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Dominion®

February 3, 2010

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

Serial No. 09-781
NSSL/WDC R0
Docket No. 50-423
License No. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
RELIEF REQUEST IR-3-10 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING ALTERNATIVE EXAMINATION CRITERIA FOR THE VISUAL EXAMINATIONS OF REACTOR COOLANT SYSTEM HOT LEG AND COLD LEG NOZZLE-TO-SAFE END WELDS FOR THE THIRD 10-YEAR INTERVAL

As a part of the inservice inspection (ISI) program, Dominion Nuclear Connecticut, Inc. (DNC) submitted Relief Request IR-3-10 for Millstone Power Station Unit 3 (MPS3) in a letter dated April 28, 2009. IR-3-10 requests relief from certain examination requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, and proposes alternative examination criterion for the third 10-year ISI interval at MPS3. Specifically, IR-3-10 requests more frequent volumetric examinations in lieu of bare metal visual examinations of the reactor vessel (RV) hot leg nozzle-to-safe end welds at MPS3. These welds are also addressed in Relief Request IR-3-13 (DNC Serial No. 10-001 dated February 1, 2010) which requests relief from ASME Section XI depth-sizing requirements for dissimilar metal weld examinations.

In a letter dated December 9, 2009, the NRC transmitted a request for additional information (RAI). The NRC requested that DNC respond to the RAI by January 4, 2010. On December 18, 2009, DNC requested a schedule extension to February 4, 2010 for submittal of the RAI response. The NRC approved this request.

Attachment 1 provides the DNC response to the NRC RAI addressing questions 1 through 9. Attachment 2 provides a revision to Relief Request IR-3-10 (Revision 1), to include changes resulting from the response to the NRC RAI questions.

If you should have any questions regarding this submittal, please contact Wanda Craft at (804) 273-4687.

Sincerely,

J. Alan Price
Vice President – Nuclear Engineering

Attachments:

1. Response to Request for Additional Information Regarding Relief Request IR-3-10 Alternative Examination Criteria for the Visual Examinations of Reactor Coolant System Hot Leg and Cold Leg Nozzle-to-Safe End Welds for the Third 10-Year Interval
2. Relief Request IR-3-10, Revision 1, Alternative Examination Criteria for the Visual Examinations of Reactor Coolant System Hot Leg Nozzle-to-Safe End Welds for the Third 10-Year Interval

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING RELIEF REQUEST IR-3-10 ALTERNATIVE EXAMINATION CRITERIA FOR THE VISUAL EXAMINATIONS OF REACTOR COOLANT SYSTEM HOT LEG AND COLD LEG NOZZLE-TO-SAFE END WELDS FOR THE THIRD 10-YEAR INTERVAL

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

By letter dated April 28, 2009, (Agencywide Document Access and Management System Accession (ADAMS) No. ML091310666), Dominion Nuclear Connecticut, Inc. (DNC) submitted Relief Request IR-3-10 for Millstone Power Station Unit 3 (MPS3). IR-3-10 requests relief from certain examination requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, and proposes alternative examination criterion for the third 10-year ISI interval at MPS3. Specifically, IR-3-10 requests more frequent volumetric examinations in lieu of bare metal visual examinations of the reactor vessel (RV) hot leg nozzle-to-safe end welds at MPS3. These welds are also addressed in relief request IR-3-13 (DNC Serial No. 10-001), which requests relief from ASME Section XI depth-sizing requirements for dissimilar metal weld examinations.

In a letter dated December 9, 2009, the NRC transmitted a request for additional information (RAI). The NRC requested that DNC respond to the RAI by January 4, 2010. On December 18, 2009, DNC requested a schedule extension to February 4, 2010 for submittal of the RAI response. The NRC approved this request. The following is the response to the RAI questions:

NRC QUESTION 1:

The NRC staff plans to perform a confirmatory flaw analysis to determine the estimated time for leak and rupture for the RV hot leg nozzle-to-safe end welds at MPS3. Please provide the distance from the center of the safe-end-to-nozzle weld to the center of the safe-end-to-pipe weld.

DNC RESPONSE:

The distance from the center of the reactor vessel (RV) hot leg nozzle-to-safe-end weld to the center of safe-end-to-pipe weld is documented as 2.26 inches.

NRC QUESTION 2:

DNC proposes to volumetrically inspect the RV cold leg nozzle-to-safe end welds at a frequency of every third refueling outage in lieu of Code Case N-722, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials," requirements. Code Case N-722 requires visual examinations of RV cold leg nozzle-to-safe end welds once per interval. Please clarify the reasons for requesting relief for the cold leg nozzle-to-safe end welds, since the proposed examination frequency appears to satisfy the requirements of Code Case N-722.

DNC RESPONSE:

DNC agrees with the NRC staff that because DNC satisfies the code requirements for volumetric inspection of the RV cold leg nozzle-to-safe end welds at MPS3, there is no need for relief from the requirements of Code Case N-722. Therefore, DNC has revised IR-3-10 to remove the references to the cold leg nozzle-to safe end welds.

NRC QUESTION 3:

Page 3, Section 4 of IR-3-10 states that, "[a]ttachment 2 contains Technical Evaluation M3-EV-08-0016...." Attachment 2 of IR-3-10 contains Technical Evaluation M3-EV-08-0018. Please explain the discrepancy.

DNC RESPONSE:

This is a typographical error. Relief Request IR-3-10, Revision 1, Section 4 has been revised to reflect the correct Technical Evaluation No. M3-EV-08-0018.

NRC QUESTION 4:

IR-3-10 does not mention the percent volume coverage achieved during the volumetric examinations of the subject welds by ultrasonic testing (UT) in the spring 2007 outage. Please state what percent volume coverage was obtained by UT in the spring 2007 outage.

DNC RESPONSE:

The volumetric UT examination coverage achieved for each of the subject welds during the spring 2007 outage was 100 percent.

NRC QUESTION 5:

Please discuss the details of the UT examinations performed in the spring 2007 outage and any indications detected including fabrication flaws and/or flaws that were not rejectable under IWB-3514 acceptance standards.

DNC RESPONSE:

The volumetric UT examination of the subject welds conducted during the spring 2007 outage was performed from the nozzle inside diameter (ID) in accordance with ASME Section XI, Appendix VIII, 1995 Edition, with the 1996 Addenda using Performance

Demonstration Initiative Program (PDI) qualified personnel, procedures and equipment. No recordable indications were detected.

NRC QUESTION 6:

DNC has proposed to volumetrically inspect the RV hot leg nozzle-to-safe end welds at the frequency of every other refueling outage in lieu of Code Case N-722, "Additional Examinations for PWR [pressurized water reactor] Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials Section XI, Division 1," which requires inspection every outage. Please articulate the basis for not performing the examinations every outage in terms of either acceptable level of quality and safety per Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i) or hardship without a compensating increase in the level of quality and safety per 10 CFR 50.55a(a)(3)(ii).

DNC RESPONSE:

In the April 28, 2009 submittal, DNC provided a technical evaluation that demonstrates the flaw growth potential for the RV hot leg nozzle-to-safe end welds as examined by UT from the internal diameter and performed every other refueling outage will bound any flaw that would potentially exceed code limits between inspections. Thus, the alternative of UT examinations every other refueling outage will provide an acceptable level of quality and safety under 10 CFR 50.55a(a)(3)(i) without performing the bare metal visual examination.

NRC QUESTION 7:

Page 1, Section 3, of Relief Request IR-3-10, indicates that the applicable code examination category is Examination Category B-P and applicable inspection item is B15.10. The applicable code requirement is Code Case N-722, which does not cite Examination Category B-P and Item Number B15.10. Please clarify IR-3-10 to reflect that the applicable inspection items for which an alternative is being requested are Item Numbers B15.90 and B15.95 of the ASME Code, Section XI, Code Case N-722.

DNC RESPONSE:

The reference to Category B-P and Inspection Item Number B15.10 on Page 1, Section 1 (not Section 3 as indicated above) was meant to indicate the ASME Section XI, Table IWB-2500-1, VT-2 inspection requirements for the subject welds in the 2004 Edition of Section XI. The applicable code requirement for which the alternative is being requested is that of Code Case N-722. Relief Request IR-3-10 has been revised to reflect the applicable Inspection Item No. B15.90 of ASME Section XI, Code Case N-722. Inspection Item No. B15.95 has been removed since relief from the requirements of Code Case N-722 is not required for the RV cold leg nozzle-to-safe end welds at MPS3.

NRC QUESTION 8:

On page 1, Section 3, of Relief Request IR-3-10, the first paragraph states that the applicable code requirement is Examination Category B-P. The applicable code requirement is Code Case N-722, which does not cite Examination Category B-P. Please clarify IR-3-10 to reflect that Examination Category B-P is not the applicable code requirement.

DNC RESPONSE:

The reference to Category B-P and Inspection Item Number B15.10 on Page 1, Section 3 was meant to indicate the ASME Section XI, Table IWB-2500-1, VT-2 inspection requirements for the subject welds in the 2004 Edition of Section XI. The applicable code requirement for which the alternative is being requested is that of Code Case N-722. Relief Request IR-3-10 has been revised to remove the reference to Examination Category B-P from Section 3.

NRC QUESTION 9:

Please clarify whether the VT-2 examination for leakage, in accordance with IWA-5241(b) requirement, will also be performed on the subject welds each refueling cycle. In the event leakage did occur through the weld, boric acid buildup may be detected by this examination.

