersion/consequence (radiological and chemical) areas.

Westinghouse Savannah River Company **1989 to 1997**
Group Manager

Dr. O'Kula managed consequence analyses associated with accident analysis sections of DSA reports and other safety basis documents. He also developed the associated methodologies and identified appropriate computer models. He was a member of the management team supporting Criticality Safety Evaluation preparation assisting Safe Sites of Colorado and the dispositioning of final criticality safety issues for the decommissioning and decontamination of nuclear facilities at the Rocky Flats Environmental Technology Site.

In a teaming arrangement with Science Applications International Corporation, Kevin initiated discussions that led to development of an emergency management enhancement tool to risk inform likely source terms. Applied this approach to a Savannah River nuclear facility (K Reactor), and was part of the team to provide this methodology for use on the British Advanced Gas-Cooled Reactors (AGRs) (for the United Kingdom's Nuclear Installation Inspectorate). Model was knowledge-based, and required development of an Accident Progression Event Tree (APET) for the facility in question.

Dr. O'Kula managed the completion of the SRS K Reactor PRA program. He was the lead for development of the K Reactor Source Term Predictor Model and assisted with the core technology lay-up program to preserve competencies in reactor safety. He coordinated a 25-person group responsible for K Reactor probabilistic and deterministic dose analyses, and led the examination of reduced power cases at project termination. He developed risk and dose management applications to cost-effectively prioritize facility modifications.

Kevin interfaced with DOE Independent and Senior Review teams to finalize study acceptance, and transitioned the risk assessment team to risk management functions for nuclear and waste processing facilities. In addition, he successfully prepared a 10 CFR 100 Siting white paper to resolve issues raised by the DNFSB, and teamed with DOE/HQ legal support to document resolutions. He led the development of a position paper demonstrating SRS Replacement Tritium Facility compliance with DOE Safety Policy (SEN-35-91).

Staff Engineer

Dr. O'Kula led an analytical team quantifying the tritium source term during a Loss of River Water design basis accident. He evaluated airborne tritium levels with multi-cell CONTAIN model, interfaced with a multidisciplinary team to resolve Operational Readiness Review concerns, developed an SRS-specific methodology for applying MACCS as a tool for Level 3 PRA Applications, and applied CONTAIN code for K Reactor source term analysis.

E.I. du Pont de Nemours & Company **1982 to 1989**
Principal Engineer, Research Engineer

Dr. O'Kula performed risk analysis duties for the Savannah River Laboratory (SRL) Risk Analysis Group, after earlier conducting research activities for the Reactor Materials and Reactor Physics Groups. He performed initial planning for offsite irradiation of test specimens to evaluate remaining reactor lifetime for Savannah River reactor components.

Westinghouse Electric Corporation **1975**
Summer Student, Reactor Licensing
Monroeville, PA

American Electric Power Corporation **1973 to 1974**
Co-op Student, Reactor Physics and Reactor Licensing
New York, NY

Long Island Lighting Company **1972**
Summer Intern
Riverhead, NY

PARTIAL LIST OF PUBLICATIONS (2000-2010):

K. R. O'Kula, D. C. Thoman, J. Lowrie, and A. Keller, *Perspectives on DOE Consequence Inputs for Accident Analysis Applications*, American Nuclear Society 2008 Winter Meeting and Nuclear Technology Expo, November 9-13, 2008 (Reno, NV).

K. R. O'Kula, F. J. Mogolesko, K-J Hong, and P. A. Gaukler, *Severe Accident Mitigation Alternative Analysis Insights Using the MACCS2 Code*, American Nuclear Society 2008 Probabilistic Safety Assessment (PSA) Topical Meeting, September 7-11, 2008 (Knoxville, TN).

K. R. O'Kula and D. C. Thoman, *Modeling Atmospheric Releases of Tritium from Nuclear Installations*, American Nuclear Society Embedded Topical Meeting on the Safety and Technology of Nuclear Hydrogen Production, Control and Management, June 24-28, 2007 (Boston, MA).

K. R. O'Kula and D. C. Thoman, *Analytical Evaluation of Surface Roughness Length at a Large DOE Site (U)*, American Nuclear Society Winter Meeting, November 12-16, 2006 (Albuquerque, NM).

K. R. O'Kula and D. Sparkman, *Safety Software Guide Perspectives for the Design of New Nuclear Facilities (U)*, Winter Meeting of the American Nuclear Society, November 13 – 17, 2005 (Washington, D.C.).

K. R. O'Kula and R. Lagdon, *Progress in Addressing DNFSB Recommendation 2002-1 Issues: Improving Accident Analysis Software Applications*, Fifteenth Annual Energy Facility Contractors Group Safety Analysis Workshop, April 30 – May 5, 2005, Los Alamos, NM (2005).

K. R. O'Kula and Tony Eng, *A "Toolbox" Equivalent Process for Safety Analysis Software*, Fourteenth Annual Energy Facility Contractors Group Safety Analysis Workshop, May 1-6, 2004, Pleasanton, CA (2004).

K. R. O'Kula, D. C. Thoman, J. A. Spear, R. L. Geddes, *Assessing Consequences Due to Hypothetical Accident Releases from New Plutonium Facilities (U)*, American Nuclear Society Embedded Topical Meeting on Operating Nuclear Facility Safety, November 14 – 18, 2004 (Washington, D.C.).

K. O'Kula and J. Hansen, *Implementation of Methodology for Final Hazard Categorization of a DOE Nuclear Facility (U)*, Annual Meeting of the American Nuclear Society, June 13-17, 2004, (Pittsburgh, PA).

K. R. O'Kula and Tony Eng, *A "Toolbox" Equivalent Process for Safety Analysis Software*, Fourteenth Annual Energy Facility Contractors Group Safety Analysis Workshop, May 1-6, 2004, Pleasanton, CA (2004).

K. R. O'Kula, et al., *Evaluation of Current Computer Models Applied in the DOE Complex for SAR Analysis of Radiological Dispersion & Consequences*, WSRC-TR-96-0126, Westinghouse Savannah River Company (2003).

K. R. O'Kula, et al., *Evaluation of Current Computer Models Applied in the DOE Complex for SAR Analysis of Radiological Dispersion & Consequences*, WSRC-TR-96-0126, Rev. 3, Westinghouse Savannah River Company (2002).

K. R. O'Kula, *A DOE Computer Code Toolbox: Issues and Opportunities*, Eleventh Annual EFCOG Workshop, also 2001 Annual Meeting of the American Nuclear Society, Milwaukee, WI (2001).

PUBLICATIONS (1988-1999):

Dr. O'Kula authored or co-authored more than 20 publications between 1988 and 1999. Details are available upon request.

PROFESSIONAL SOCIETIES AND STANDARDS COMMITTEES

- American Nuclear Society
- Health Physics Society
- Level 3 ANS PRA Standard Committee 58.2

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