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Ken Peters
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Waterford 3

W3F1-2003-0005

January 31, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Waterford Steam Electric Station, Unit 3
Docket No. 50-382
Response to NRC Request for Additional Information Related to the Waterford 3 60 Day Response to NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity

REFERENCES:
1. Entergy letter dated April 1, 2002, 15 Day Response to NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity (W3F1-2002-0032)
2. Entergy letter dated April 16, 2002, 30 Day Response to NRC Bulletins 2001-01 and 2002-01 for Vessel Head Inspection Findings (W3F1-2002-0037)
3. Entergy letter dated May 16, 2002, 60 Day Response to NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity (W3F1-2002-0051)

Dear Sir or Madam:

By letter dated March 18, 2002, the NRC issued Bulletin 2002-01, *Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity*, requiring licensees to provide 15-day and 60-day responses. The Entergy Operations, Inc. (Entergy) Waterford Steam Electric Station, Unit 3 (Waterford 3) 15-day and 60-day responses were provided on April 1, 2002 (Reference 1) and May 16, 2002 (Reference 3), respectively. The 15-day response focused on actions to address the recent findings at Davis-Besse, while the 60-day response focused on the adequacy of the Waterford 3 boric acid control programs.

On November 22, 2002, the NRC issued a letter to Waterford 3 regarding an industry-wide request for additional information (RAI) related to NRC Bulletin 2002-01 based on a review of licensee's 60-day responses. The Entergy response to this request for additional information for Waterford 3 is provided in the attachment to this letter.

This letter contains information responding to NRC Bulletin 2002-01 for Waterford 3 and is being submitted pursuant to 10CFR50.54(f). No new commitments are made in this letter. If you

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have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 31, 2003.

Sincerely,

K.J. Peters 1/31/03
Director, Nuclear Safety Assurance

KJP/DBM/cbh

Attachment: Response to NRC Request for Additional Information Related to the Waterford 3 60 Day Response to NRC Bulletin 2002-01

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office
Louisiana DEQ/Surveillance Division
American Nuclear Insurers

Attachment To

W3F1-2003-0005

Response to NRC Request for Additional Information Related to the Waterford 3 60 Day Response to NRC Bulletin 2002-01

Response to NRC Request for Additional Information Related to the Waterford 3 60 Day Response to NRC Bulletin 2002-01

NRC Question 1:

Clarify whether or not the bottom of the reactor pressure vessel (RPV) head is inspected. If the bottom of the RPV head is inspected, provide detailed information on the inspection techniques and the basis for the chosen techniques, scope and frequency of inspections, personnel qualifications, and degree of insulation removal for the examination. If not, provide the technical basis for not performing the inspection.

Response to Question 1:

Waterford Steam Electric Station, Unit 3 (Waterford 3) is a Combustion Engineering (CE) designed vessel with no penetrations on the bottom head of the reactor vessel. However, visual inspections of the reactor vessel bottom head are performed by VT-2 qualified personnel as required by ASME Section XI In-service Inspection (ISI) program in accordance with the Entergy Operations, Inc. (Entergy) In-service Inspection procedure CEP-ISI-001, "Waterford 3 ISI Plan." This inspection is a general area inspection of the exterior surface of the reflective insulation, which is performed at normal operating pressure following a four hour hold time. No evidence of boric acid has been detected on the insulation of the vessel bottom head during the last three refueling outages.

NRC Question 2:

Provide the technical basis for determining whether or not insulation is removed to examine <u>all</u> locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

Response to Question 2:

Insulation is removed or repositioned to allow for bare metal examinations of all reactor vessel head penetrations, all reactor coolant system (RCS) small bore nozzles, all pressurizer heater sleeves and instrument nozzles, and all primary side steam generator instrument nozzles in accordance with the Alloy 600 Program Plan. Insulation is periodically removed for large bore Alloy 82/182 weld material and volumetric and surface examinations performed in accordance with ASME Section XI. In systems borated for the purpose of controlling reactivity, insulation is removed from connections where the bolting material is susceptible to boric acid corrosion during the normal Section XI system leakage test.