DNC RESPONSE:

VT-2 examination of the subject welds in accordance with IWA-5241(b) will be performed with the system depressurized during each refueling outage to detect any evidence of leakage.

ATTACHMENT 2

RELIEF REQUEST IR-3-10, REVISION 1
ALTERNATIVE EXAMINATION CRITERIA FOR THE VISUAL EXAMINATIONS OF REACTOR COOLANT SYSTEM HOT LEG NOZZLE-TO-SAFE END WELDS FOR THE THIRD 10-YEAR INTERVAL

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

- Alternative Provides Acceptable Level of Quality and Safety –

1. ASME Code Components Affected

ASME Code Class:	Code Class 1
References:	ASME Section XI, IWA-5241 and Table IWB-2500-1 10 CFR 50.55a(g)(6)(ii)(E), Code Case N-722
Examination Category:	N/A
Item Number:	B15.90 of ASME Section XI, Code Case N-722
Description:	Alternative Examination Criteria for the Visual Examination of Reactor Coolant System Hot Leg Nozzle-to-Safe End Welds
Components:	Reactor Pressure Vessel (RPV) Nozzle-to-Safe End Welds: Outlet Nozzles: 302-121-A 302-121-B 302-121-C 302-121-D

2. Applicable Code Edition and Addenda

ASME Section XI, 2004 Edition (No Addenda)

3. Applicable Code Requirement

10 CFR 50.55a(g)(6)(ii)(E)(1), states (in part) that "all licensees of pressurized water reactors shall augment their inservice inspection program by implementing ASME Code Case N–722, Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials Section XI, Division 1," dated: July 5, 2005, Attachment 1 [Reference 8.3].

ASME Code Case N-722 , requires a Visual, VE "Bare Metal Visual" of all listed locations (See Attachment 1, Code Case N-722), which includes RPV outlet (hot leg) nozzle welds (Item Nos. B15.90).

4. Reason for the Request

The ASME Boiler and Pressure Vessel Code, Section XI, 2004 Edition, No Addenda, [Reference 8.1] is the Code Edition to be used for the Third Inservice Inspection Interval at MPS3, which began on April 23, 2009, and is scheduled to end on April 22, 2019. The requirements for Class 1 Visual, VT-2 examinations for leakage of pressure retaining components are included in this Code Edition under Examination Category B-P, "All Pressure Retaining Components." Visual, VT-2 examinations are performed each refueling outage. For insulated components, IWA-5241(b) allows this examination to be performed without insulation removal, stating, "only the examination of the surrounding area (including floor areas or equipment surfaces located underneath the components) for evidence of leakage shall be required." Based on these requirements, visual examinations for leakage may be performed on Class 1 components with insulation in place.

However, both the industry as represented in this case by the Materials Reliability Program (MRP) and the NRC staff along with ASME have concluded that a visual examination for leakage performed on insulated components is inadequate for the identification of leakage that can potentially occur as a result of primary water stress corrosion cracking (PWSCC) in items made with Alloy 600/82/182 materials. This relief request is needed to address this inadequacy.

The NRC staff provided their position on these visual examination requirements in the 10 CFR 50.55a rulemaking issued September 10, 2008 (effective date October 10, 2008) under 73 FR 52748. It included the mandatory use of ASME Code Case N-722 Attachment 1 with conditions specified in the 10 CFR 50.55a(g)(6)(ii)(E), "Reactor Coolant Pressure Boundary Visual Inspections."

Meeting these requirements is a concern because the reactor pressure vessel (RPV) nozzles at MPS3 have an insulation package surrounding each nozzle and its corresponding nozzle-to-safe end Alloy 82/182 welds, which makes them inaccessible for these required bare metal visual examinations.

Prior to the fall 2008 MPS3 refueling outage, DNC submitted a letter[1] to the NRC notifying the staff that DNC was implementing a deviation from the requirements of MRP-139 [Reference 8.2]. This deviation was to not perform the required bare metal visual examinations specified in MRP-139 based on the restricted access to the MPS3 RPV nozzle welds caused by the insulation package surrounding each nozzle. Attachment 2 contains the Technical Evaluation M3-EV-08-0018 that provided the basis for not performing these bare metal visual examinations. The restrictions from performing the bare metal visual examinations of MRP-139 are the same as the restrictions of the requirements of Code Case N-722. This relief request is written to address the restrictions and propose an alternative in lieu of the requirements of Code Case N-722.

[1] DNC Letter to NRC, "Dominion Nuclear Connecticut, Inc. Millstone Power Station Unit 3 Electric Power Research Institute MRP-139 Deviation Notification," dated October 27, 2008, ADAMS Accession No. ML083010233

5. Proposed Alternative and Basis for Use

Table 1 is provided to illustrate the examinations that will be performed under this requested alternative. These examinations are derived from the flaw tolerance evaluation contained in Attachment 2.

These modifications to the requirements of Code Case N-722 (Attachment 1) are shown in Table 1 below, with the modifications bolded and underlined for ease of identification.

Table 1

Examination Categories

MPS3 Class 1 PWR Components Alloy 600/82/182

Item No.	Parts Examined	Examination Requirements	Examination Method	Acceptance Standards	First Inspection Interval	Successive Inspection Intervals	Deferral of Inspection To end of Interval
B15.90	Hot leg nozzle-to-pipe connections	All 4 hot leg nozzle-to-safe end welds	**Volumetric[a,b]**	**IR-3-10** **Para. 5.1**	**Every other refueling outage[c]**	Same as for 1st interval	Not permissible

Table Notes:

(a) UT will be performed from the inside diameter of these welds in lieu of the Visual Examination, (VE).

(b) All UT examinations will meet the appropriate supplement of Section XI, Appendix VIII of the ASME Boiler and Pressure Vessel Code. The required weld volume shall be as shown on Fig. IWB-2500-8(c) of ASME Section XI, 2004 Edition (No Addenda), [Reference 8.1].

(c) UT will be performed every other refueling outage. These welds were last examined in the spring of 2007 (3R11) outage with 100 percent volumetric UT examination coverage achieved for each of the subject welds. The welds are due to be examined in the spring 2010 (3R13) outage based on the analysis in Attachment 2, which supports UT approximately every 36 months.

5.1 Acceptance Standards

5.1.1 Evaluation of Examination Results

5.1.1.1 General

(a) The volumetric examinations performed in accordance with IWA-2200 shall be evaluated by comparing the examination results with the acceptance standards in 5.1.2.

(b) Volumetric examination results shall be compared with recorded results of the preservice examination and prior inservice examinations. Acceptance of welds for continued service shall be in accordance with 5.1.2.

5.1.2 Acceptance

5.1.2.1 Acceptance by Volumetric Examination

(a) A weld whose volumetric examination confirms the absence of flaws shall be acceptable for continued service.

(b) A weld with planar surface flaws in the butt weld or base metal inside surface shall be accepted for continued service in accordance with the provisions of 5.1.2.2 or 5.1.2.3. Other flaws shall meet the acceptance standards of IWB-3514 or be accepted for continued service in accordance with 5.1.2.2 or 5.1.2.3.

5.1.2.2 Acceptance by Repair/Replacement Activity or Corrective Measures

(a) A weld whose volumetric examination reveals a flaw not acceptable for continued service in accordance with the provisions of 5.1.2.3 is unacceptable for continued service until the additional examinations of 5.2 are satisfied and the weld is corrected by repair/replacement activity in accordance with IWA-4000 or by corrective measures beyond the scope of this relief request (e.g. stress improvement).

5.1.2.3 Acceptance by Evaluation

(a) A weld whose volumetric examination detects planar surface flaws in the butt weld or base metal inside surface, or other flaws (5.1.2.1(b)) in the required examination volume that exceed the acceptance standards of IWB-3514, is acceptable for continued service if an analytical evaluation meets the requirements of IWB-3600 and the additional examinations of 5.2 are performed in the current outage.

(b) Any weld containing a planar surface flaw in the butt weld/base metal inside surface will be reexamined at every refueling outage frequency, unless mitigated by an approved mitigation technique.

5.2 Additional Examinations

Note: MPS3 plans to examine 100% of the welds for the inspection item number in Table 1 every other refueling outage, therefore, no additional examinations would be required.

5.2.1 Examinations which reveal unacceptable flaws as defined in 5.2.1(a) and (b), below shall be extended to include examinations of additional welds during the current outage. The use of IWB-3514 is for the purpose of determination of scope expansion and not for the purposes of determining acceptability of the flaws. Acceptability of flaws is determined in accordance with 5.1.

(a) Planar surface flaws in the butt weld or base metal inside surface exceeding the surface flaw sizes of IWB-3514 are revealed.

(b) Examination volumes that reveal axial crack growth beyond the specified examination volume.

5.2.2 The number of additional weld examinations shall be equal to the number of welds for the inspection item in Table 1, originally scheduled to be performed during the present inspection period. The additional examinations shall be selected from the same inspection item and, where applicable, from welds of similar materials, construction, and the same or higher operating temperatures.

