June 6, 2011

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR GENERATION	)	
COMPANY AND ENTERGY NUCLEAR	)	Docket No. 50-293-LR
OPERATIONS, INC.	)	
	)	
(Pilgrim Nuclear Power Station)	)	

AFFIDAVIT OF DR. NATHAN E. BIXLER AND DR. S. TINA GHOSH IN SUPPORT OF THE NRC STAFF'S ANSWER IN OPPOSITION TO PILGRIM WATCH'S REQUEST FOR HEARING ON POST FUKUSHIMA SAMA CONTENTION

Nathan E. Bixler and S. Tina Ghosh, do hereby state as follows:

1. [NEB] I am a Principal Member of the Technical Staff in Sandia National Laboratory's ("SNL") Analysis and Modeling Department and have been at SNL for 29 years. During the past 19 years, my work has included projects for the NRC. I have a Ph. D. in Chemical Engineering and have been primarily involved in computer modeling of fluid dynamics and more recently in simulation of nuclear accidents and consequences. I have led the development and application efforts on a variety of NRC codes, including VICTORIA, RADTRAD, MACCS2, MELMACCS, and SECPOP2000. I am the SNL project manager for development and application of the WinMACCS code suite,[1] which the NRC Staff uses in performing consequence analysis for level-3 Probabilistic Risk Assessments ("PRAs"). I am the primary instructor for a one-week class offered to NRC staff on Level-3 PRA (the P-301 course in the NRC curriculum). I am also currently working on consequence analyses for safety documentation of the Mars Science Laboratory Mission scheduled for 2011. I serve as the SNL consequence analysis technical lead for the State-of-the-Art Reactor Consequence Analyses

[1] The WinMACCS code suite is the MACCS2 code with a graphical, window-based user interface.

project. A statement of my professional qualifications is available in the Agency-wide Document Access and Management System ("ADAMS") Accession No. ML071800287.

2. [STG] I am a senior program manager employed by the U.S. Nuclear Regulatory Commission (NRC). I have been employed by the NRC for over six years. My current primary responsibility is to be the NRC lead for the State of the Art Reactor Consequence Analysis's ("SOARCA") uncertainty analysis. In my previous position as a reactor engineer in the Office of Nuclear Reactor Regulation's (NRR) Division of Risk Assessment, one of my primary responsibilities was to review SAMA analyses submitted in support of nuclear power plant license renewal applications, and write the corresponding portions of the NRC's supplemental environmental impact statements. I also reviewed risk-informed licensing applications that used level 2 and level 3 PRA results (i.e., analyses of accidents that involve potential radioactive releases outside the reactor containment). In my first position at the NRC in the Division of High-Level Waste Repository Safety in the Office of Nuclear Material Safety and Safeguards, my primary responsibility was to review different aspects of the Department of Energy's total-system performance assessment ("TSPA") and preclosure safety analysis ("PCSA") for the Yucca Mountain repository license application. The TSPA is analogous to a level 3 PRA applied to a geologic waste disposal system, and the PCSA is analogous to a PRA for the waste-handling facilities in the operational phase in the Yucca Mountain license application. For my doctoral thesis at the Massachusetts Institute of Technology, I developed a sensitivity analysis method to generate risk information that would be useful for making decisions about high-level nuclear waste repositories given the uncertainty in the risk analyses. I demonstrated the application of the method using the proposed Yucca Mountain repository as an example, and subsequently published a paper on the method in the journal, Nuclear Technology. My statement of qualifications is available at ADAMS Accession No. ML110030972.

3. This affidavit responds to two new contentions raised by Pilgrim Watch ("PW") that are claimed to be related to the Fukushima Daiichi event in Japan, which are as follows:

> 1. The [MACCS2] code limits the total duration of a radioactive release to no more than four (4) days, if the Applicant chooses to use four plumes occurring sequentially over a four day period. Entergy chose not to take that option and limited its analysis to a single plume having a total duration of the maximum-allowed 24 hours. In any case either a 24-hour plume or a four-day plume is insufficient duration in light of lessons learned from Fukushima. The Fukushima crisis now stretches into its second month and shows that releases can extend into many days, weeks, and months; a longer release can cause offsite consequences that will affect cost-benefit analyses.
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> 2. Computer codes in use are totally incapable of modeling an 8-week chain reaction that continues after a scram. MACCS2 is no exception. Like all the computer codes, it is incapable of modeling a "severe accident" release that lasts 8 weeks or longer. The MACCS2 code used by Entergy, and all other codes, assumes that the reactor is scrammed when the accident begins, the reactor is scrammed, and that the production of all fission products ceases at that time.
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> We know that criticality was continuing at Fukushima Unit 2 through April 27, 2011, and to shorter duration at Unit 1, because of their continued post-scram high findings of I-131 reported by TEPCO. The reactors were shut down, scrammed, on March 11.th I-131 has an 8-day half-life. If criticality had stopped after the reactors scrammed, the I-131 would have largely decayed. It would not, be at the levels we have seen reported, that exceed the Cesium readings.
>
> Conventional accident analysis of reactor accidents begin at reactor scram, t=0, and assume that the fission chain reaction ceases completely at that time, and that thereafter there is only "spontaneous" nuclear decay, with it being common practice to ignore the very tiny amount of "spontaneous fission" triggered by random neutrons from cosmic radiation hitting a fissile atom and creating infinitesimal amounts of I-131.
>
> A large problem created by the ongoing chain reaction is the calculation of food doses. The code has no way of modeling the continual production of I-131 and I-134 which can get to people both by milk and from fresh leafy-vegetable consumption.

