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Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

MAR 3 1 2008

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U S Nuclear Regulatory Commission
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Rockville, Maryland 20852

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Supplemental Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," for the Prairie Island Nuclear Generating Plant (TAC Nos. MC4707 and MC4708)

References:

1. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors", dated September 13, 2004, Accession Number ML042360586.
2. Request for Extension of Supplemental Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," for the Prairie Island Nuclear Generating Plant, dated December 5, 2007, Accession Number ML073400458.
3. NRC letter to Nuclear Management Company, LLC (NMC), "Generic Letter 2004-02 'Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors' Extension Request Approval for Prairie Island Units 1 and 2", dated December 21, 2007, Accession Number ML073520053.
4. NRC letter to the Nuclear Energy Institute (NEI), "Supplemental Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation During Design Bases Accidents at Pressurized-Water Reactors'", dated November 30, 2007, Accession Number ML073320176.
5. NMC Supplemental Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," for the Prairie Island Nuclear Generating Plant, dated February 28, 2008, Accession Number ML080590629.

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6. "Prairie Island Nuclear Generating Plant Corrective Actions for Generic Letter 2004-02," Audit Report, dated May 2, 2007, Accession Number ML070750065.

By letter dated September 13, 2004, the NRC issued GL 2004-02 (Reference 1) which requested that all actions for resolution of GL 2004-02 issues be completed by December 31, 2007.

By letter dated December 5, 2007 (Reference 2), NMC requested an extension to March 31, 2008 for completion of the ex-vessel downstream effects analysis and in-vessel effects analysis for the Prairie Island Nuclear Generating Plant (PINGP). By letter dated December 21, 2007 (Reference 3), the NRC granted the extension.

In a letter to NEI dated November 30, 2007 (Reference 4), the NRC authorized all pressurized water reactor licensees until February 29, 2008, to provide the supplemental responses to the NRC with the stipulation that:

> Licensees for plants for which corrective actions are still incomplete by February 29, 2008 (i.e., plants with approved extensions to a later date), should submit a supplemental response to GL 2004-02 by February 29, 2008, noting remaining corrective actions and plans for accomplishing them, including target dates and milestones.

In conformance with the guidance of Reference 4, NMC submitted a supplemental response on February 28, 2008 (Reference 5) noting remaining corrective actions relating to ex-vessel and in-vessel downstream effects analyses and specified March 31, 2008 as the date for completion.

In the autumn of 2006, the NRC conducted an audit of PINGP corrective actions for GL 2004-02. The NRC audit report (Reference 6) identified open items which must be addressed to close GL 2004-02 issues for PINGP. The supplemental responses provided in Reference 5 addressed the open items with the exception of those related to ex-vessel and in-vessel downstream effects.

The Enclosure to this letter provides additional supplemental information which addresses ex-vessel and in-vessel downstream effects analyses. This information supplements the responses provided in Reference 5, Sections 3.m. and 3.n., and closes the NRC audit report open items related to ex-vessel and in-vessel downstream effects which Reference 5 did not close.

If there are any questions or if additional information is needed, please contact Mr. Dale Vincent, P.E., at 651-388-1121.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on MAR 3 1 2008

Michael D Wadley

Michael D. Wadley
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Nuclear Management Company, LLC

Enclosures (1)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC

ENCLOSURE

Supplemental Response to NRC Generic Letter 2004-02
Prairie Island Nuclear Generating Plant

By letter dated February 28, 2008, the Nuclear Management Company (NMC) provided supplemental responses (Reference 1) to the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02. With the exception of two areas, Reference 1 responded to the issues identified for resolution as part of GL 2004-02. The two areas that necessitate further evaluation are as follows:

- Downstream Effects in Components and Systems.
- Downstream Effects in the Fuel and Vessel.

NMC recognized that these two areas would not be fully resolved in time to be included in the February 28, 2008 supplemental response, thus NMC requested, and was granted, an extension until March 31, 2008, to provide the responses for these areas. This additional supplemental response provides the information for these two areas. The other areas addressed in Reference 1 are not affected by this response.