5.2.3 If the additional examinations required by 5.2.1 reveal flaws exceeding the requirements of 5.2.1(a)or (b), the examinations shall be further extended to include additional examinations during the current outage. These additional examinations shall include the remaining number of welds for that inspection item in Table 1, at the same or higher operating temperature conditions. In addition a 25% sample of welds of that Inspection Item at lower operating temperatures shall be sampled. If the examinations of this sample of welds at lower operating temperature reveal flaws exceeding the requirements of 5.2.1(a) or (b), the examinations shall be further extended to include all welds of that Inspection Item, regardless of operating temperature, within the scope of this relief request.

Basis for Use

The basis for this relief request is derived from the technical evaluation contained in Attachment 2 of the revised relief request. The evaluation, originally prepared in support of a deviation from the industry guidance document MRP-139 [Reference 8.2], reviews a flaw tolerance analysis performed by the original piping designer and concludes that the maximum flaw resulting from postulated growth during the period between volumetric examinations would remain bounded by ASME Code limits. The flaw tolerance analysis does not assume or credit that any bare metal visual examinations have been performed. It determines that volumetric examinations every other refueling outage are acceptable in lieu of bare metal visual examinations for the hot leg nozzles. It is noted that the proposed volumetric examination frequency of every other refueling outage exceeds the frequency of every five

years required by ASME Code Case N-770 for unmitigated hot leg butt welds. Additional details of the assumptions and analyses performed in support of this relief request are contained in Attachment 2 of the revised relief request.

Bare metal visual examinations are effective only after a flaw has exceeded ASME Code limits, which is unacceptable, and therefore they do not provide a direct increment of safety above that provided by the proposed program of an increased frequency of volumetric examinations. Therefore the proposed alternative in this relief request provides an acceptable level of quality and safety comparable to current requirements.

It is noted that the technical evaluation addresses a deviation from MRP-139 requirements on both hot leg and cold leg reactor vessel nozzles, however only the evaluation of the hot leg nozzles is relevant to this relief request. The evaluation that was performed for the cold legs was applicable only to MRP-139 bare metal examination frequency requirements that have since been superseded by MRP-139, Revision 1 [Reference 8.4], ASME Code Case N-722, and 10CRF50.55a requirements. DNC will comply with the current requirements for the cold leg reactor vessel nozzles.

6. Duration of the Proposed Alternative

This relief is requested for the duration of the Third Inservice Inspection Interval, which began on April 23, 2009, and is scheduled to end on April 22, 2019. This request will be used to continue scheduling the sequence examinations into the next inspection interval or until the items associated with this request are either replaced, repaired, or mitigated.

7. Precedents

Because of the unique insulation package that surrounds the MPS3 RPV nozzles, there are no precedents for this specific type of request.

8. References

8.1 ASME Boiler and Pressure Vessel Code, Section XI, 2004 Edition, No Addenda.

8.2 "Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline (MRP 139)," EPRI, Palo Alto, CA 1010087, dated August 2005.

8.3 ASME Code Case N-722, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1," dated July 5, 2005.

8.4 "Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline (MRP 139, Revision 1)," EPRI, Palo Alto, CA. 1015009 dated December 2008); as supplemented by EPRI letter MRP 2009-031, "MRP-139, Revision 1 Interim Guidance on Reconciliation of BMV Requirements with Code Case N-722 (Mandatory Element)", dated June 8, 2009.

ASME CODE CASE N-722, ADDITIONAL EXAMINATIONS FOR PWR PRESSURE RETAINING WELDS IN CLASS 1 COMPONENTS FABRICATED WITH ALLOY 600/82/182 MATERIALS SECTION XI, DIVISION 1

DATED: JULY 5, 2005

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CASES OF ASME BOILER AND PRESSURE VESSEL CODE

CASE
N-722

Approval Date: July 5, 2005

The ASME Boiler and Pressure Vessel Standards Committee took action to eliminate Code Case expiration dates effective March 11, 2005. This means that all Code Cases listed in this Supplement and beyond will remain available for use until annulled by the ASME Boiler and Pressure Vessel Standards Committee.

Case N-722
Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials
Section XI, Division 1

Inquiry: What examinations, in addition to those of Table IWB-2500-1, may be performed to provide additional detection capability for pressure boundary leakage in pressurized water reactor plants having pressure retaining partial or full penetration welds in Class 1 components fabricated with Alloy 600/82/182 material?

Reply: It is the opinion of the Committee that in addition to the examination requirements of Table IWB-2500-1 the additional examinations of Table 1 shall be performed for pressurized water reactor plants having partial or full penetration welds in Class 1 components fabricated with Alloy 600/82/182 material.

The Committee's function is to establish rules of safety, relating only to pressure integrity, governing the construction of boilers, pressure vessels, transport tanks and nuclear components, and inservice inspection for pressure integrity of nuclear components and transport tanks, and to interpret these rules when questions arise regarding their intent. This Code does not address other safety issues relating to the construction of boilers, pressure vessels, transport tanks and nuclear components, and the inservice inspection of nuclear components and transport tanks. The user of the Code should refer to other pertinent codes, standards, laws, regulations or other relevant documents.

TABLE 1
EXAMINATION CATEGORIES

CLASS 1 PWR COMPONENTS CONTAINING ALLOY 600/82/182[1]

Item No.	Parts Examined[2]	Examination Requirements	Examination Method[3,4,5]	Acceptance Standard	Extent and Frequency of Examination: First Inspection Interval	Extent and Frequency of Examination: Successive Inspection Intervals	Deferral of Inspection to End of Interval
Reactor Vessel[2]							
B15.80	RPV bottom-mounted instrument penetrations	All penetrations	Visual, VE	IWB-3522	Every other refueling outage	Same as for 1st interval	Not permissible
B15.90	Hot leg nozzle-to-pipe connections	All connections	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.95	Cold leg nozzle-to-pipe connections	All connections	Visual, VE	IWB-3522	Once per interval[6,7]	Same as for 1st interval	Not permissible
B15.100	Instrument connections	All connections	Visual, VE	IWB-3522	Once per interval[6,7]	Same as for 1st interval	Not permissible
Steam Generators							
B15.110	Hot leg nozzle-to-pipe connections	All connections	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.115	Cold leg nozzle-to-pipe connections	All connections	Visual, VE	IWB-3522	Once per interval[6,7]	Same as for 1st interval	Not permissible
B15.120	Bottom channel head drain tube penetration	All penetrations	Visual, VE	IWB-3522	Once per interval[6,7]	Same as for 1st interval	Not permissible
B15.130	Primary side hot leg instrument connections	All connections	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.135	Primary side cold leg instrument connections	All connections	Visual, VE	IWB-3522	Once per interval[6,7]	Same as for 1st interval	Not permissible
Pressurizer							
B15.140	Heater penetrations	All penetrations	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.150	Spray nozzle-to-pipe connections	All connections	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.160	Safety and relief nozzle-to-pipe connections	All connections	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.170	Surge nozzle-to-pipe connections	All connections	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.180	Instrument connections	All connections	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.190	Drain nozzle-to-pipe connections	All connections	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
Piping							
B15.200	Hot leg instrument connections	All connections	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.205	Cold leg instrument connections	All connections	Visual, VE	IWB-3522	Once per interval[6,7]	Same as for 1st interval	Not permissible
B15.210	Hot leg full penetration welds	All welds	Visual, VE	IWB-3522	Each refueling outage	Same as for 1st interval	Not permissible
B15.215	Cold leg full penetration welds	All welds	Visual, VE	IWB-3522	Once per interval[6,7]	Same as for 1st interval	Not permissible

TABLE 1
EXAMINATION CATEGORIES (CONT'D)

NOTES:

(1) Alloy 600/82/182 are equivalent to UNS N06600 (SB-163, SB-166, SB-167, SB-168 and SB-564), UNS N06082 (SFA 5.14 ERNiCr-3) and UNS W86182 (SFA 5.11 ENiCrFe-3).

(2) The reactor vessel closure head is not addressed in this Case.

(3) The Visual Examination (VE) performed on Alloy 600/82/182 components for evidence of pressure boundary leakage and corrosion on adjacent ferritic steel components shall consist of the following:

(a) A direct VE of the bare-metal surface performed with the insulation removed. Alternatively, the VE may be performed with insulation in place using remote visual inspection equipment that provides resolution of the component metal surface equivalent to a bare-metal direct VE.

(b) The VE may be performed when the system or component is depressurized.

(c) The direct VE shall be performed at a distance not greater than 4 ft (1.2 m) from the component and with a demonstrated illumination level sufficient to allow resolution of lower case characters having a height of not greater than 0.105 in (2.7 mm).

(4) Personnel performing the VE shall be qualified as VT-2 visual examiners and shall have completed a minimum of four (4) hours of additional training in detection of borated water leakage from Alloy 600/82/182 components and the resulting boric acid corrosion of adjacent ferritic steel components.

(5) An ultrasonic examination, performed from the component inside or outside surface in accordance with the requirements of Table IWB-2500-1 and Appendix VIII (1995 Edition with the 1996 Addenda or later) shall be acceptable in lieu of the VE requirement of this table.

(6) VE shall be performed in accordance with the schedule in IWB-2400.

(7) The detection of evidence of pressure leakage at a VE location shall require the VE of all components within the Examination Item No. prior to reactor startup. These additional VEs shall not affect the original VE schedule of the components within the Examination Item No.