4. The concerns raised by PW would not alter the conclusions of Entergy's Severe Accident Mitigation Alternatives ("SAMA") analysis for Pilgrim Nuclear Power Station ("Pilgrim") and are unsupported because the asserted facts are speculative and more readily explained by our current accident modeling. Based on our review of PW's Motion, the Affidavit of David

Chanin, and our review of the anonymously authored paper attached as an exhibit to PW's Motion, we conclude that the issues raised in PW's Motion do not raise a safety significant issue within the scope of this license renewal proceeding.

5. PW has not demonstrated how the issues it alleges in its Motion were not already captured by Pilgrim's probabilistic risk assessment ("PRA") and SAMA analysis which considers station blackout scenarios (including those initiated by seismic events) and where five out of the seven potentially cost-beneficial SAMAs identified in the FSEIS are mitigation measures for loss-of-power scenarios. As our previous testimony[2] demonstrated, it would require at least a doubling of benefits before the next SAMA on the candidate list could become potentially cost-beneficial (hence at a minimum a doubling of benefits is required to change the results of the SAMA analysis).

6. PW alleges that re-criticality occurred based on an anonymously authored article with an unknown provenance. Contrary to PW's assertions, re-criticality is unlikely and the assertions supporting PW's claim of re-criticality are more simply explained from the known and well studied methods for iodine and cesium behavior during an accident, as more fully explained in paragraphs 11—18, below. Even in the unlikely event that re-criticality occurred at the Fukushima plant, the impact of this re-criticality would be negligible on the SAMA analysis and would not change the results of the Pilgrim SAMA analysis.

7. Finally, PW does not demonstrate that on-going releases (beyond the initial release) at Fukushima are consequential from the perspective of a SAMA analysis, compared to modeling a 24-hour release as was done in the Pilgrim SAMA analysis.

8. As noted in Prior Testimony A 7 at 6, a SAMA analysis is a systematic search for potentially cost beneficial enhancements *to further reduce* nuclear power plant accident risk. A

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[2] NRC Staff Testimony of Nathan E. Bixler and S. Tina Ghosh Concerning the Impact of Alternative Meteorological Models on the Severe Accident Mitigation Alternatives Analysis (Jan. 3, 2011) (ADAMS Accession No. ML110030966) ("Prior Testimony")

SAMA analysis allows for the comparison of benefits derived from particular mitigation measures with their cost to implement. Thus a SAMA analysis is a cost-benefit analysis. It is not a safety analysis. Furthermore, Pilgrim had previously satisfied the NRC's Generic Letter 88-20 requirements through completion of an individual plant examination (IPE) and individual plant examination for external events (IPEEE) to search for any plant vulnerabilities. Any remaining potentially cost-beneficial enhancements uncovered in the SAMA analysis performed for license renewal could only expect to *further reduce* the risk from a plant that had no identified safety vulnerabilities. Therefore, potential shortcomings in the SAMA analysis cannot be safety significant, both by definition and by practice at Pilgrim.

9. PW implicitly assumes that the same type of accident that occurred at Fukushima could also occur at Pilgrim. It is important to note that these types of hazards are site-specific, and the effects of these hazards are plant-specific. The issue is not whether Pilgrim's SAMA analysis accounted or could account for the events that occurred at the Fukushima Daiichi plant but whether it accounts for the hazards present at the Pilgrim site. The Pilgrim SAMA analysis relies on site-specific hazard analyses that considered all potential challenges to the plant from environmental hazards such as seismic events and tsunami events. Those hazards that are found to be credible challenges at Pilgrim are modeled further, including plant-specific response to the challenges. The Pilgrim SAMA analysis accounts for seismic events and already captures the remote possibility that a seismic event could affect the plant and induce a station blackout. In fact, as noted in Entergy's Environmental Report (ER), station blackout scenarios account for ~2% of the core damage frequency (which attests to the low frequency of these challenges) and ~57% of the large-early release frequency which contributes to off-site risk; and loss-of-power scenarios account for ~82% of core damage frequency. Furthermore, the Final Supplemental Environmental Impact Statement ("FSEIS") indicates that five of the seven potentially cost-beneficial SAMAs identified in the ER and as a result of the NRC's SAMA review mitigate the loss-of-power scenarios (SAMAs 30, 34, 57, 58, and unnumbered SAMA on use of a portable

