August 12, 2008

Mr. Charles Pardee
Chief Nuclear Officer
AmerGen Energy Company, LLC
200 Exelon Way, KSA 3-N
Kennett Square, PA 19348

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION REGARDING GENERIC LETTER 2004-02, SUPPLEMENTAL RESPONSE (TAC NO. MC4724)

Dear Mr. Pardee:

By letter dated December 28, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML073620535), AmerGen Energy Company, LLC (AmerGen) submitted a supplemental response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors" for Three Mile Island, Unit 1 (TMI-1). The cognizant Nuclear Regulatory Commission (NRC) staff has reviewed your TMI-1 submittal. The review process involved a detailed evaluation by a team of 10 subject matter experts, with focus on the review areas described in the NRC's "Content Guide for Generic Letter 2004-02 Supplemental Responses" (ADAMS Accession No. ML073110389). The review process also included a separate "holistic" review of your submittal, that focused on whether you have demonstrated that your corrective actions for Generic Letter 2004-02 are adequate.

Based on these reviews, the NRC staff has concluded that additional information is needed to assess whether there is reasonable assurance that Generic Letter 2004-02 has been satisfactorily addressed at TMI-1. The specific information needed is found in the enclosed request for additional information (RAI). The RAI was sent, in draft form, via electronic transmission, on June 26, 2008, to Ms. Wendi Rapisarda of your staff. The draft RAI was sent to ensure that the specific requests were understandable, the regulatory basis was clear, and to determine if the information requested was previously docketed. The RAI was discussed with your staff in a teleconference on July 21, 2008.

The NRC requests that AmerGen respond to this RAI within 90 days of the date of this letter. However, the NRC wishes to only receive one response letter for the items listed in the enclosure except the last (number 30). If you conclude that more than 90 days is needed to respond to the RAI, you should request additional time, including a basis for why such time is needed.

If you conclude, based on its review of the RAI, that additional corrective actions for GL 2004-02 are needed, you should request an extension of time to complete such corrective actions if needed. Criteria for such extensions are contained in SECY-06-0078 (ADAMS Accession No. ML053620174), and many examples of previous requests and approvals can be found on the NRC's sump performance website,
http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance.html.

The exception to the above response timeline is the last RAI in the attachment (number 30). The NRC staff considers in-vessel downstream effects to not be fully addressed at TMI-1 as well as at other pressurized water reactors. Your submittal refers to draft WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid." The NRC staff has not issued a final safety evaluation (SE) for WCAP-16793-NP. You may demonstrate that in-vessel downstream effects issues are resolved for TMI-1 by showing that the TMI-1 plant conditions are bounded by the final WCAP-16793-NP and the corresponding final NRC staff SE, and by addressing any conditions and limitations in the final SE. You may also resolve this item by demonstrating without reference to WCAP-16793 or the staff SE, that in-vessel downstream effects have been addressed at TMI-1. In any event, you should report how you have addressed the in-vessel downstream effects issue within 90 days of issuance of the final NRC staff SE on WCAP-16793. The NRC staff is developing a Regulatory Issue Summary to inform the industry of the staff's expectations and plans regarding resolution of this remaining aspect of GSI-191.

Please contact me at 301-415-2833, if you have any questions.

Sincerely,

/ra/ (JLamb for)

Peter Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosure: As stated

cc w/encl: See next page

The exception to the above response timeline is the last RAI in the attachment (number 30). The NRC staff considers in-vessel downstream effects to not be fully addressed at TMI-1 as well as at other pressurized water reactors. Your submittal refers to draft WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid." The NRC staff has not issued a final safety evaluation (SE) for WCAP-16793-NP. You may demonstrate that in-vessel downstream effects issues are resolved for TMI-1 by showing that the TMI-1 plant conditions are bounded by the final WCAP-16793-NP and the corresponding final NRC staff SE, and by addressing any conditions and limitations in the final SE. You may also resolve this item by demonstrating without reference to WCAP-16793 or the staff SE, that in-vessel downstream effects have been addressed at TMI-1. In any event, you should report how you have addressed the in-vessel downstream effects issue within 90 days of issuance of the final NRC staff SE on WCAP-16793. The NRC staff is developing a Regulatory Issue Summary to inform the industry of the staff's expectations and plans regarding resolution of this remaining aspect of GSI-191.