<u>Reactor head penetrations</u>: The original insulation installed on the reactor vessel was metal reflective insulation manufactured by Transco. It was not designed to be removable and contained fiberglass "donuts" at each nozzle penetration to seal a gap of approximately 3 inches between the reflective insulation and the nozzles. The majority of the reflective insulation and fiberglass donuts were permanently removed during RF-11 in 2002 to allow for a 360 degree

visual examination of the nozzles. The remaining "dome panel" of reflective insulation was temporarily repositioned (lifted) to allow for a 360 degree visual examination of the remaining twenty nozzles (including the head vent) using a borescope or camera mounted on a robotic crawler. The reactor head was found to be clean of boric acid and boric acid corrosion per Waterford 3 procedure QAP-410, "Reactor Vessel Head VT Examination Alloy 600" as documented in reference 2. Removable soft blanket replacement insulation manufactured by Nukon has been installed where reflective insulation was removed to facilitate future inspections.

<u>RCS hot leg small bore instrument nozzles</u>: The original Transco insulation has been replaced with Nukon blanket insulation in the areas of the small bore nozzles and the new insulation has removable windows to facilitate routine bare metal visual inspection of the penetrations. In accordance with the Alloy 600 Program Plan, these nozzles are required to be 100% bare metal inspected each refueling outage until they are repaired and were most recently inspected during RF-11 with no boric acid detected.

<u>RCS cold leg small bore instrument nozzles</u>: The original Transco totally encapsulated mineral wool or glass wool fiber insulation, covered with a stainless steel jacket, is installed on the cold leg nozzles. Insulation on the small bore cold leg nozzles is not readily removable. Scaffolding is required to provide safe access for insulation removal, so the insulation is typically repositioned to allow for bare metal examination without completely removing the insulation.

Half (six) of the RCS cold leg nozzles are inspected every refueling outage on a staggered basis as described in the Waterford 3 Alloy 600 Program Plan. If evidence of boric acid is found, the Alloy 600 Program Plan requires that the inspection scope be increased to include all twelve (100%) of the cold leg nozzles. The technical basis for the 50% staggered examination criteria for these nozzles is their low susceptibility to PWSCC at lower operating temperatures. The RCS cold leg small bore nozzles operate at approximately 545°F compared to the 606°F for the hot leg nozzles. CE Owners Group (CEOG) Report number CE-NPSD-690-P, "Evaluation of Pressurizer Penetrations and Evaluations of Corrosion after Unidentified Leakage Develop," dated January 1992, reflects that PWSCC is thermally activated where time to failure is dependent on temperature. The report predicts a two-fold increase in time to failure for each 18°F decrease in temperature. However, as a result of previous RCS small bore nozzle leaks on the hot legs and pressurizer, a 100% bare metal inspection was performed on all RCS small bore cold leg nozzles during each of the past three refueling outages, with no evidence of boric acid observed.

<u>Pressurizer small bore instrument nozzles and heater sleeves</u>: The original Transco insulation has been replaced with Nukon soft blanket insulation. During RF-9 (1999), 10 (2000), and 11 (2002) the insulation was removed from the pressurizer bottom head and a complete 100% bare metal examination was performed. The Nukon blanket insulation at the top and side instrument nozzles is repositioned to allow for visual inspection in accordance with the Alloy 600 Program Plan. During RF-11, the insulation was repositioned at the side instrument nozzles to allow bare metal inspection, and no leaks were observed. The four instrument nozzles on top of the pressurizer were not inspected during RF-11 because they were previously replaced (two in RF-9 and the remaining two in RF-10) and significant scaffolding is required for access. The replacements were made utilizing PWSCC crack resistant Alloy 690 nozzle materials and 152 weld material. The two nozzles replaced in RF-9 were inspected in RF-10 and no leaks were observed.

Steam Generators: On the primary bowl of steam generator No. 1, the removable Transco insulation has been replaced with Nukon soft blanket insulation. On the primary bowl of steam generator No. 2, the original Transco totally encapsulated mineral wool or glass wool fiber insulation, covered with a stainless steel jacket is still installed. On both steam generators, the insulation is repositioned to allow for 100% bare metal inspection.