Dominion Nuclear Connecticut, Inc Millstone Power Station Unit 3 Electric Power Research Institute MRP-139 Notification and Technical Evaluation for Technical Justification for deviation from Mandatory Requirements of MRP-139 Millstone Unit 3 M3-EV-08-0018 Rev.0 - 9/25/08

MILLSTONE POWER STATION UNIT 3

DOMINION NUCLEAR

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Dominion®

October 27, 2008

Mr. Dennis P. Weakland
Materials Reliability Program - EPRI
c/o Jennifer Ma
Administrative Assistant
ANT, MRP & SGMP
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Palo Alto, CA 94304

Memo No. RA-08-026
NLOS/GAW R0

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
MRP-139 DEVIATION NOTIFICATION

In accordance with the Nuclear Energy Institute (NEI) Guideline for the Management of Materials Issues (NEI 03-08, Rev. 1), Dominion Nuclear Connecticut, Inc. (DNC) is providing a report supporting the deviation from the requirements of Electric Power Research Institute (EPRI) Materials Reliability Program (MRP): Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139) at Millstone Power Station Unit 3 (MPS3).

The deviation report was approved by senior management on September 29, 2008 and is included as an enclosure to this letter. Specifically, the deviation relates to the mandatory visual examination requirements contained in the MRP-139 Table 6-2. It is expected that this deviation will continue while the MRP-139 requirement remains in effect, or until the locations are mitigated to prevent propagation of potential primary water stress corrosion cracking (PWSCC).

If you have any questions regarding this report, please contact Mr. Geoffrey Wertz at (804) 273-3572.

Sincerely,

J. Alan Price
Vice President – Nuclear Engineering

Enclosure: TECHNICAL EVALUATION for Technical Justification for Deviation from Mandatory Requirements of MRP-139, Millstone Unit Three, M3-EV-08-0018 Rev. 0, September 25, 2008.

Memo No. RA-08-026
MRP-139 Deviation Notification
Enclosure

Enclosure

TECHNICAL EVALUATION

for

Technical Justification for Deviation from Mandatory Requirements of

MRP-139

Millstone Unit Three

M3-EV-08-0018

Rev. 0

9/25/08

MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.

QA ☒ Non-QA ☐

DB or LB document change required? yes ☐ no ☒

TECHNICAL EVALUATION

for

Technical Justification for Deviation from Mandatory Requirements of MRP-139

Millstone Unit Three

M3-EV-08-0018

Rev. 0

9/25/2008

Total Number of Pages = 36

Glenn Gardner Glenn A Gardner Preparer	9/26/2008 Date
Robert Schonenberg Robert Y. Schonenberg Independent Reviewer	9/26/2008 Date
Martin Van Haltern / Martin L Van Haltern Engineering Approver	9/26/2008 Date

Additional Concurrence per NEI 03-08 Addendum E Rev. 3

William McBrine GM per telecon – See Attach. 4 Independent Materials Expert – Altran Solutions	9/26/2008 Date
Alan Price Responsible Dominion Vice President	9-29-08 Date

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1.0 PURPOSE

This technical evaluation (TE) documents the technical justification for Millstone Unit 3 (MPS3) to deviate from certain bare metal visual examination (VE) requirements of MRP-139 [1], the industry-endorsed guideline for management of Alloy 600 issues on piping and nozzle butt welds. The TE is intended for independent materials expert concurrence and transmittal to the Materials Reliability Program (MRP) for notification in accordance with the industry initiative on materials, NEI 03-08 [2].

2.0 BACKGROUND

2.1 Materials Aging Issue

Primary Water Stress Corrosion Cracking (PWSCC) of nickel-based alloys has been an on-going industry issue for several years. The cracking occurs in susceptible materials when subjected to high stress levels in the PWR reactor coolant environment [1]. The susceptible materials include weld filler materials A182/A82, which are utilized at Millstone Unit 3 to weld the stainless steel safe-ends to the reactor vessel nozzles. Both inlet (RCS cold leg) and outlet (RCS hot leg) nozzles are potentially affected by PWSCC at the nozzle to safe end welds. The subject weld joints include A182 buttering on the ferritic vessel nozzle and A182 weld filler between the buttering and the forged stainless steel safe end. References [5] and [6] provide greater detail on these locations.

2.2 Applicable MRP-139 Requirements

The applicable MRP-139 requirements for managing PWSCC at the nozzle welds are presented in the Table 1 on the next page along with the current inspection status, just prior to the 3R12 refueling outage (RFO) in Fall 2008. MPS3 is on an 18 month refueling cycle.

Table 1 - MRP-139 Inspection Requirements

	Hot Leg (Outlet)		Cold Leg (Inlet)	
	MRP Requirements Tbl 6-1 Cat. D and Tbl 6-2 Cat. J	Current Status & Next req'd inspection	MRP Requirements Tbl 6-1 Cat. E and Tbl 6-2 Cat. K	Current Status & Next req'd inspection
Volume-tric (UT)	Every 5 yrs	3R11, Spring 07 3R14, Fall 11	Every 6 yrs	3R11, Spring 07 3R15, Spring 13
Bare Metal Visual Examina-tion (VE)	Every RFO except ones with Volumetric	Not inspected 3R12, Fall 08	Within 3 RFO (4.5 yrs) of volumetric exam	Not inspected 3R12, Fall 08

The tabular listing of requirements is simplified but presents MRP-139 schedule requirements accurately. The current inspection status shows that the subject welds were UT inspected from the ID during 3R11 in the spring, 2007. No indications were recorded. At issue is the fact that nozzle inaccessibility prevents bare metal visual examinations as required by MRP-139 Table 6-2 for visual examination categories J and K.

2.3 Millstone 3 Unique Design Features and Nozzle Accessibility

The vessel nozzle accessibility for Millstone 3 is very difficult because of the insulation package design at the nozzles. The insulation package comprises at least 14 heavy blocks weighing from 200 lbs to 1200 lbs each, bolted in place, in a very restricted location under the pit seal of the reactor vessel flange. A sketch of the package is shown in Attachment 2 on page 10. Scaffolding must be erected and each of the blocks needs to be rigged in and out. Removal of these blocks to permit the bare metal visual examination is estimated to require 105 work hours per nozzle, with a dose impact of 3.69 Rem per nozzle. The total dose impact for examination of the eight vessel nozzles is approximately 29.5 Rem.

2.4 Previous Evaluations

Technical Evaluation M3-EV-05-0024 [6] performed a similar evaluation for the initial bare metal visual examination of the nozzles required prior to the issuance of MRP-139. The TE documents an extensive review of original fabrication radiography of the nozzle to safe end welds. The review showed that, "For the nozzle to safe end welds, this review did not show any unusual results. All the welds had some porosity and slag inclusions but they were within acceptable limits. There were not any multiple reader sheets or repair weld numbers indicating weld repairs."

Technical Evaluation M3-EV-07-0026 [5], in addition to mapping out a mitigation plan for A600 locations, documents a complete listing of them along with any repair records available from Westinghouse records. Records showed only minor local repairs were performed.

2.5 References:

1 "Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guidelines (MRP-139)", Technical Report 1010087, EPRI, Palo Alto, CA: 2005. (retrievable from Portal – Virginia)

2 NEI 03-08, "Guideline for the Management of Materials Issues", Nuclear Energy Institute (NEI), Rev. 1 dated April 2007, with Addendum E Rev. 3 dated April 2008

3 CR-08-07092, "Millstone Unit 3 Can Not Do a Mandatory Requirement of MRP-139", initiated 6/18/2008

4 LTR-PAFM-08-127 Rev. 2, "Technical Justification for Deviation from MRP-139 Visual Inspection Schedules for Millstone Unit 3 Reactor Vessel Inlet and Outlet Nozzles", dated July 2008, © 2008, Westinghouse Electric Company LLC (Attachment 3)

5 M3-EV-07-0026 Rev. 0, "Technical Evaluation For The Control And Remediation Plan For Alloy 600 MPS 3", dated 6/22/2007

6 M3-EV-05-0024 Rev. 0, "Justification for the Deferral of Visual Examination of the Millstone Unit 3 Reactor Vessel Nozzle to Safe End Welds", dated 6/28/2007

7 ASME BPV Code Section XI, 1989 Edition, no Addenda

8 ASME BPV Code Section XI, 1998 Edition with 2000 Addenda

3.0 DISCUSSION

In summary of the issue at hand, MRP-139 requires bare metal visual examination of the nozzle to safe end welds of both the inlet and outlet of the RPV during 3R12 in Fall 2008. These are mandatory requirements under NEI 03-08. However in view of the almost 30 Rem dose impact of the inspections, ALARA principles compel an alternative approach unless the examinations provide an essential increment of assurance and safety that cannot be otherwise obtained. As a result, MPS3 has developed a justification for waiving the visual examinations while concurrently increasing volumetric inspection frequency, thereby achieving the same objective and intent of the original MRP-139 requirement.

The basis and intent of the visual examination requirements in MRP-139 are discussed in Section 6.10 and 6.11 for examination categories J and K respectively. Section 6.10.3 states "Visual examination capable of detecting any leakage must be performed in lieu of UT inspections." Section 6.11.3 has a similar statement for category K welds. Visual examinations are required only when UT examinations are not performed. Therefore the intent of the examination is to detect leakage, as a supplement to the primary strategy of relying on volumetric examinations to

confirm the absence of initiated flaws. In effect, the visual examinations address any uncertainties regarding the possibility of not detecting existing flaws and of crack growth rates for the PWSCC mechanism. The approach of this deviation is that such uncertainties may equivalently be addressed by a higher inspection frequency for the volumetric examinations.