Please contact me at 301-415-2833, if you have any questions.

Sincerely,

/ra/ (JLamb for)

Peter Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosure: As stated

cc w/encl: See next page

DISTRIBUTION:
PUBLIC
RidsAcrsAcnw_MailCTR Resource
RidsRgn1MailCenter Resource
LWhitney, NRR
RidsOgcRp Resource
RidsNrrPMPBamford Resource
RidsNrrDorlDpr Resource
RidsNrrLAABaxter Resource
LPLI-2 R/F
RidsNrrDssSsib Resource
RidsNrrDorlLpl1-2 Resouce

ADAMS Accession Number: ML082040755

OFFICE	**LPLI-2/PM**	**LPLI-2/LA**	**SSIB/BC**	**CSGB/BC**	**LPL1-2/BC**
NAME	**PBamford**	**ABaxter (w/comments)**	**MScott (w/comments)**	**AHiser**	**HChernoff (JLamb for)**
DATE	**7/28/08**	**7/25/08**	**7/29/08**	**8/5/08**	**8/12/08**

OFFICIAL RECORD COPY

cc:

Site Vice President - Three Mile Island Nuclear Station, Unit 1
AmerGen Energy Company, LLC
P. O. Box 480
Middletown, PA 17057

Senior Vice President - Operations, Mid-Atlantic
AmerGen Energy Company, LLC
200 Exelon Way, KSA 3-N
Kennett Square, PA 19348

Vice President - Licensing and Regulatory Affairs
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Courthouse
Harrisburg, PA 17120

Chairman
Board of Supervisors
of Londonderry Township
R.D. #1, Geyers Church Road
Middletown, PA 17057

Senior Resident Inspector (TMI-1)
U.S. Nuclear Regulatory Commission
P.O. Box 219
Middletown, PA 17057

Director - Licensing and Regulatory Affairs
AmerGen Energy Company, LLC
Correspondence Control
P.O. Box 160
Kennett Square, PA 19348

Director
Bureau of Radiation Protection
Pennsylvania Department of
Environmental Protection
Rachel Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469

Plant Manager - Three Mile Island Nuclear Station, Unit 1
AmerGen Energy Company, LLC
P. O. Box 480
Middletown, PA 17057

Regulatory Assurance Manager - Three Mile Island Nuclear Station, Unit 1
AmerGen Energy Company, LLC
P.O. Box 480
Middletown, PA 17057

Ronald Bellamy, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Ronnie Gardner, PMP
Manager, Site Operations and Corporate Regulatory Affairs
AREVA NP Inc.
3315 Old Forest Road OF-16
Lynchburg, VA 24501

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

Eric Epstein
TMI Alert
4100 Hillsdale Road
Harrisburg, PA 17112

Correspondence Control Desk
AmerGen Energy Company, LLC
200 Exelon Way, KSA 1-N-1
Kennett Square, PA 19348

Three Mile Island Nuclear Station, Unit 1

cc:

Manager Licensing - Three Mile Island Nuclear Station, Unit 1
Exelon Generation Company, LLC
Correspondence Control
P.O. Box 160
Kennett Square, PA 19348

Mr. Charles G. Pardee
Chief Nuclear Officer
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Associate General Counsel
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Chief Operating Officer (COO)
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Senior Vice President – Operations Support
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