Attachment 1 to the Reference 1 Enclosure included a table showing the ties between the NRC Content Guide section and the Prairie Island Nuclear Generating Plant (PINGP) Audit Report section, Open Items from the Audit Report, and resolution of the Open Items. This same table is also included as Attachment 1 in this additional supplemental response and updated to reflect resolution of the two items noted above.

The format for this additional supplemental response follows the same format used in Reference 1, but only addresses the areas that were still open in Reference 1.

1. Overall Compliance

   Refer to discussion in Reference 1.

2. General Description of and Schedule for Corrective Actions

   Refer to discussion in Reference 1. This additional supplemental response provides the information noted in Section 2 of the Reference 1 Enclosure. There are no additional outstanding actions.

3. Specific Information Regarding Methodology for Demonstrating Compliance

This section only addresses subsections m and n. Refer to Reference 1 for all other subsections.

m. Downstream Effects – Components and Systems (Ex-vessel)

The downstream effects analyses for the components and systems, outside of the reactor vessel, were reviewed in detail during the NRC Staff audit of PINGP. A summary of the analyses that were reviewed by the NRC Staff during the audit was provided in Section 3.m of the supplemental response submitted February 28, 2008 (Reference 1). The downstream effects analyses concluded that debris ingestion would not prevent the emergency core cooling system (ECCS) from performing its required design functions.

During the NRC Staff audit, several open items were identified regarding downstream effects on ECCS components outside of the reactor vessel which were designated as Open Items 5.3-3, 5.3-4, 5.3-5, 5.3-6 and 5.3-7. The NRC Audit Report (Reference 2), Section 5.3.2, pages 66 and 67, discussed the finding with respect to the ex-vessel component evaluations and the associated Open Items. NMC has taken actions to address these Open Items as follows.

In August, 2007, Westinghouse issued a revised version of WCAP-16406 (Reference 5) to provide methods for evaluating wear effects downstream of the sump strainer. In Reference 6, the NRC Staff documented the acceptability of using the methods in Reference 5. NMC has subsequently revised the analyses of downstream (ex-vessel) wear effects to be consistent with the methods in Reference 5, as approved in Reference 6. The revised analysis incorporates Pressurized Water Reactor Owner's Group (PWROG) updates provided in letters OG-07-510 and OG-07-412 (References 7 and 8, respectively) and fully addresses the limitations in Reference 6.

In order to address the ex-vessel Open Items from Reference 2, Section 5.3.2, the revised analyses focus on the components in the Residual Heat Removal (RHR) system and in the Safety Injection (SI) system, that is, pumps, pump seals, heat exchangers, orifices and valves, with the exception of the SI pumps. The previous evaluation of the SI pumps is considered acceptable based on the conservative nature of the analysis, minimal predicted wear and short operating time period during recirculation from the containment sump.

Wear due to both abrasion and erosion is considered in the revised analyses. The abrasive models are used to evaluate the pumps and include consideration of both free flow type and packing (or Archard's) type.

The methods used for these wear and abrasion evaluations are consistent with Reference 5. Utilizing the evaluation methods in Reference 5, the analyses include the following conservative inputs and assumptions:

- Debris quantities used in the analyses envelope those identified inside of either containment. Debris size distributions are selected to maximize the predicted wear on the downstream components.
- 100% of the enveloping debris quantity is assumed to reach the strainer.
- Once in solution, the debris size is assumed to remain constant providing a conservative wear and abrasion evaluation.
- No credit is taken for debris depletion due to the sump strainer, the fuel screen or fuel space grids. The only debris depletion mechanism credited in the analysis is settling of particulate debris in the reactor vessel lower plenum. The amount of particulate debris that settles is determined analytically. The analysis of debris settlement in the reactor lower plenum uses conservative inputs and assumptions, such as using conservatively low flow rates.
- The flow velocity is assumed to be maximum flow rate corresponding to runout flow of the applicable pump except for RHR and SI minimum flow lines. This assumption maximizes the predicted wear.
- The RHR pump performance is assumed to be at the minimum acceptable performance per the In-Service Testing (IST) program prior to the accident. This is conservative as it minimizes the margin available for flow degradation due to postulated wear.
- A mission time of 30 days is used for the RHR system components.
- A mission time of 10 hours is used for the SI system components.

The following acceptance criteria are used in the analyses:

RHR Pumps: After the 30 day mission time, the RHR pump performance is verified to be sufficient to provide the required flow rates.