Half (four) of these nozzles are inspected every refueling outage on a staggered basis as described in the Waterford 3 Alloy 600 Program Plan. If evidence of boric acid is found, the Alloy 600 Program Plan requires that the inspection scope be increased to include all eight (100%) of the instrument nozzles. The technical basis for the 50% staggered examination criteria for the steam generator instrument nozzles is their low susceptibility to PWSCC at lower operating temperatures. The steam generator small bore nozzles operate at approximately 545°F compared to the 606°F for the hot leg nozzles. CE Owners Group (CEOG) Report number CE-NPSD-690-P, "Evaluation of Pressurizer Penetrations and Evaluations of Corrosion after Unidentified Leakage Develop," dated January 1992, reflects that PWSCC is thermally activated where time to failure is dependent on temperature. The report predicts a two-fold increase in time to failure for each 18°F decrease in temperature. However, as a result of previous RCS small bore nozzle leaks on the hot legs and pressurizer, a 100% bare metal inspection was performed on all steam generator instrument nozzles during each of the past three refueling outages, with no evidence of boric acid observed.

Reactor Vessel Monitor Tubes: There are two ¾ inch reactor vessel flange o-ring leakage monitor tubes which are manufactured from Alloy 600 material and welded to the reactor vessel flange. These monitor tubes monitor for leakage passed the first of two reactor vessel head o-rings. The ¾ inch piping connected to the monitor tube connection on the vessel flange is insulated with removable Transco totally encapsulated mineral wool or glass wool fiber, covered with a stainless steel jacket. This piping is normally isolated by the inner o-ring and is not inspected as part of the Alloy 600 Program Plan. Additional information is provided in the response to Question 3.

Large Bore 82/182 welds: The RCS large bore piping and nozzles were originally insulated with removable Transco totally encapsulated mineral wool or glass wool fiber insulation, covered with a stainless steel jacket. Portions of this insulation have been replaced with Nukon soft blanket insulation which is also removable. The large bore 82/182 welds on the RCS are periodically inspected in accordance with ASME Section XI Inservice Inspection (ISI) program. This program includes the examination of welds, rigid restraints and pressure boundaries of components and piping on ASME Class 1, 2 and 3 systems. The weld and rigid restraint examinations are performed during specified periods within the ISI interval. The system pressure tests of piping are performed during regular 10-year intervals which include inspections every refueling outage for Class 1 piping and once a period for Class 2 and 3 piping systems. The Waterford 3 ISI program identifies the specific welds and rigid restraints that have been selected for examination during the 10-year interval. Insulation is removed for volumetric and surface examinations performed in accordance with ASME Section XI.

Bolted connections in borated systems: Bolted connections in systems borated for the purpose of controlling reactivity are screened for boric acid corrosion susceptibility based on chemistry factors of the bolting material as defined in Entergy's ASME Section XI System Pressure Testing Program, CEP-PT-001, "ASME Section XI System Pressure Testing." As currently applied at Waterford 3, in systems borated for the purpose of controlling reactivity, insulation is removed from connections where the bolting material is susceptible to boric acid corrosion during the normal Section XI system leakage test frequency for the components.

NRC Question 3:

Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in <u>inaccessible areas</u>. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

Response to Question 3:

There are no small bore nozzles, heater sleeves, penetrations, dissimilar welds, or bolted connections on the Waterford 3 RCS that are inaccessible for inspections. The accessibility of these nozzles, heater sleeves, penetrations, dissimilar welds, and bolted connections is described above in the response to Question 2.

As discussed in the response to Question 2, there are two monitor tubes installed on the reactor vessel flange to detect leakage passed the first of two reactor vessel head o-rings. The welds to the vessel flange for these monitor tubes are in an area difficult to access and have not been inspected previously. The monitoring tubes are not normally exposed to a primary water environment during power (high temperature) operation. If there is leakage passed the inner reactor vessel o-ring into one or both monitor tubes, an annunciator will alarm in the control room making operations personnel aware of the condition. This condition would be addressed by Entergy's 10CFR50, Appendix B corrective action program and follow-up investigation would take place during the next refueling outage. The set point for this alarm is 1,600 PSIG.