REQUEST FOR ADDITIONAL INFORMATION

REGARDING SUPPLEMENTAL RESPONSE TO GENERIC LETTER 2004-02

THREE MILE ISLAND, UNIT 1

DOCKET NO. 50-289

On September 13, 2004, the Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," as part of the NRC's efforts to assess the possibility that the emergency core cooling system and containment spray system pumps at domestic pressurized water reactors (PWRs) would experience a debris-induced loss of net positive suction head margin during sump recirculation. By letters dated March 7, 2005, and September 1, 2005, and as supplemented by letters dated July 27, 2005, and January 31, 2006,[1] AmerGen Energy Company, LLC (AmerGen) provided a response to the GL for Three Mile Island, Unit 1 (TMI-1). By letter dated February 9, 2006,[2] the NRC requested additional information regarding the TMI-1 GL 2004-02 response. By letters dated March 3, 2006, March 28, 2006, and November 21, 2007,[3] guidance on GL supplemental responses was provided by the NRC staff.

By letter dated December 28, 2007,[4] AmerGen provided the supplemental response for TMI-1 to GL 2004-02. The NRC staff is reviewing and evaluating the supplement and has determined that responses to the following are necessary in order for the staff to complete its review.

1. Please describe the approach to the break selection process used (e.g., incrementing the break location along the potential high pressure lines) and explain how it is systematic and effective in bounding the amounts of debris generated from the various potential loss-of-coolant accident (LOCA) locations.

2. Please provide justification to support the characteristically smaller size distribution of destroyed fibrous insulation within the 7 diameter (7D) zone-of-influence (ZOI) for jacketed low-density fiberglass versus the size distribution which would exist for a larger ZOI. Include an explanation of how Table 2 (on page 8 of 65) of the GL supplemental response is consistent with the 7D ZOI assumed size distribution of 60 percent small fines and 40 percent large pieces.

3. Please provide the assumed size distribution for reflective metal insulation (RMI) debris.

4. Please provide the post-transport size distributions for the RMI, and jacketed and unjacketed Nukon insulation debris with justifications for the transport fractions (e.g., erosion effects).

---

1 Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML050670026, ML052450029, ML052140271, and ML060320725, respectively
2 ADAMS Accession No. ML060380153
3 ADAMS Accession No. ML060620050, ML060870274, and ML073110269, respectively
4 ADAMS Accession No. ML073620535

5. Please provide a detailed description of any methods or assumptions used by the transport evaluation vendor for the refined transport analysis as discussed in Section 3.e.2 of the GL supplemental response. This would specifically include any items that are not consistent with the approved guidance (Safety Evaluation by the Office of Nuclear Reactor Regulation related to NRC Generic Letter 2004-02, Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), ADAMS Accession No. ML043280007) for debris transport.

6. The graph of head loss vs. time in Section 3o.2.17.i of the GL supplemental response indicates that a large vortex occurred during testing. The vortex resulted in significantly decreased head loss that did not recover. However, in Section 3f.3 it is stated that no significant vortex formations occurred during testing with non-chemical debris. Please address whether the vortex was prototypical for strainer operations under LOCA conditions. Further, please address whether the vortex that occurred during testing potentially disturbed the debris bed. Such a disturbance could result in non-conservative head loss results or non-conservative temperature scaling (due to turbulent flow through a discontinuity in the debris bed).

7. The response to Section 3f.6 states that the Enercon strainers were shown to resist the formation of a thin debris bed by testing. It is not clear that the strainers were tested with prototypically fine fibrous debris, introduced conservatively with prototypical flow conditions to ensure that a thin bed would not occur in the plant. The debris generation section of the GL supplemental response gives a breakdown of large pieces and small-fines, but, for the purposes of a thin bed, fines are the important debris bed constituent. The debris characteristics section for a 17D Nukon ZOI in the GL supplemental response states that the size distribution is per the safety evaluation (SE) and has been reviewed by the NRC during audits. This is an insufficient rationale for debris sizing. Most NRC audits have identified issues relating to fibrous debris generation and debris size distribution. Please provide information to demonstrate that the fibrous debris sizes used for testing match the debris reaching the strainer as calculated in the debris transport calculation, differentiating between fine and small fibrous debris. Thin bed testing should be conducted with the finest debris predicted to reach the strainer. A specific reference to consider is staff review guidance, Enclosure 1 of a March 28, 2008, NRC letter to the Nuclear Energy Institute (ADAMS Accession No. ML080230038).