RHR Pump Seals: Wear of the pump seals is evaluated and the leakage determined. The acceptance criteria for the seal leakage used in the analysis is 50 gallons per minute (gpm).

Heat Exchangers: Worn tube wall thickness is compared to minimum tube wall thickness at accident condition differential pressures (internal and external) across the tube. This acceptance criteria is used to ensure that the tube will not fail.

Orifices and Manual Throttle Valves: The additional opening of the orifices due to wear is evaluated to ensure that the amount of orifice wear will have negligible effect on system flow rate. The system may see an increase in flow rate but the pumps must continue to be protected from runout conditions.

Instrumentation: Instrumentation taps are evaluated to ensure that debris will not settle in the tubing such that the debris could affect functionality of instrumentation needed for post-accident mitigation.

Lift Check Valves: Lift check valves are evaluated for the potential to stick open or leak due to debris accumulation.

The results of the revised analyses show that all of the acceptance criteria are satisfied and that the ECCS can perform the required functions for the stated mission time with the debris laden fluid. This is summarized in Table 1 which follows.

Table 1
Ex-vessel Analyses Results

Component	Results Summary
RHR Pumps	Performance capability of RHR pump after 30 days of wear is adequate to provide more than minimum required flow rates.
RHR Pump Seals	Maximum predicted flow rate through the seals is well within the acceptance criteria of 50 gpm.
RHR Heat Exchanger	Wear on RHR heat exchanger tubes after 30 days is insignificant.
RHR Orifices	Affect on RHR system flow rate due to wear of the orifice will not preclude the pumps from providing more than the minimum required flow rates.
SI Orifices and Manual Throttle Valves	Affect on SI system flow rate due to wear of the orifice and manual throttle valves will continue to provide acceptable flow rate and prevent the SI pumps from runout conditions.
Instrumentation	Postulated debris settlement in instrument taps will not affect the functionality of instrumentation needed for post-accident mitigation.
Lift Check Valves	Postulated debris settlement in lift check valves will not preclude the ECCS from performing the required post-accident functions.

The analyses support resolution of each of the Audit Report (Reference 2) Open Items as follows:

Audit Report Open Item 5.3-3:

> The licensee did not document a basis for the assumption of 95% efficiency in system depletion calculations. .

Depletion is not credited in the revised analysis in any area other than in the reactor vessel lower plenum. The particulate debris quantity that settles in the reactor vessel lower plenum is determined analytically based on flow rate, fluid viscosity, and particulate size.

Audit Report Open Item 5.3-4:

> The licensee did not evaluate pump hydraulic degradation due to RHR pump internal wear.

The revised analyses include a determination of RHR pump hydraulic degradation due to internal wear. The pump is assumed to initially be operating at the minimum IST acceptance criteria. At the end of the 30 day mission time, pump performance is determined based on the wear. The pump performance capability at this time is then used to show that the pump can still provide the required flow rates.

Audit Report Open Item 5.3-5:

> PI [NMC] did not provide an evaluation supporting using the criterion contained in American Petroleum Institute Standard 610 for pump vibration, which applies to new pumps.

The revised analysis does not use the American Petroleum Institute (API) Standard 610 criterion. Instead, as discussed above, the revised analyses include a determination of RHR pump hydraulic degradation due to internal wear. The pump is assumed to initially be operating at the minimum IST acceptance criteria. At the end of the 30 day mission time, pump performance is determined based on the wear. The pump performance capability at this time is then used to show that the pump can still provide the required flow rates.

Audit Report Open Item 5.3-6:

> PI [NMC] did not justify use of a three-body, erosive wear model for pump internals. The industry standard model is to consider internal wear mechanism for internal, non-impeller wear is two-body.

The revised analyses use erosive and abrasive wear models from Reference 5 which were determined to be acceptable by the NRC Staff in Reference 6.

Audit Report Open Item 5.3-7:

> The licensee did not quantify seal leakage associated with downstream effects into the auxiliary building, nor evaluate the affects on equipment qualification, sumps and drains operation or room habitability.

The revised analyses include an evaluation of pump seal leakage and conclude that the predicted leak rate is acceptable.

