Exhibit No. ENT000014
Pilgrim LR Proceeding
50-293-LR, 06-848-02-LR

ENT –Rebuttal Source Term Testimony

February 1, 2011

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION,

Before the Atomic Safety and Licensing Board Panel

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

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Q1: Dr. O'Kula, have you reviewed the claims made at pages 43-44 of "Pilgrim Watch SAMA Remand Prefiled Testimony" ("PW Statement" or "Statement") regarding the source term used in the Pilgrim SAMA analysis and Exhibit PWA00012, Dr. Lyman's paper, on which Pilgrim Watch bases its source term claims?

A1. (KRO) Yes.

Q2: At page 43 of its Statement, Pilgrim Watch asserts that the MAAP code should not be used because it has not been validated by the NRC. Do you agree with this assertion?

A2. (KRO) No, I disagree. The Modular Accident Analysis Program (MAAP) code is a state-of-the-art code developed by the Electric Power Research Institute (EPRI) for use by utilities to quantify accident progression and source terms in the plant-specific Individual Plant Examinations (IPEs) and Probability Safety Assessments (PSAs). The use of MAAP in IPE and PSA applications has been accepted for regulatory use by the NRC Staff for many years.[1] MAAP is used throughout the world and, as discussed below, produces results comparable to MELCOR, a similar code developed by Sandia National Laboratories ("Sandia") for the Nuclear Regulatory Commission (NRC) in the modeling of severe accidents.

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[1] For example, the Staff accepted the use of MAAP in its 1994 design certification approval for the ABWR in NUREG-1503, FSER Related to Certification of the ABWR Standard Design (1994), at 19-55 and has done so for other subsequent design certification approvals, e.g., NUREG-1793, FSER Related to Certification of the AP1000 Standard Design (2004), at 19-61.

Q3: At page 43 of the PW Statement, Pilgrim Watch states that that the MAAP code release fractions are consistently smaller for key radionuclides than the release fractions specified in NUREG-1465. Please explain this assertion.

A3. (KRO) MAAP produces results that are different from, and consistently smaller than, the release fractions specified in NUREG-1465[2] because MAAP is used in these cases to model different phenomena. MAAP models the release of radionuclides from the containment into the environment following a postulated severe accident. In contrast, the NUREG-1465 source term solely represents radionuclides released into the containment, and is often used for design basis accident evaluation in plant safety analyses and in meeting NRC safety and licensing requirements. The NUREG-1465 source term does not specify the source term released from containment into the environment following a severe accident and does not take into account the reductions of the source term that would occur in those circumstances.

Q4: Please elaborate on the different functions of NUREG-1465 and MAAP and why that explains the difference in the NUREG 1465 source term and MAAP source term.

A4. First of all, it should be recognized that the reactor accident source term, the focus of this set of questions, is used generally in two distinct ways in the U.S. regulatory process. The first purpose is for licensing, safety analysis, and regulatory compliance, in particular meeting the siting requirements of 10 CFR Part 100. For this purpose, a source term representing the release of radioactive materials into the reactor containment is used to assess the adequacy of reactor containments and engineered safety systems as well as the environmental qualification of equipment inside the containment that must function following a design-basis accident. This source term is also used to show that dose criteria at the exclusion area boundary are met by assuming the maximum allowable design leak rate from the containment.[3] The NUREG-1465 source term is applicable for this first purpose.

The second purpose for which a reactor accident source term is developed is to simulate a release of radioactive material to the environment following a hypothetical

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[2] Accident Source Terms for Light-Water Nuclear Power Plants, Final Report, NUREG-1465 (Feb. 1995).

[3] See 10 C.F.R. § 50.34(a)(1)(ii)(D) and 10 C.F.R. § 100.11.

reactor accident. This second source term is input to models of radionuclide dispersal and accident consequences that are used for Level 3 PSAs and severe accident mitigation alternatives ("SAMA") cost-benefit analysis for which best estimate analyses are used, among other purposes. The use of the MAAP-based source term for the Pilgrim Station PSA and SAMA analysis supports this latter purpose, i.e., as a crucial element of Level 3 PSA and SAMA cost-benefit analyses.

The distinct phenomenological bases and regulatory purposes of the two source terms explain their numerical differences in the amount of radionuclides and the timing for the release. As a result of containment engineered safety features (ESFs) (e.g., containment sprays, filters, and coolers) and natural depletion processes (e.g., aerosol deposition and containment holdup), the source term released from the reactor coolant system into containment should be different from the source term released from containment into the environment. Hence, the NUREG-1465 and MAAP source terms should differ, with the MAAP source term being the smaller of the two.

Indeed, NUREG-1465 discusses fission product removal mechanisms such as ESFs, but NUREG-1465 does not specify the final quantitative effects of these features. Comparatively, the MAAP code does credit these ESFs as fission product removal mechanisms. Thus, there would be obvious differences between a source term into the containment and that of a source term from the containment into the environment.