Entergy is aware that Davis-Besse has reported cracking of the stainless steel monitor tube and that the root cause is still under investigation. Operating experience from Davis-Besse, like other industry operating experience, is screened for applicability to Waterford 3 in accordance with the Entergy Operating Experience Program.

NRC Question 4:

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head in-core instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head in-core instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response to Question 4:

Waterford 3 is a CE designed vessel and there are no in-core instrumentation nozzles penetrations or any other penetrations on the bottom head of the Waterford 3 reactor vessel.

NRC Question 5:

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response to Question 5:

As discussed above in the response to question 2, Waterford 3 RCS pressure boundary susceptible small bore nozzles receive a bare metal visual examination for visible leakage or the presence of boric acid deposits. These bare metal visual examinations are capable of identifying boric acid deposits resulting from leakage significantly less than that detectable by the Waterford 3 online leakage detection systems. These leakage detection systems are used to verify that the technical specification limiting condition for operation for unidentified leakage is limited to one gallon per minute. Small pressure boundary leakage experienced during an operating cycle, not identified by online leak detection systems, would be detected either when performing the outage walkdowns per UNT-007-027, "Control of Boric Acid Corrosion on the Reactor Coolant System Pressure Boundary" or discovered during the refueling outage by the bare metal nozzle inspections performed in accordance with the Alloy 600 Program Plan.

In addition, walkdowns performed in accordance with UNT-007-027, would include an assessment of any corrosion damage that may have occurred and its affect on the integrity of the RCS pressure boundary. The work management system is used to track required repairs and follow-up actions. Condition Reports are generated should conditions warrant.

Evidence of the effectiveness of the Waterford 3 boric acid corrosion prevention program was provided in the 60-day response to Bulletin 2002-01 (reference 3.)

NRC Question 6:

Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

Response to Question 6:

Susceptibility models and consequence models are not normally utilized at Waterford 3 for determining the extent of inspection or insulation removal. Waterford 3 considers all Alloy 600 nozzles exposed to T-hot temperatures as equally susceptible. Susceptibility studies have shown that PWSCC is temperature dependent and the nozzles on the cold leg side are deemed to be less susceptible. To date, there have been no PWSCC failures of cold leg nozzles reported in the industry (reference WCAP-15700, Revision 1.) Therefore, the RCS cold leg nozzles and steam generator instrument nozzles are only required to be inspected every other outage. (For additional information, refer to the discussion provided in the response to Question 2.) The top four pressurizer instrument nozzles were replaced during RF-9 and RF-10 utilizing PWSCC resistant Alloy 690 nozzle materials and therefore, were not inspected during RF-11.

As previously described, all other susceptible nozzles (including the cold leg and steam generator nozzles) have been bare metal inspected during the past three outages.

NRC Question 7:

Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

Response to Question 7:

Westinghouse reported in letter CEOG-02-259 dated December 13, 2002 that they reviewed databases and communications to determine what recommendations were made to owners of CE nuclear steam supply systems on visual inspections of Alloy 600/82/182 materials in the reactor coolant system pressure boundary. The letter summarizes the following three recommendations related to pressurizer nozzles:

(1) Inspect pressurizer small diameter Alloy 600 nozzles and heater sleeves during each refueling outage for signs of primary coolant leakage.
(2) Inspect with the insulation in place or removed (either approach is acceptable). The presence of boric acid deposits or corrosion products should be assumed to be an indication of leakage until proven otherwise and appropriate actions taken to stop the leakage.
(3) Inspect low alloy steels exposed to boric acid and promptly repair primary coolant leaks.

As described in the response to Question 2, the Alloy 600 Program Plan effectively implements these vendor recommendations in that it requires bare metal inspections of the Alloy 600 pressurizer nozzles each refueling outage.