generator to power the battery chargers identified in response to NRC staff inquiry), of which station blackout is a subset. To account for the benefit of reducing risk from external events (of which seismic is the main contributor), Entergy multiplied the calculated benefits from internal events by a factor of 5 (see FSEIS section G.6.2). In addition, as noted in the FSEIS, plant improvements identified to mitigate potential seismic events were already implemented at Pilgrim as part of the Unresolved Safety Issue A-46 resolution and IPEEE programs (see FSEIS section G.2.2). These improvements included structural modifications to the station blackout diesel and the main transformer. Thus, Entergy's SAMA analysis and NRC's review identified past plant improvements that were already implemented at Pilgrim to mitigate potential seismic and station blackout events, and identified several potentially cost-beneficial SAMAs that would mitigate loss-of-power scenarios.

10. In order to affect the results of the SAMA analysis, we would have to postulate identification of additional potentially cost-beneficial SAMAs. As noted in Prior Testimony A 42 at 23, the cost of the next most cost-beneficial SAMA is more than a factor of two greater than its benefit (this factor of two would be above and beyond the multiplicative factor of 5 that accounts for the potential benefits from external events, as noted in paragraph 5 above). Hence the benefit would have to double, at a minimum, to affect of the results of the SAMA analysis. For the reasons we offer below, it is highly unlikely that the two concerns raised by PW can account for a doubling of benefits in the SAMA analysis, and hence highly unlikely to affect the results of the SAMA analysis.

11. The second item in PW's first set of Fukushima-related proposed contentions states that "[c]omputer codes in use are totally incapable of modeling an 8-week chain reaction that continues after a scram." This statement is based on PW's conclusion that a nuclear chain reaction continued at one or more of the Fukushima Daiichi units for an extended period of time, roughly 8 weeks. The basis for this conclusion is highly speculative and likely incorrect. The only evidence upon which PW bases this conclusion is the ratio of ^{131}I to ^{134}Cs and ^{137}Cs in the

groundwater collected in the sub-drain at Unit 2, where levels of ^{131}I appear to hold steady over time rather than diminishing. PW notes that ^{131}I only has an 8-day half life while ^{137}Cs has a 30-year half life, so one would expect the ratio of ^{131}I to ^{137}Cs to diminish over time. PW's anonymous article appears to make several incorrect assumptions and we believe that alternative assumptions would more readily and simply explain the ratios of ^{131}I to ^{134}Cs and ^{137}Cs. The article's incorrect assumptions include that (1) the initial activities of ^{131}I, ^{134}Cs, and ^{137}Cs in the reactor core are about the same at reactor shutdown (p. 13 of PW contention); (2) releases of iodine and cesium from Unit 2 throughout the accident are in proportion to the core inventories; and (3) the only explanation for ^{131}I activity in the sub-drain water staying at or above the activities of ^{134}Cs and ^{137}Cs is that ^{131}I is continually being produced as a result of an ongoing nuclear chain reaction. Each of these incorrect assumptions is discussed in turn, below.

12. PW's analysis of activities is based on an incorrect assumption stated on page 13 of the proposed contention, "At the time of the scram (t-0) the Bq of I-131 and Cs-134 and Cs-137 would all have been approximately equal:…". In a typical BWR at the middle of fuel cycle, there would be about $3x10^{18}$ bequerels of ^{131}I, $4x10^{17}$ bequerels of ^{134}Cs, and $4x10^{17}$ of ^{137}Cs. Thus, there is almost 10 times more ^{131}I than there is of either of the cesium isotopes mentioned in the PW contention. Because the rate of buildup of these isotopes differs during normal operation of the reactor, this ratio would be even higher early in the fuel cycle and would diminish to a factor of about 5 at end of fuel cycle (just before refueling). There is no point during the fuel cycle that the activity of ^{131}I would be roughly equal to the activities of the cesium isotopes.

13. Following reactor shutdown, these isotopes would decay according to their half lives, which are 8 days for ^{131}I, 2 years for ^{134}Cs, and 30 years for ^{137}Cs. Because the half lives of the two cesium isotopes are much longer than the timeframe discussed in the PW contention, their activities would diminish only a little. On the other hand, the activity of ^{131}I would decrease by half over each 8-day period. That means that at 16 days after reactor shutdown, the activity

of ^{131}I in the reactor would still be about twice that of the cesium isotopes; at 32 days after reactor shutdown, the activity of ^{131}I in the reactor would be about half that of the cesium isotopes; at 48 days after reactor shutdown, the activity of ^{131}I in the reactor would be about one tenth that of the cesium isotopes. It is important to understand that these ratios only apply under the assumption that ratios of iodine and cesium utilize the entire core inventory or that the source being measured is proportional to the iodine to cesium ratios found in the core.