Therefore, in lieu of the required visual examination schedule MPS3 plans to rely on volumetric examinations that will be performed on a schedule consistent with the results of a flaw tolerance evaluation. A table of the inspection plan is provided below and is justified in the following text.

Table 2 - Comparison of MRP-139 and MPS3 Inspection Plan

	Hot Leg (Outlet)		Cold Leg (Inlet)	
	MRP Requirements Tbl 6-1 Cat. D and Tbl 6-2 Cat. J	Next and subsequent inspections	MRP Requirements Tbl 6-1 Cat. E and Tbl 6-2 Cat. K	Next and subsequent inspections
Volume-tric (UT)	Every 5 yrs	3R13, Spring 10 and every other RFO (every 3 yrs)	Every 6 yrs	3R14, Fall 12 and every third RFO (every 4.5 yrs)
Bare Metal Visual Examin-ation (VE)	Every RFO except ones with Volumetric	Not required	Within 3 RFO (4.5 yrs) of volumetric exam	Not required

As shown in the above table, the planned volumetric (UT) inspection schedule is at a greater frequency than the generic requirement of MRP-139, compensating for the lack of visual examinations in intervening outages. This schedule will be followed until revised due to mitigation of the affected welds or being superseded by regulatory action. The basis for this schedule is documented in the flaw tolerance evaluation performed by Westinghouse [4] and included as attachment 3 to this TE. The Westinghouse evaluation is discussed below.

The flaw tolerance evaluation postulates an initial flaw and projects its subsequent growth in the interval between examinations based on accepted flaw growth correlations and the limits of flaw stability identified in ASME Section XI IWB-3640. The 1989 Edition [7] is the basis for the current ISI program at MPS3, while the 1998 Edition [8], which is approved by the NRC, is used for the flaw tolerance evaluation. The acceptability of a flaw tolerance evaluation as a basis for an alternative to the MRP-139 inspection schedule is based on the example of Section XI Appendix L acceptance of flaw tolerance for actual flaws, and its prior use in similar deviation reports submitted to the MRP.

The initial flaw assumption for the flaw tolerance evaluation relies on having performed a recent volumetric examination, with no recordable indications, performed in accordance with qualified

UT techniques and techniques that meet MRP-139 Section 5.1.5 coverage requirements. In the 3R11 examinations there were no exceptions to coverage requirements, and none are expected in the future. Based on the clean examination results with no detected flaws, a postulated initial circumferential flaw with 10% through-wall depth and limited length is assumed. The axial flaw length assumed is governed by the width of the weld. The assumed stress field with no major repairs was assumed based on the lack of such repairs for MPS3, as discussed in Section 3.4 of this TE. The flaw growth correlation used is referenced to NUREG/CR-6964 and is consistent with the MRP-139 recommendation in Section 2.6.2 for A182 materials. The uprated reactor power RCS temperatures are conservatively used in the flaw growth analysis.

The results of the flaw tolerance evaluation are summarized in the flaw growth limit curves contained in Attachment 3. Figure 3-1 shows that the axial flaw growth governs for the inlet (cold leg) nozzle but is not limiting for long periods up to 72 months. For conservatism and to limit the deviation from MRP-139, a limit of 54 months (4.5 years) inspection interval is specified. For the outlet nozzle (hot leg) the circumferential flaw governs and the higher temperature reduces the allowable inspection interval to less than 46 months. For conservatism a 36 month (3.0 years) inspection interval is specified for this nozzle.

In summary, the plant specific flaw tolerance analysis shows with reasonable margin that the selected inspection frequencies for the inlet and outlet nozzles will ensure that an initiating flaw will not propagate to the extent that IWB-3640 limits are exceeded. In addition, it shows even greater margin against propagation to pressure boundary leakage. It is only this through-wall condition that is detectable by bare metal visual examination. Therefore, an alternative that waives visual examinations for times prior to challenging IWB-3640 limits does not introduce any significant increment of risk, while allowing a nearly 30 Rem reduction in personnel exposure. It is thus a justified deviation to MRP-139 requirements.

As a final remark, this evaluation and notification does not meet the usual MRP expectation regarding timeliness of notification. However the original examination plan for 3R12 had been developed under the assumption that the NRC would soon issue a revision to 10CFR 50.55a requiring inspections in accordance with ASME Code Case N-722, plus additional stipulations that would accompany the rulemaking. Since the mandated Code Case would have overriding effect on MRP-139, a relief request was prepared in anticipation of this new rule, and no deviation would be required per MRP-139 Section 5.1.7. However, the NRC issuance of the rule change was delayed beyond its original scheduled date such that there is no assurance that review of the proposed relief request would be completed prior to the Fall 2008 outage. Therefore the plan to seek a relief request was modified to instead develop the justification for a deviation from the MRP-139 mandatory visual examination requirements and provide a deviation report to the MRP in accordance with NEI 03-08 Addendum E. The late notification to the MRP is therefore unavoidable.

4.0 SAFETY-SIGNIFICANCE

This technical evaluation performs an evaluation only and does not implement any change. Since there is no required change to the Technical Specifications, there is no change to the facility, there is no change to a procedure, and there is no test involved with this technical evaluation, no 50.59 screen is required.

5.0 CONCLUSION

Based on the discussions and information presented, the following is concluded.

- Visual examination of the MPS3 vessel nozzle welds would entail nearly 30 Rem dose impact to personnel and could be avoided by a justified deviation to MRP-139
- Elimination of visual examinations are compensated by an increased frequency of volumetric examinations
- A plant specific flaw tolerance evaluation demonstrates that maximum flaw growth during the period between volumetric inspections will remain with ASME Code limits
- The objective and intent of MRP-139 visual inspection requirements for the nozzles are satisfied by the alternative volumetric inspection plan

Therefore this deviation from the mandatory visual examination requirements is justified for the MPS3 RPV nozzle welds.

6.0 ATTACHMENTS

1. Independent Reviewer's Comment Sheet (1 page)
2. Sketch of RPV Nozzle Insulation Package Design (1 page)
3. Flaw tolerance evaluation LTR-PAFM-08-127 Rev. 2 (Westinghouse non-proprietary) (19 pages)
4. Independent materials expert concurrence opinion letter (6 pages)

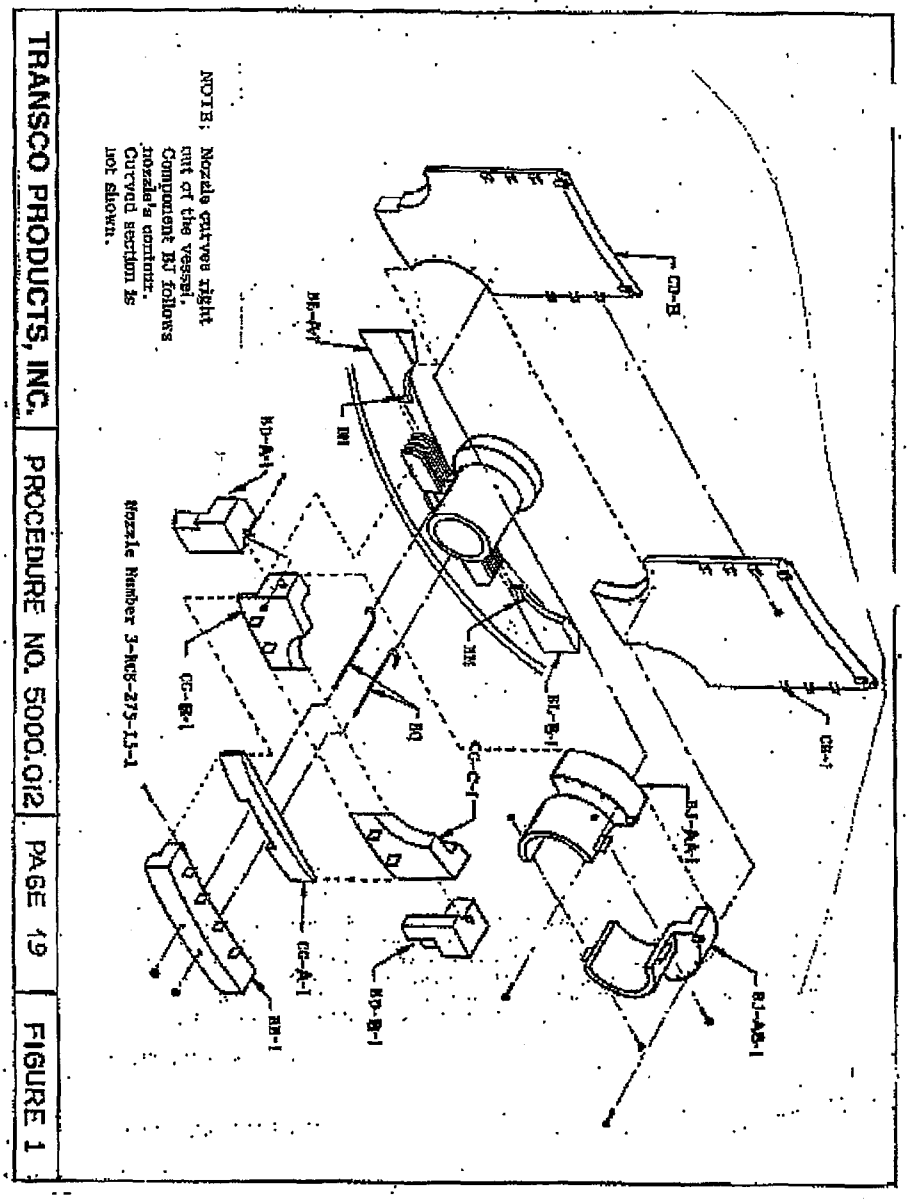
Independent Reviewer Comment and Resolution Sheet(s)