In conclusion, NRC Audit Report Open Items 5.3-3, 5.3-4, 5.3-5, 5.3-6 and 5.3-7 have been adequately addressed by the revised analyses and NMC considers them closed.

n. Downstream Effects – Fuel and Vessel (In-vessel)

As discussed in Reference 1, the downstream effects analyses for the in-vessel components (fuel assemblies and other components inside the reactor vessel) were reviewed in detail during the NRC Staff audit of PINGP (Reference 2). Two Open Items were identified in Reference 2 related to the downstream effects analyses for the in-vessel components. To summarize, these two Open Items are:

Open Item 5.3-1 states:

> The licensee evaluations of downstream component effects are preliminary; based in part on the generic methodology of WCAP-16406-P, currently under review by the NRC staff. Conclusions and findings need to be applied to the evaluation of post-LOCA downstream effects for PI [PINGP].

Open Item 5.3-2 states:

> The licensee had not completed in-vessel downstream evaluations, including the effect on core heat transfer of plate-out of material on the surface of fuel rods during long-term boiling and the effect of any debris trapped between fuel element spacer grids and the adjacent fuel rod in the production of local hot spots.

NMC has taken the following actions to close Open Items 5.3-1 and 5.3-2.

Subsequent to the NRC audit of PINGP, the PWROG issued WCAP-16793 (Reference 3) to provide analyses that bound most, if not all, operating Pressurized Water Reactors (PWRs). WCAP-16793 considers the following three topical areas:

1. Evaluation of fuel clad temperature response to blockage at the inlet to the core;

2. Evaluation of fuel clad temperature response to local blockages or chemical precipitation on fuel clad surface; and

3. Evaluation of chemical effects in the core region, including potential for plate-out on fuel cladding.

For PINGP, NMC completed analyses to evaluate downstream effects in the reactor vessel internals and fuel assemblies following the methodology in WCAP-16793. WCAP-16793 was reviewed for applicability to PINGP to confirm that the PINGP design is bounded by the analyses contained therein. Plant specific considerations such as the quantity of fiber that reaches the reactor vessel (i.e., fiber that bypasses the strainer), the size of the fibers that bypass the strainer, strainer perforation size, time to initiation of recirculation, debris quantities, and plate-out on fuel cladding were reviewed as part of the site specific evaluation. For fiber quantities that reach the vessel, fiber size, strainer perforation size, time to initiation of recirculation, and debris quantities, the site specific conditions for the PINGP are bounded by the considerations used in WCAP-16793.

As discussed in Reference 1, for the analysis of the potential for plate-out on the fuel assemblies, without more details of the example analyses in WCAP-16793, it could not be shown definitively that the PINGP is bounded by WCAP-16793. Thus, to address this potential issue, NMC performed a site specific plate-out analysis. The Loss Of Coolant Accident Deposition Analysis Model (LOCADM) was used in this analysis to predict plate-out of chemicals on fuel rods and maximum subsequent fuel assembly cladding temperatures.

The overall methodology for running LOCADM utilized in the NMC analysis is as follows:

1. Determine initial inputs and run LOCADM to determine initial outputs;

2. Double aluminum surface areas, input the total corroded aluminum from step 1 to satisfy the NRC expectation of doubling corrosion rates and rerun LOCADM with the new values;

3. Calculate the required bump-up factor to account for potential screen bypass of fibrous material and modify material inputs as applicable; and

4. Perform final LOCADM run to calculate final scale thickness and cladding temperature.

Assumptions and Design Inputs used in the LOCADM analysis were selected to provide conservative predictions of deposit thickness and subsequent fuel surface temperatures in order to provide bounding results. Several of the key assumptions used in the analysis are as follows:

- Once formed, deposits on fuel heat transfer surfaces are assumed not to be thinned by flow attrition or by dissolution.

- All deposition is assumed to take place on the fuel heat transfer surfaces. A best-estimate approach would have accounted for deposition on non-fuel surfaces such as the RHR heat exchangers and surfaces in containment, resulting in thinner deposits.

- No moisture carry-over is assumed to be present in the steam exiting the reactor vessel. Experimental measurements simulating the post-loss of coolant accident (LOCA) environment indicate that concentration of non-volatile material within the reactor vessel will be considerably reduced if moisture carryover is included in the estimation. Not including boron and coolant impurities in the moisture carryover is conservative but unrealistic.