8. Section 3f.13 of the supplemental response describes how the test results were scaled for strainer approach velocity. Please provide the tested velocity and velocities to which the test results were scaled. In addition, please justify that the scaling was conservative or prototypical. For example, scaling to a higher velocity would not be considered conservative because of the potential for effects such as bed compression.

9. It appears that the final chemical effects test was run over a period of 2 days. The head loss trend appeared to be increasing slowly over the last 12 hours of the test. The response to Section 3o.2.17 says no extrapolation was conducted. Other long-term testing has shown a slow but significant increase in head loss over a matter of days after initial bed formation/head loss increases. This is particularly important since one TMI-1 net positive suction head (NPSH) case shows only 0.1 ft. of margin (although it cannot be determined from the submittal when this occurs). Please provide an extrapolation of the data to the mission time or explain why one is not necessary.

10. Section 3f.8 of the supplemental response does not identify any conservatisms or margins for the head loss calculations. Please list any conservatisms or margins associated with the head loss and vortexing evaluation.

11. Section 3f.14 of the supplemental response states that containment accident pressure was not credited to ensure that flashing would not occur. However, the response (Section 3f.10) states that above 140 degrees the head loss is 1.7 ft. This is more than the submergence of the strainer (15 inches). Please explain the apparent discrepancy, and describe and justify whether flashing occurs.

12. The response to Section 3o.2.2.i is not sufficiently justified. The question asks if the test debris selection will result in the highest head loss. The AmerGen response was that the maximum amounts of fibrous and particulate debris were used in integrated chemical effects testing. This response does not address the potential for the formation of a chemical thin bed. Testing that searches for potential thin beds should use prototypically fine fiber introduced in graduated amounts. Please address the potential formation of a thin bed during integrated chemical effects testing.

13. The response to Section 3o.2.15.i stated that some debris settlement occurred during stirred integrated chemical effects testing, but concluded that the quantities were negligible and reasonable without apparent justification. Please provide a justification for why the settlement that occurred during integrated chemical effects testing did not result in non-conservative head loss values.

14. Please supplement the response to Section 3o.2.16.i to provide the test termination criteria.

15. Sections 3o.2.19 and 3o.2.20 were evaluated as not applicable to TMI-1. Please provide a more comprehensive answer to these questions regarding representative testing, specifically addressing tank scaling, bed formation, and debris transport.

16. Please provide a more detailed description of the NPSH margins calculation methodology, including a description of the time-dependent analysis specifying selected values for NPSHa (NPSH available) and NPSHr (NPSH required) throughout the mission time. This description should include significant time-dependent variables and how they change throughout the postulated event. For example, head loss changes due to chemical effects at different temperatures and changes in head loss and NPSHa due to sump temperature changes should be discussed.

17. Please clarify the NPSH result Tables 14 and 15, including a statement as to whether or not the screen and debris losses are included in all of the results. The clarifying information should specify the equipment qualification and maximum reactor building cooling scenarios, and provide an evaluation as to why these scenarios represent the most limiting cases for NPSH margin calculations throughout the mission time.

18. Please provide a technical basis for Building Spray flow rate. In doing so, please describe whether it is a maximum/runout flow rate, a calculated flow, or a flow rate set by an operator under proceduralized criteria. If Building Spray rate is calculated, provide a description of the system lineup, the plant conditions, the calculation method, and a list of assumptions and conservatisms.

19. Please provide a discussion of how the single failure criterion was used in determining the bounding NPSH margin and why there is confidence that the worst-case single failure was identified and considered.

20. Please provide a description of the qualified sealant mentioned as being used in the Reactor Building Moisture Barrier and provide a brief summary of the testing performed to qualify the sealant.