(ER/EV) No. M3-EV-05-0016 Rev. 0

Page 9 of 10

Independent Reviewer: Robert Schonenberg Date

Comment No.	ER/EV Section	Comment
1	Miscellaneous clarifications	Comments incorporated R.Y. Schonenberg 9/26/2008


CG-E
CG-F
BJ-AA-1
BJ-AB-1
BJ-B-1
CG-C-1
BJ
BJ
BD-B-1
CG-A-1
BD-1
CG-B-1
BD-A-1
BD
BS-A-1
NOTE: Nozzle curves right out of the vessel. Component BJ follows nozzle's contour. Curved section is not shown.
Nozzle Number 3-NCS-275-15-1
TRANSCO PRODUCTS, INC. | PROCEDURE NO. 5000.012 | PAGE 19 | FIGURE 1

Reference 4 -- Flaw Tolerance Evaluation

(19 pages follow)

LTR-PAFM-08-127 Revision 2

Technical Justification for Deviation from MRP-139 Visual Inspection Schedules for Millstone Unit 3 Reactor Vessel Inlet and Outlet Nozzles

September 2008

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Approved: S. A. Swamy*, Manager, Piping Analysis and Fracture Mechanics

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Westinghouse

Revision Record

Revision	Date	Description
0	August 2008	Original Issue
1	September 2008	Incorporate third party review comment by revising the technical justification for the assumed circumferential flaw aspect ratio in Section 2.3
2	September 2008	Incorporate Westinghouse comment on the third party review comment by revising the technical justification for the assumed circumferential flaw aspect ratio in Section 2.3 with concurrence from Dominion.

1.0 Introduction

Recent field experiences and the potential for Primary Water Stress Corrosion Cracking (PWSCC) at the Alloy 82/182 dissimilar metal (DM) butt welds require reassessment of the examination frequency and the overall examination strategy for these butt welds. MRP-139 (Reference 1) provided the inspection and evaluation guidelines for the primary system piping dissimilar metal butt welds. Millstone Unit 3 had performed a volumetric and 100% surface examination of the reactor vessel inlet and outlet nozzle to safe end dissimilar metal butt welds during the Spring 2007 outage and no indications were detected. For the butt welds at the outlet nozzles, since they are being exposed to the hot leg temperatures, are not made of PWSCC resistant material and also have not been mitigated, visual inspection is required per MRP-139 in every outage when volumetric examinations are not being performed, until these butt welds are replaced or mitigated. A less frequent visual inspection schedule is required for the inlet nozzles per MRP-139 due to the lower normal operating temperature at these nozzles.

Flaw tolerance analyses have been performed for the Millstone Unit 3 reactor vessel inlet and outlet nozzle DM welds in order to provide technical justification for deviating from the MRP-139 visual inspection requirements, by not performing visual inspection of the reactor vessel inlet and outlet nozzle butt welds for at least two operating cycles (36 months). The following provides a discussion of the methodology, results and conclusion of the flaw tolerance analysis for both nozzles.

2.0 Methodology

2.1 Maximum End-of-Evaluation Period Flaw Size

The maximum end-of-evaluation period flaw sizes for axial and circumferential inside surface flaws at the Alloy 82/182 welds of the inlet and outlet nozzle are determined using the IWB-3640 evaluation procedure and acceptance criteria in the ASME Section Xi Code (Reference 2) including the use of Z-factor for flux welds. The nozzle geometry (Reference 3) for the reactor vessel nozzles is shown in Table 2-1. The piping reaction loads from various loading conditions that are used in determining the most limiting end-of-evaluation period flaw sizes are summarized in Appendix A and taken from References 3, 4, and 5.

Table 2-1

Millstone Unit 3 Reactor Nozzle Geometry and Operating Parameters (Reference 3)

	Inlet Nozzle	Outlet Nozzle
Outside Diameter (in)	32 $^{15}/_{32}$	34 $^{7}/_{32}$
Inside Diameter (in)	27 $^{15}/_{32}$	28 $^{31}/_{32}$
Thickness (in)	2.500	2.625
Normal Operating Temperature (°F)	556.4	622.6

2.2 PWSCC Crack Growth Analysis

The Millstone Unit 3 reactor vessel inlet and outlet nozzle to safe end dissimilar metal weld regions are made of nickel based alloys. This nickel based alloy material (Alloy 82/182) is susceptible to PWSCC crack growth mechanism. The PWSCC crack growth rate used in the crack growth analysis is based on the EPRI recommended crack growth curves for Alloy 182 material (Reference 6) and shown below.

$$\frac{da}{dt} = \exp\left(-\frac{Q_g}{R}(1/T - 1/T_{ref})\right)\alpha(K)^{\beta}$$

where:

$\frac{da}{dt}$	=	Crack growth rate in m/sec
Q_g	=	Thermal activation energy for crack growth =130 kJ/mole (31.0 kcal/mole)
R	=	Universal gas constant = 8.314 x 10^{-3} kJ/mole-K (1.103 x 10^{-3} kcal/mole-°R)
T	=	Absolute operating temperature at the location of crack (K or °R)
T_{ref}	=	Absolute reference temperature used to normalize data = 598.15 K (1076.67°R)
α	=	Crack growth amplitude = 1.50 x 10^{-12} at 325°C (617°F)
β	=	Exponent = 1.6
K	=	Crack tip stress intensity factor (MPa$\sqrt{m}$)

It should be noted that the PWSCC crack growth mechanism is applicable only to the inside surface flaws since they are exposed to the primary water environment. The stresses used for PWSCC evaluations included normal operating condition piping reaction loads, pressure, and residual stresses at the DM welds. The normal operating temperatures for the inlet and outlet nozzles are 556.4°F and 622.6°F respectively,

The impact of fatigue crack growth mechanism is considered in the flaw tolerance analysis. Fatigue crack growth is negligible, especially for short plant operation duration (2 to 3 refueling cycles) when compared to that due to PWSCC because the locations of interest at the inlet and outlet nozzles are not subjected to any significant thermal transient loadings.

The residual stresses considered in the analyses were based on the reactor vessel nozzle residual stress profiles from Reference 7 for the case with no inside surface weld repair. This is acceptable since a review of all the available manufacturing records for the reactor vessel did not show any significant inside surface weld repairs made to either the inlet or outlet nozzle dissimilar metal welds (Reference 8).

Using the applicable stresses at the DM welds, the crack tip stress intensity factors can be determined based on the stress intensity factor expressions from References 9 and 10. The through-wall stress distribution profile is represented by a cubic polynomial:

$$\sigma(x) = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

where:

A_0, A_1, A_2, and A_3 are the stress profile curve fitting coefficients,

x is the distance from the wall surface where the crack initiates, and

σ is the stress perpendicular to the plane of the crack.

The stress intensity factor calculations for semi-elliptical inside surface flaws with various aspect ratios (flaw length/depth) for axial and circumferential flaws are performed. The influence coefficient at any points on the crack front can be obtained by using an interpolation method. The crack tip stress intensity factors can be expressed in the general form as follows:

$$K_I = \left(\frac{\pi a}{Q}\right)^{0.5} \sum_{j=0}^{3} G_j(a/c, a/t, t/R, \Phi)\, A_j a^j$$

where:

a: Crack Depth

c: Half Crack Length Along Surface

t: Thickness of Cylinder

R: Inside Radius

Φ : Angular Position of a Point on the Crack Front

G_j: G_j is influence coefficient for j^{th} stress distribution on crack surface (i.e., G_0, G_1, G_2, G_3).

Q: The shape factor of an elliptical crack, which is the square of the complete elliptical integral of the second kind or

Shape Factor $=[\int_0^{\pi/2}(\cos^2\Phi+\frac{a^2}{c^2}\sin^2\Phi)^{1/2}\,d\Phi]^2$. Q is approximated by:

$Q = 1 + 1.464(a/c)^{1.65}$ for $a/c \leq 1$ or $Q = 1 + 1.464(c/a)^{1.65}$ for $a/c > 1$.

Once the crack tip stress intensity factors are determined, PWSCC crack growth calculations can be performed using the crack growth rate discussed in Section 2.2 for the applicable normal operating temperature.

2.3 Maximum Undetected Flaw size

The initial flaw size used in the flaw tolerance analysis is assumed to be the maximum undetected flaw size since no indications were detected during the Spring 2007 volumetric and surface examination. The maximum undetected flaw depth is assumed to be 10% of the wall thickness. This assumed flaw depth is similar to the in-service inspection acceptance criteria in Table IWB-3514-2 of the ASME Section XI Code for returning components into service and therefore is a conservative and reasonable assumption. An aspect ratio (flaw length/depth) of 2 is assumed for the axial flaw since PWSCC is limited to the width of the A82/182 weld. For the circumferential flaw, an aspect ratio (AR) of 6 is assumed. As for the circumferential flaw, an initial flaw depth of 0.25 inch (10% through wall) and initial flaw length of 1.5 inches (aspect ratio of 6) is conservatively assumed for the inlet nozzle. Assuming the same aspect ratio for the outlet nozzle, the initial flaw length is assumed to be 1.58 inch. Since no detectable flaws were found in the dissimilar metal welds of these nozzles during the spring 2007 volumetric examination, it is considered highly unlikely, with a qualified volumetric examination, a flaw of this size would go undetected.

