PWROG
PWR Owners Group

Program Management Office
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BAW-1543-NP, Rev.4
Project Number 694

U.S. Nuclear Regulatory Commission
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Washington DC 20555-0001

Subject: Pressurized Water Reactor Owners Group
Responses to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) Report BAW-1543 (NP), Revision 4, Supplement 6 "Supplement to the Master Integrated Reactor Vessel Surveillance Program" (TAC No. MC9608) PA-MSC-0230

References:
1. B&WOG Letter from Ronnie Gardner and Howard Crawford to Document Control Desk, Request for Review and Approval of BAW-1543 (NP) Revision 4, Supplement 6, Supplement to the Master Integrated Reactor Vessel Surveillance Program" OG-05-1877, December 20, 2005.
2. Acceptance for Review of Babcox & Wilcox Owners Group (B&WOG) Topical Report BAW-1543(NP), Revision 4, Supplement 6, "Supplement to the Master Integrated Reactor Vessel Surveillance Program" (TAC NO. MC9608) PA-MSC-0230, OG-06-336, October 17, 2006.
3. NRC email from Sean E. Peters of NRR to Tom Laubham of PWROG dated January 12, 2007, "RAIs for BAW-1543".

In December 2005, the B&WOG, now known as the Pressurized Water Reactor Owners Group (PWROG), submitted Topical Report BAW-1543(NP), Revision 4, Supplement 6, "Supplement to the Master Integrated Reactor Vessel Surveillance Program" for review and approval (Reference 1). In September 2006, the NRC accepted the topical report (Reference 2) and provided an informal Request for Additional Information (RAIs) (Reference 3) on January 12, 2007.

Attachment 1 to this letter provides RAI responses to the 1 of the 2 questions received in Reference 3. Based on a teleconference with the NRC Staff the second question was withdrawn.

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If you have any questions, please do not hesitate to contact me at (630) 657-3897, or if you require further information, please contact Mr. Jim Molkenthin of the PWR Owners Group Project Management Office at (860) 731-6727.

Regards,

P Molkenthin for

Frederick P. "Ted" Schiffley, II, Chairman
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FPS:JPM:las

Enclosure (1)

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<u>Response to</u>

<u>REQUEST FOR ADDITIONAL INFORMATION</u>
<u>REVIEW OF BAW-1543, REVISION 4, SUPPLEMENT 6,</u>
<u>"SUPPLEMENT TO THE MASTER INTEGRATED REACTOR</u>
<u>VESSEL SURVEILLANCE PROGRAM"</u>

Topical report BAW-1543(NP), Revision 4, Supplement 6, "Supplement to the Master Integrated Reactor Vessel Surveillance Program" lists the changes that were made to the report in the newest supplement. The NRC Staff requires additional information about some changes to complete the review of the topical report.

1. In Table III and Table VI, the withdrawal schedules for capsules A2 and A4 were changed from the end of the seventeenth fuel cycle to the end of the twenty-ninth fuel cycle. Please describe the significance of this change. ...

Response:

<u>Capsule A4 in CR-3</u>

Capsule A4 contains specimens from three Linde 80 weld wire heats (72442, 72105 and 299L44). The 72442 heat is limiting in Oconee 3 and Point Beach 2 (PB-2). Ductile-to-brittle transition temperature (DBTT) data is already available at a fluence corresponding to 80+ years of operation for Oconee 3. Therefore, no further 72442 data is needed for Oconee 3. The neutron flux of weld heat 72442 in Capsule A4 is such that the fluence does not lead the PB-2 reactor vessel (RV) heat 72442 weld at the inside surface until about 2011. Withdrawal of A4 at the end of cycle (EOC) 17 (planned for 2011) would correspond to a PB-2 RV fluence at about 40 years of operation. The A4 data is not useful for the 40-year fluence since it does not lead at this point. However, it will lead enough to provide 60-year or 80-year fluence for the PB-2 RV heat 72442 weld. Withdrawal around 2036 (60 years of operation) would provide 80-year data. These withdrawal dates assume that CR-3 operates into the license renewal period. Withdrawal of the PB-2 supplemental capsule (which also contains heat 72442) around 2016 will provide specimens with a fluence corresponding to 60 years of operation.

The 72105 heat is limiting in CR-3 and TMI-1 (with application of BAW-2308 Revision 1). There is DBTT and upper-shelf fracture-toughness data with fluence corresponding to ~80 years of operation for these two RVs for this wire heat. To provide upper-shelf toughness data that can be used to generally validate the Linde 80 upper-shelf toughness model to higher fluences which are applicable to the Westinghouse designed plants, the A4 capsule can be tested at a fluence ~7 x10^{19} n/cm^2. The typical projected peak fluence after 60 years of operation for the B&W designed plants is ~1.4x10^{19} n/cm^2, while for the Westinghouse designed plants the typical peak fluence is ~5x10^{19} n/cm^2. Withdrawal of capsule A4 at EOC 17 would provide wire heat 72105 specimens with a fluence of ~3x10^{19} n/cm^2.