- The boiling point is assumed not to be affected due to the concentration of solutes. This simplification results in over-prediction of boiling in the core and thus any error introduced by the simplification will be conservative.

- Only species that have dissolved into solution or species that have dissolved and then precipitated into suspended particles are considered. The transport of large debris particles from containment and re-deposition of debris from fuel failures have not been included. Larger debris will either settle or will be physically retained by the sump screen, the fuel assembly inlet debris filters, or in other locations where flow is restricted.

- All impurities transported into a deposit by boiling are assumed to be deposited at a rate that is equal to the product of the steaming rate and the coolant impurity concentration.

- The non-boiling rate of deposit build-up is proportional to heat flux and is 1/80th of the rate of boiling deposition at the same heat flux.

- A conservative minimum time frame for the initiation of recirculation from the containment sump is used to ensure that dissolved chemicals in the sump pool are circulated through the reactor vessel at the earliest time possible, thereby maximizing the decay heat at initiation of recirculation. This will maximize boiling in the presence of recirculation fluid and initiate plating on fuel rods earlier.

- The operating time for the Containment Spray system is maximized to maximize the time that unsubmerged surfaces are exposed to the spray flow and therefore maximize dissolution from unsubmerged surfaces.

The analysis based on the above conservative methods, inputs and assumptions shows that the maximum expected scale thickness on fuel rods will be significantly less than the 50 mil acceptance criteria with a peak cladding surface temperature less than 800 °F.

NMC recognizes that the NRC is currently reviewing WCAP-16793 and has provided several Requests for Additional Information (Reference 4). The resolution of the requests in Reference 4 may necessitate revising WCAP-16793. NMC will evaluate any subsequent revisions to WCAP-16793 for the affect on the analysis and revise the analysis accordingly. Based on the bounding approach to these analyses and the margin available in the results, NMC has reasonable assurance that any subsequent revisions to WCAP-16793 will not result in the analyses results exceeding the acceptance criteria.

Based on the in-vessel downstream effects analysis, discussed in Reference 1, and the plate-out analysis discussed above, PINGP is bounded by WCAP-16793. Therefore, Open Items 5.3-1 and 5.3-2 can be closed for PINGP.

References

1. Supplemental Response to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," for the Prairie Island Nuclear Generating Plant, dated February 28, 2008, Accession Number ML080590629.

2. "Prairie Island Nuclear Generating Plant Corrective Actions for Generic Letter 2004-02," Audit Report, dated May 2, 2007, Accession Number ML070750065.

3. WCAP-16793-NP, Revision 0, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid", dated May 2007, Accession Number ML071580139.

4. NRC Letter, "Request for Additional Information Re: Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16793-NP, Revision 0, 'Evaluation of Long-Term Core Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid'," dated September 10, 2007, Accession Number ML072410036.

5. Westinghouse Report, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," WCAP-16406-P, Revision 1, August 2007.

6. NRC, "Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report (TR) WCAP-16406-P, Revision 1, Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Pressurized Water Reactor Owners Group, Project No. 694, dated December 20, 2007, Accession Number ML73520295.

7. OG-07-510, PWR Owners Group, Suction Multiplier for WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," (PA-SEE-0195), dated November 20, 2007.

8. OG-07-412, PWR Owners Group Web-Cast on October 3, 2007 to discuss implementation of Revision 1 to WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," (PA-SEE-0195), dated September 14, 2007.

Attachment 1
GL 2004-02 Supplemental Response Content Guide

The NRC provided a content guide (CG) for Generic Letter (GL) 2004-02 Supplemental Responses to describe content in the supplemental response that the NRC believes would be sufficient to close GL 2004-02. In accordance with the guidance of the NRC CG, for plants that were subject to an NRC audit of corrective actions for GL 2004-02 (e.g., PINGP), the supplemental response should specifically address the open items. Furthermore, for any subject area found to be acceptable during an audit, the licensee may briefly describe the approach taken in that area and refer to the Audit Report. The following table provides the ties between the CG section and the PINGP Audit Report section, noting any Open Items and how the Open Items have been addressed. PINGP also received a set of RAIs related to GL 2004-02 response (NRC letter dated February 9, 2006, “Prairie Island Nuclear Generating Plant, Units 1 and 2 – Request for Additional Information RE: Response to Generic Letter 2004-02, ‘Potential Impact of Debris Blockage on Emergency Recirculation During Design-basis Accidents at Pressurized-water Reactors’ (TAC Nos. MC4707 and MC4708)”, Accession Number ML060370394). Per the CG, specific responses to RAIs are not required provided the information in the supplemental response addresses the issues identified in the RAIs.