3.0 Flaw Tolerance Analysis Results

Figures 3-1 to 3-4 display the maximum allowable initial flaw size for the axial and circumferential flaws at the nozzle to safe end Alloy 82/182 welds for the Inlet and Outlet nozzles based on the IWB-3640 acceptance criteria. The horizontal axis displays the flaw depth to length ratio or the inverse of the flaw aspect ratio. The vertical axis shows the flaw depth to wall thickness ratio (a/t). The flaw evaluation chart displays allowable flaw size curves for plant operation duration up to 54 months. If the flaw parameters of a given flaw fall below the allowable flaw size curve for a given plant operation duration, then the flaw will not grow to the maximum end-of-evaluation period allowable flaw size within that plant operation duration. For comparison purposes, the maximum undetected

flaw size as discussed in Section 2.3 is also shown in Figures 3-1 to 3-4 to show the available margins for this assumed initial flaw size.
Figures 3-5 to 3-8 display the maximum allowable initial flaw size for the axial and circumferential flaws at the nozzle to safe end Alloy 82/182 welds for the Inlet and Outlet nozzles based on leakage instead of limit load failure. Leakage is assumed to occur once the initial inside surface flaw becomes a 100% through-wall flaw. If the flaw parameters of a given flaw fall below the allowable flaw size curve for a given plant operation duration, then the flaw will not grow to a 100% through-wall flaw within that plant operation duration. For comparison purposes, the maximum undetected flaw size as discussed in Section 2.3 is also shown in Figures 3-5 to 3-8 to show the available margins for this assumed initial flaw. The margins shown are slightly larger than those based on the IWB-3640 acceptance criteria.

As shown in Figures 3-3 and 3-4, the flaw tolerance result for the outlet nozzle is more limiting and continued plant operation duration of only 36 months is acceptable for the assumed undetected flaw size. There is adequate margin for the inlet nozzle (Figures 3-1 and 3-2) for continued plant operation duration of 54 months. Additionally, this margin is demonstrated by the 72 month curves identified in the inlet nozzle flaw tolerance charts. Since no indications were detected during the Spring 2007 refueling outage, crack growth due to PWSCC for the maximum undetected flaw size would not reach the end-of-evaluation period allowable flaw size per IWB-3640 or result in leakage for continued plant operation of at least 36 months for the reactor vessel inlet and outlet nozzles.

PWSCC crack growth curves for the limiting reactor vessel outlet nozzles are shown in Figures 3-9 to 3-10 for axial (AR=2) and circumferential flaw (AR=6) respectively with the initial flaw size equals to the assumed maximum undetectable flaw size. The horizontal axis displays the service life in effective full power months (EFPM), while the vertical axis shows the flaw depth to wall thickness ratio (a/t). These curves demonstrated the service life required to reach the IWB-3640 acceptable flaw size and a 100% through-wall flaw. Based on the IWB-3640 end-of-evaluation period allowable flaw size, it would take at least 48 EFPM for an axial flaw (AR=2), with an initial flaw depth of a/t=0.10, to reach the end-of-evaluation period allowable flaw depth. For a circumferential flaw (AR=6) with the same initial flaw depth, it would take 46.2 EFPM to reach the end-of evaluation period allowable flaw depth. The service life required is therefore more than 2 operating cycles (36 months) at Millstone Unit 3. Also as illustrated in Figures 3-9 and 3-10, the service life required to reach 100% through-wall thickness is slightly longer.