21. Please summarize the evaluation of the flow paths from the postulated break locations and containment spray washdown to identify potential choke points in the flow field upstream of the sump.

22. The TMI-1 GL 2004-02 response dated September 1, 2005, identified the doorways to the D-ring and the incore chase areas as potential choke points for water flow to the sumps, whereas the December 29, 2007, GL 2004-02 supplemental response stated a replacement of the doors to the entrances to the D-rings was performed as a configuration change. Please describe the new door configuration and provide the basis for the statement that the D-ring and incore chase area doorways are no longer choke points based on plant modification or other considerations.

23. Please describe how potential blockage of the fuel transfer canal drain was evaluated, including likelihood of blockage. The response mentions the existence of a 4" drain line in the fuel transfer canal which drains to the recirculation sump. Please summarize how any temporary water holdup is integrated into the overall analysis.

24. The supplemental response notes that the following drain lines were redirected to the new normal sumps:

    Four-inch FTC [Fuel Transfer Canal] drain downstream of valve SF-V-31
    Two-inch Reactor cavity drain line discharging through WDL-V-520
    Two other four-inch embedded RB [Reactor Building] floor drain lines
    One-half inch leak off drain line from SF-V-24

    Please provide information on how the flow from these drain lines, and any other lines which drain to the normal sumps, is integrated into the overall sump water level analysis. If these drain flows are credited on the sump water level evaluation, please provide the basis for ensuring that these drain lines will not become blocked by debris during a LOCA.

25. The supplemental response notes that a fuel and vessel downstream effects evaluation prepared by AREVA (using WCAP-16406-P) to estimate the effect of core blockage on core cooling results in a calculated cladding temperature of less than 904 degrees Fahrenheit (F). The response also states that the core chemical effects evaluation using LOCADM spreadsheet software calculated a peak fuel cladding temperature of 439 degrees F. The acceptance criterion for cladding temperature in WCAP-16793-NP, revision 0, is 800 degrees F, as noted in the supplemental response. Please describe how these differing results are reconciled and if a calculated temperature above 800 degrees F is found to be acceptable, please provide the basis for this conclusion. Specifically, if the calculated temperature exceeds 800 degrees F, include cladding strength data for oxidized and pre-hydrided cladding material that exceeds this temperature limit.

26. The supplemental response states that aluminum-based precipitates do not form above 140 degrees F, but it does not provide data to support this assertion. Please provide the test data that forms the basis for this assertion.

27. Please explain what test parameters (e.g. visual, pressure drop) were measured to determine that no precipitates were formed above 140 degrees F, and explain whether it is possible that precipitates formed at temperatures above 140 degrees F but were not detected during the test.

28. Please compare the test loop pH to the expected equilibrium containment pool pH following a LOCA. Discuss any differences between the pH values in terms of potential effects on aluminum solubility.

29. In the supplemental response, Table 19, sodium aluminum silicate settled volumes, do not meet the criteria in the NRC's SE for WCAP-16530 for "aluminum containing precipitate" (1 hour, >6 ml). Given this discrepancy, please explain why the settlement values used for TMI-1 testing are acceptable.

30. The NRC staff considers in-vessel downstream effects to be not fully addressed at TMI-1 as well as at other pressurized water reactors. The TMI-1 submittal refers to draft WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid." The NRC staff has not issued a final SE for WCAP-16793-NP. AmerGen may demonstrate that in-vessel downstream effects issues are resolved for TMI-1 by showing that the TMI-1 plant conditions are bounded by the final WCAP-16793-NP and the corresponding final NRC staff SE, and by addressing the conditions and limitations in the final SE. AmerGen may also resolve this item by demonstrating, without reference to WCAP-16793 or the staff SE that in-vessel downstream effects have been addressed at TMI-1. In any event, AmerGen is requested to report how it has addressed the in-vessel downstream effects issue within 90 days of issuance of the final NRC staff SE on WCAP-16793.