Q5: At pages 43-44 of its Statement, Pilgrim Watch refers to two comparisons involving MAAP for which Pilgrim Watch claims that the environmental release fractions estimated by MAAP were smaller than those estimated using the codes that were used for NUREG-1150 study. Would you please comment on this assertion?

A5: (KRO) Yes. Pilgrim Watch refers to a February 1987 draft of the NUREG-1150 study for a comparison for the Zion plant (a four-loop PWR) between MAAP and the Source Term Code Package (STCP), which was the primary code used in the NUREG-1150 study.[4] Pilgrim Watch also refers to a comparison of the PRA performed in the mid-1990s using MAAP for Duke Power's Catawba Pressurized

[4] It should be noted that the Zion comparison referred to by Pilgrim Watch is not contained in the final NUREG-1150 study.

Water Reactor (4-loop PWR plant with ice condenser containment) with a typical NUREG-1150 release based on the Sequoyah PWR plant analyzed using STCP and MELCOR. Neither comparison is directly applicable to the use of MAAP for Pilgrim (a BWR plant) because of the differences between severe accident phenomenology in BWR and PWR plants.

Furthermore, neither comparison is relevant because of the significant advancements that have been made in the modeling of severe accidents since the NUREG-1150 study was performed in the 1980s. For example, the comparison between the Catawba PRA and the NUREG-1150 Sequoyah release represents a difference of more than ten years in terms of severe accident modeling. The comparison uses a release category that represents an early containment failure with large unmitigated releases. Since the issuance of NUREG-1150, modeling improvements and experimental results have shown that the consequences of a typical large early, unscrubbed release are smaller than previously thought.

Thus, a comparison of MAAP-based source terms with those estimated over ten years previously with STCP (a simpler code) and an earlier version of MELCOR should show differences. A more technically correct and meaningful comparison is to compare severe accident models at a similar point in development.

Q6: Have such comparisons involving MAAP and MELCOR been performed?

A6. (KRO) Yes, comparisons have been done between MAAP and MELCOR, which is a similar code for modeling severe accident progression developed by Sandia for the NRC. Both MAAP and MELCOR are now used throughout the world, and are much more advanced than predecessor versions or simpler models such as STCP that were applied over twenty years ago in the NUREG-1150 study. They are both integrated codes that allow the calculation of accident sequence progression from the initiating event while taking into account important inter-related phenomena (e.g., reactor coolant system, containment thermal-hydraulics, in-vessel core degradation, molten core concrete interaction, fission product release and transport into the environment).

A 2004 comparison using MELCOR and MAAP for a PWR accident sequence demonstrates that the two codes provide similar calculated results for thermal-hydraulic and core degradation response of the plant, with minor differences in various timings of phenomena. The authors indicated that these minor differences in results were within the uncertainties of the code numerical computations and the physics models.[5]

Q7: Are you aware of instances where similar source term claims in other license renewal proceedings were raised?

A7. Yes. In the license renewal proceeding for Entergy's Indian Point Nuclear Generating Units 2 and 3, the petitioner, Riverkeeper, in its contention EC-2 made a nearly identical assertion to what Pilgrim Watch proffered in its Statement. The petitioner alleged that Entergy's estimate of accident consequences were too low by a factor of three or more, primarily because Entergy used an unusually low source term, among other reasons. Based on the same claims as advanced in PWA00012 relied upon by Pilgrim Watch here, Riverkeeper claimed that Entergy should have used the NUREG-1465 source term rather than a source term derived from the MAAP application.

Q8: How did the Atomic Safety and Licensing Board in the Indian Point proceeding rule on this claim?

A8. Based on the documents I reviewed, including the Licensing Board's July 31, 2008 order, the Licensing Board rejected the contention based on the NRC Staff's explanation that the NUREG-1465 source term methodology addresses only releases into containment and assumes that containment remains intact but leaks. Therefore, the methodology is not appropriate for analyzing an early energetic containment breach in support of a PSA and SAMA analysis. I have reviewed the relevant Staff's pleadings referred to by the Board and agree with the statements and position of the Staff. Based on the explanation provided by the Staff, the Board rejected the application of the NUREG-1465 methodology for SAMA analyses.

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[5] K. Vierow, Y.Liao, J.Johnson, M.Kenton, and R.Gauntt, "Severe accident analysis of a PWR station blackout with the MELCOR, MAAP4, and SCDAP/RELAP5 codes, Nuclear Engineering and Design 234, 129-145, (2004).

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AFFIDAVIT OF KEVIN R. O'KULA

I, Kevin R. O'Kula, do hereby declare under penalty of perjury that my statements in the foregoing testimony are true and correct to the best of my knowledge and belief.

Executed in Accord with 10 CFR 2.304(d)

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