The 299L44 heat is limiting in Oconee 2, Surry 1 and TMI-1. DBTT and upper-shelf fracture toughness data through 60+ years of operation for these plants is available for wire heat 299L44.

To provide upper-shelf toughness data that can be used to generally validate the Linde 80 upper-shelf toughness model to higher fluences, the A4 capsule should be tested at a fluence ~$7x10^{19}$ n/cm^2. In conclusion, the data produced from withdrawal of capsule A4 at EOC 17 would contain specimens with a fluence that would be minimally useful. Delaying the withdrawal to ~EOC 29 would produce data at a fluence which leads the PB-2 limiting material and is 1 to 2 times the fluence at 60 years of operation. In addition, the fracture toughness specimens should be used to validate the upper-shelf fracture toughness model currently in use for the fluence that the Westinghouse designed vessels are projected to experience at ~80 years of operation.

<u>Capsule A2 in CR-3</u>

Capsule A2 (in CR-3) contains welds wire heats 72445, 299L44, 71249, and 61782

The 72445 heat is not limiting in any of the participating plants.

See the discussion above for the 299L44 heat. Capsules A4 and A2 are contained in the same surveillance capsule holder tube and therefore have the same fluence projection.

The 71249 heat is limiting in Turkey Point (TP)-3, TP-4, and PB-1. The PB-2 supplemental capsule will provide data with a fluence near the 60-year vessel fluence for these three RVs. Specimens with a fluence corresponding to 40 years of operation have already been tested. After 60 years (~2040) of operation, the 71249 heat in A2 will have 80-year fluence for these three units. The flux of the weld heat 71249 is such that the fluence does not lead the PB-1 RV ID heat 71249 weld until after about 2011. The A2 capsule also does not lead the TP-3 and TP-4 vessels until about 2015. Withdrawing capsule A2 at EOC 17 would produce specimens with a fluence corresponding to ~40 years of operation, but it would not be useful since it does not lead the applicable RVs at this point. However, it will lead enough to provide 60-year or 80-year fluence for the TP-3, TP-4, and PB-1 RVs. In addition, the TP4-X capsule contains this wire heat and is slated for withdrawal with a target fluence of ~$6x10^{19}$ n/cm^2.

Heat 61782 may be P-T limiting in the PB units after BAW-2308, Revision 1 is implemented. DBTT data are currently available beyond the 80-year 1/4T fluence of PB-1, therefore additional data is not required.

In conclusion, withdrawal and testing of the A2 capsule at EOC 17 would produce data that lags the fluence of the vessels containing the weld in which it is limiting. In addition, there is already DBTT data available for heats 71249 and 61782.

1. continued: ... In addition, list the units (if any) that will require data from these capsules to meet regulatory requirements (i.e., should capsules A2 and A4 be included in Table VIII?).

Response:

The capsules listed in Table VIII fulfill the requirements of ASTM E185-82 (CFR 50 Appendix H), however they do not all contain the limiting RV material, since these programs were developed prior the publication of ASTM E185-82. For the B&W designed plants, the capsules listed in Table VIII complete the data required to meet ASTM E185-82 for the current license periods shown in

Table IX. For the B&W designed plants, all these capsules have been tested and DBTT and upper-shelf data is available for all the limiting materials for the license period.

The TP units have an integrated program for these two units. The TP units have the same limiting Linde 80 weld with the same projected fluence. The TP capsules contain this weld heat (71249) and the TP integrated program provides capsules that meet ASTM E185-82. The PB units surveillance programs did not include the limiting materials, however the PB capsules were withdrawn and tested to satisfy ASTM E185-82. The 'P' capsules are being held and not tested as allowed in ASTM E185-82. Participation in the Master Integrated Reactor Vessel Surveillance Program (MIRVP) has provided data for the PB limiting Linde 80 welds. The PB-2 supplemental capsule contains limiting materials for both PB units and is scheduled for withdrawal and testing to provide data for 60-year-fluence exposure of the RV. Limiting materials from both PB units are also contained in capsules A2 and A4. The withdrawal and testing of the Surry capsules meet the requirements of ASTM E185-82. Additional data has been provided through the MIRVP for use by Dominion. Additional capsules are not needed for the Surry units to meet the requirements of ASTM E185-82.

In conclusion, the capsules listed in Table VIII meet the requirements of ASTM E185-82 for all the plants listed. The PB-2 supplemental and the A2 and A4 MIRVP capsules will provide supplemental data for limiting Linde 80 heats with fluences corresponding to 60 and 80 years of operation for the Westinghouse designed plants.

2. Table VIII shows a capsule W1 that has been irradiated and tested to meet the ASTM E 185-82 5 Capsule Program requirements for Three Mile Island, Unit 1. However, the NRC Staff had no record of this particular capsule in the NRC Staff's January 6, 2005 safety evaluation for Supplement 5 of BAW-1543. It is the NRC Staff's understanding that the table in the prior safety evaluation was reviewed by the B&WOG for accuracy and consistency with the Master Integrated Reactor Vessel Surveillance Program. Please explain this discrepancy.

Response:

Matt Mitchell (NRC) agreed that no response is required for question 2.