14. The plots of sub-drain activities provided in the PW contention extend to 4/27, which is 47 days after the accident began on 3/11/2011. If the releases of the iodine and cesium isotopes from the containment were in proportion to their activities in the reactor core, then we would expect the ^{131}I would to be about one tenth that of the cesium isotopes on 4/27. But, iodine can be released in a variety of chemical forms, of which only one is bound with cesium. The current models predict that much of the late release of iodine from the containment is in the form of molecular iodine (I_2), which tends to evolve from the iodine dissolved in the aqueous solution in the wet well of the containment. Molecular iodine is very volatile. This increased volatility over forms of cesium results in more efficient transport of the iodine into the environment than the less volatile cesium. Transport of molecular iodine from the containment increases the level of iodine contamination in the vicinity of the plant, but does not increase the level of cesium contamination.

15. The likely explanation for the larger activity of ^{131}I compared with ^{134}Cs and ^{137}Cs observed in the groundwater collected in the sub-drain at Unit 2 (c.f. the first figure on p. 11 of the PW contention) is that greater quantities of iodine continued to be released into the groundwater than of cesium due to the more efficient transport mechanisms for iodine. This explanation agrees very well with our current understanding of how molecular iodine evolves from the aqueous solution in the wet well over an extended period. On the other hand, there is little continuing release of cesium from the wet well because cesium tends to remain dissolved in the aqueous solution.

16. The activities represented in the samples drawn from the groundwater discussed in the PW contention represent an extremely tiny fraction of the remaining ^{131}I activity in the reactor. The observed groundwater activities can easily be explained by continuing evolution of iodine from the containment. Thus, there is no reason to believe that a nuclear chain reaction was required to produce the ^{131}I found in these samples. As stated above, it is far more likely that continuing evolution of molecular iodine from the wet well caused the elevated levels of ^{131}I in the groundwater samples.

17. Even if re-criticality were to occur, it would not have a material effect on the SAMA analysis. Achieving sustained critical reaction in a light-water reactor core of US design requires: (1) favorable geometry and (2) sufficient moderator (water). During a severe accident, when the core materials are melted and geometry is lost, it is not easy to achieve good conditions for re-criticality. Control rod materials (poisons) will be part of the fuel melt too, and sufficient water and the right configuration (geometry) must be present to sustain a chain reaction. Although re-criticality might occur in very small isolated pockets of slumped (melted) fuel where sufficient water is present. Such conditions, if possible, would occur in only small localized regions, for short periods of time. It would be nothing like producing 100% power from the entire core.

18. The net effect of re-criticality (if it occurred) would be to slightly change the source terms for a small subset of accidents in the SAMA analysis. For these accidents, the change in source term would be a small fraction of the total source term (e.g., small increase in short-lived isotopes such as ^{131}I later in time from the start of the accident). The subset of accidents that might be affected is also limited to a small fraction. Hence the net effect on source term is expected to be a small fraction of a small fraction, resulting in no appreciable change in the SAMA results (which we previously noted would require at least a doubling of benefits before the next SAMA on the list could become potentially cost-beneficial – see paragraph 9 above).

19. PW has not demonstrated that modeling a 24-hour release is inadequate for the purpose of the Pilgrim SAMA analysis. Most of the radioactivity released into the atmosphere at Fukushima Daiichi occurred early in the accident, within the first week. Since that time, offsite atmospheric and ground contamination levels have been decreasing. PW does not provide evidence that any continued small releases from the Fukushima accident could be consequential, i.e., large enough to affect the results of the Pilgrim SAMA analysis.

20. Finally, PW asserts that no computer codes exist that are capable of modeling releases over an 8 week period. We would agree that there is no computer code capable of modeling severe accidents for a SAMA analysis that is currently capable of modeling an extended but slow release over 8 weeks.[3]

Executed in Accord with 10 CFR 2.304(d).
Nathan E. Bixler
Principal Member of the Technical Staff
Sandia National Laboratories
PO Box 5800
Albuquerque, NM 87185-0748
505-845-3144
nbixler@sandia.gov

Executed in Accord with 10 CFR 2.304(d)
S. Tina Ghosh
Senior Program Manager
U.S. Nuclear Regulatory Commission
Mail Stop: C-3A07M
Washington, DC 20555
Phone: 301-251-7984
Tina.Ghosh@nrc.gov

[3] A number of codes that are not appropriate for SAMA analysis can model extended releases. These models are more appropriate to emergency management than the type analysis necessary for an accurate SAMA analysis. Most importantly, these codes do not have the capability to model emergency responses or evaluate economic impacts.