This table was initially provided in the supplemental response to GL 2004-02 (Reference 1). The attached table is updated to reflect the information provided in this additional supplemental response.

Content Guide Section	Content Guide Section Title	Audit Report Section	Audit Report Open Items	How is Open Item Addressed	Comments
1	Overall Compliance	N/A			
2	General Description of and Schedule for Corrective Actions	N/A			
3.a	Break Selection	3.1	None	N/A	
3.b	Debris Generation/Zone of Influence	3.2	None	N/A	
3.c	Debris Characteristics	3.3	None	N/A	
3.d	Latent Debris	3.4	3.4-1	Latent debris program established – Discussed in Reference 1	
3.e	Debris Transport	3.5	None	N/A	
3.f	Head Loss and Vortexing	3.6	3.6-1	Subsequent sampling in Unit 2 showed that there is insufficient latent debris to result in a thin bed. Additional	

Content Guide Section	Content Guide Section Title	Audit Report Section	Audit Report Open Items	How is Open Item Addressed	Comments
				testing demonstrates that PINGP testing results were very conservative. Discussed in Reference 1.	
			3.6-2	Clean strainer head loss addressed by Performance Contracting, Inc. (PCI). Discussed in Reference 1.	
			3.6-3	Vortex potential addressed by PCI. Discussed in Reference 1.	
			Action Request (AR) 01058100	Calculation revised and AR closed. Discussed in Reference 1.	

Content Guide Section	Content Guide Section Title	Audit Report Section	Audit Report Open Items	How is Open Item Addressed	Comments
3.g	Net Positive Suction Head (NPSH)	3.7	3.7-1	Calculation revised to address open item. Discussed in Reference 1.	
3.h	Coatings Evaluation	3.8	3.8-1	Coatings program provides reasonable assurance of coating qualification. Discussed in Reference 1.	
3.i	Debris Source Term	4.1	None	N/A	
3.j	Screen Modification Package	4.2	None	N/A	
3.k	Sump Structural Analysis	5.1	None	N/A	
3.l	Upstream Effects	5.2	5.2-1	Evaluation of potential hold-up regions in containment (i.e., upstream effects) documented.	

Content Guide Section	Content Guide Section Title	Audit Report Section	Audit Report Open Items	How is Open Item Addressed	Comments
				Discussed in Reference 1.	
3.m	Downstream Effects – Components and Systems	5.3.2	5.3-3	Included as part of revision to downstream effects analysis to implement NRC approved version of WCAP-16406.	Addressed in this additional supplemental response.
			5.3-4	Evaluation to be performed following revision to downstream effects analysis to implement NRC approved version of WCAP-16406.	Addressed in this additional supplemental response.
			5.3-5	Included as part of revision to downstream effects analysis to implement NRC approved version of WCAP-16406.	Addressed in this additional supplemental response.
			5.3-6	Included as part of revision to downstream effects analysis to	Addressed in this additional supplemental response.

Content Guide Section	Content Guide Section Title	Audit Report Section	Audit Report Open Items	How is Open Item Addressed	Comments
				implement NRC approved version of WCAP-16406.	
			5.3-7	Evaluation of RHR pump seal leakage	Addressed in this additional supplemental response.
3.n	Downstream Effects – Fuel and Vessel	5.3.1	5.3-1	Addressed in evaluation that reviews applicability of WCAP-16793.	Addressed in this additional supplemental response.
			5.3-2	Addressed in evaluation that reviews applicability of WCAP-16793.	
3.o	Chemical Effects	5.4	5.4-1	Addressed in a new analysis. Discussed in Reference 1.	
3.p	Licensing Basis	2.2	N/A	N/A	.