Figure 3-1

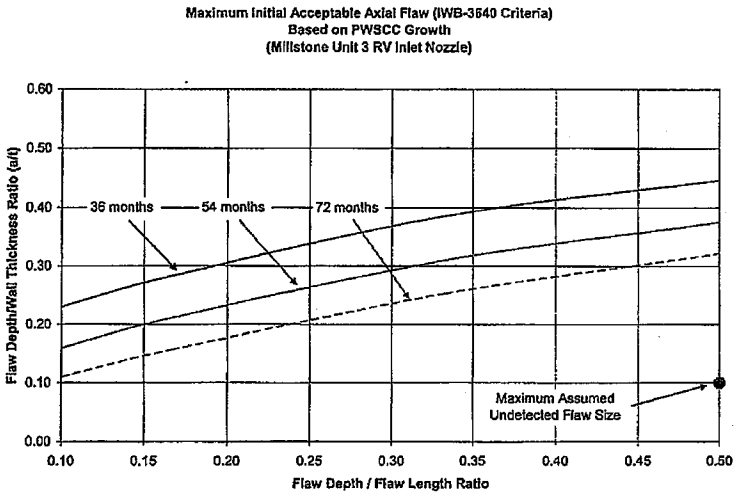

Figure 3-2

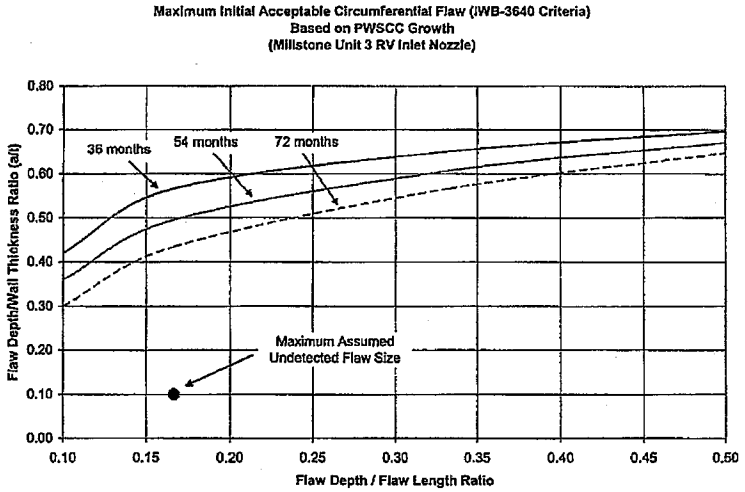

Figure 3-3

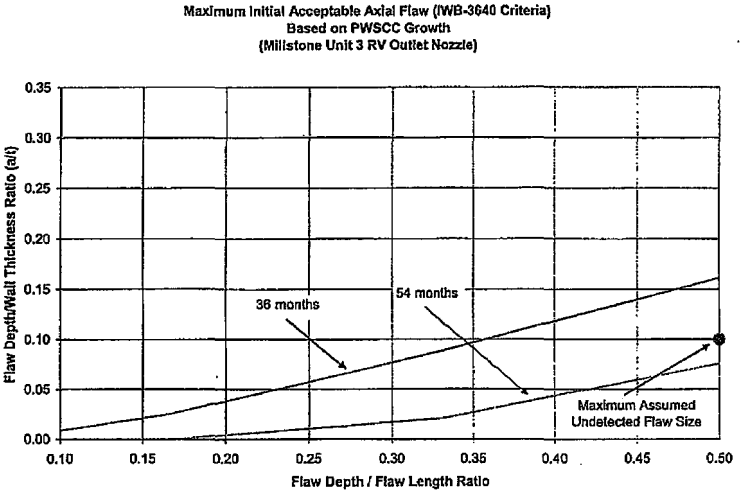

Figure 3-4

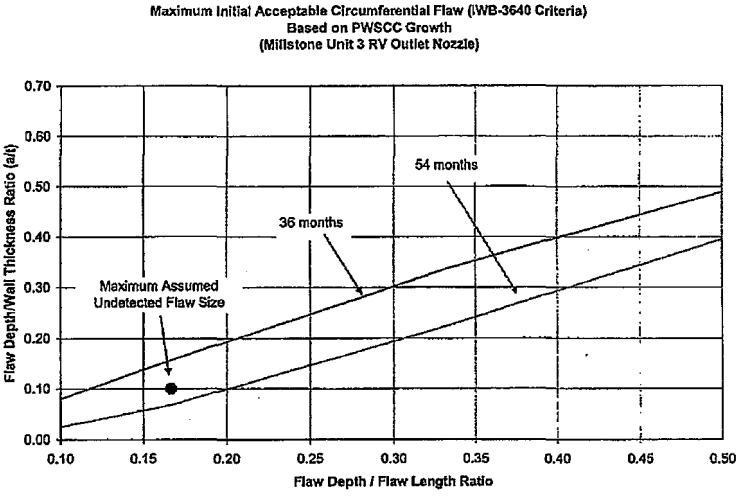

Figure 3-5

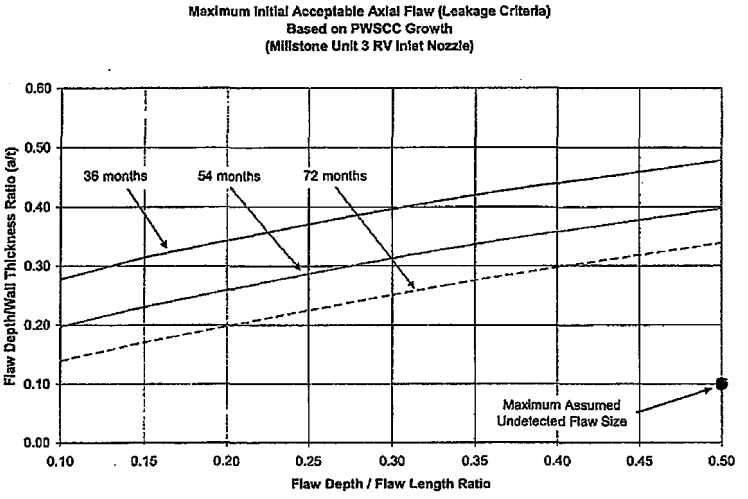

Figure 3-6

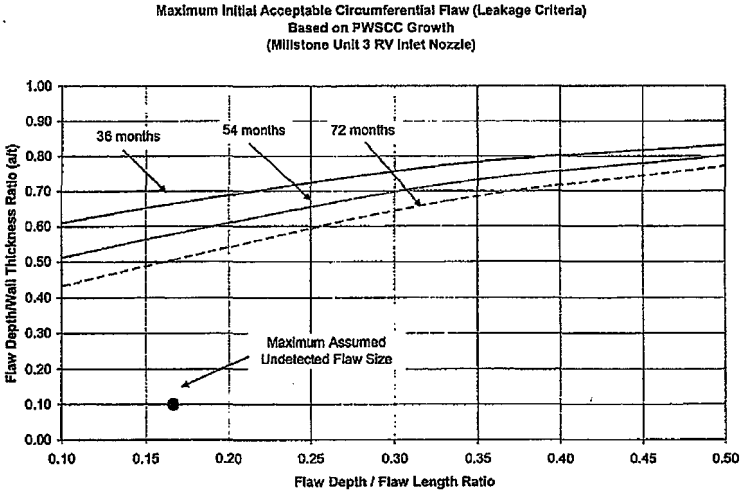

Figure 3-7

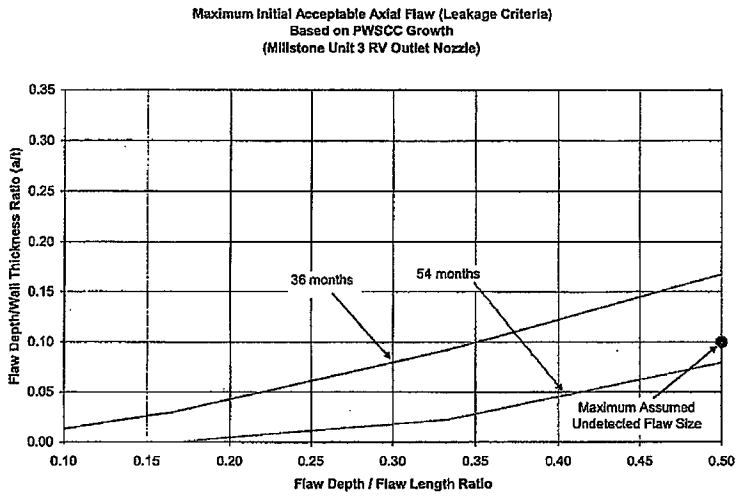

Figure 3-8

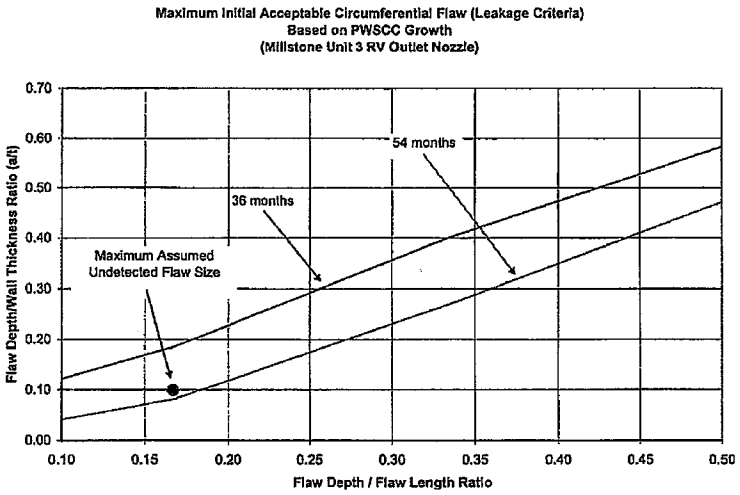

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Figure 3-9

PWSCC Axial Crack Growth Curves for Outlet Nozzle Alloy 82/182 Weld Region

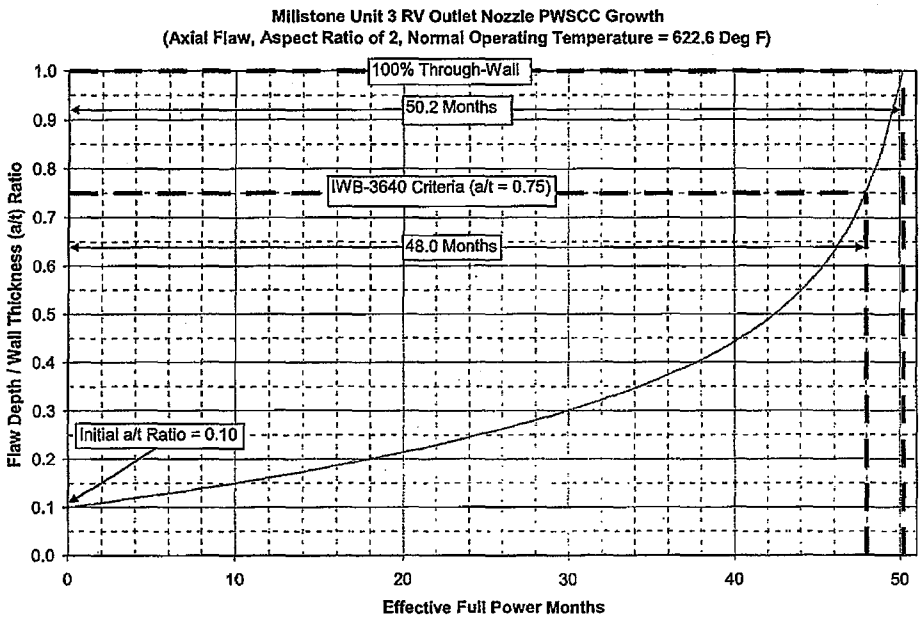
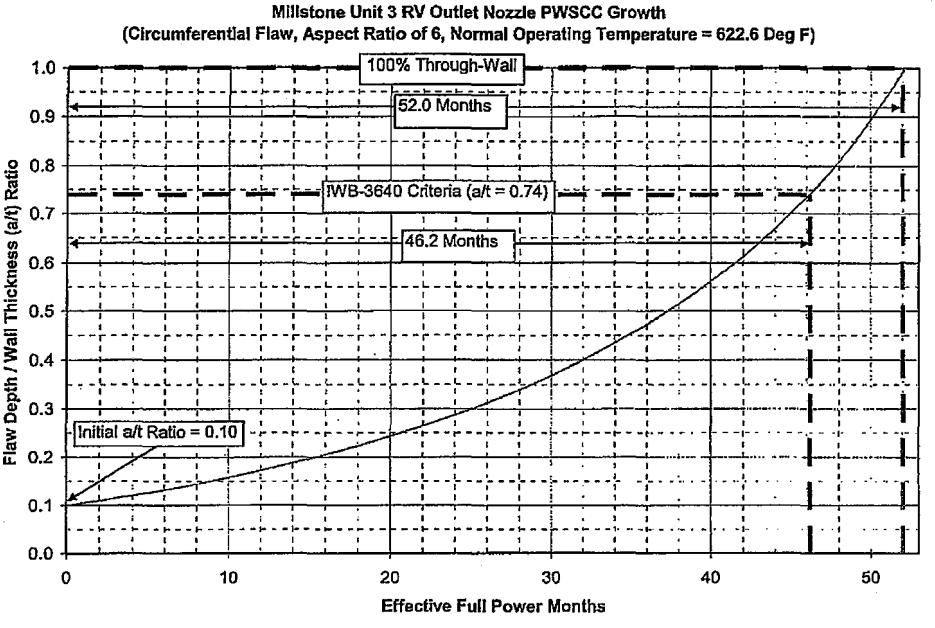


Figure 3-10

PWSCC Circumferential Crack Growth Curves for Outlet Nozzle Alloy 82/182 Weld Region



4.0 Discussion and Conclusion

The required visual inspection schedules for the inlet and outlet nozzles are shown in Table 6-2 of MRP-139. For the outlet nozzle, visual inspection is required in every outage when volumetric examinations are not being performed until the nozzle is being mitigated or replaced. The required volumetric examination for the outlet nozzle is every 5 years per Table 6-1 of EPRI Report MRP-139. Based on the MRP-139 volumetric examination schedule, the Millstone Unit 3 outlet nozzle would perform volumetric inspection every 3 refueling cycles since the refueling cycle interval for Millstone Unit 3 is 18 months. Based on the flaw tolerance results shown in Figures 3-3 and 3-4, it is acceptable to deviate from the MRP-139 visual inspection schedule by performing a visual inspection every other refueling outage when volumetric examinations are not being performed instead of every outage.

For the inlet nozzle, visual inspection is required once every three refueling cycles until the nozzle is being mitigated or replaced. The required volumetric examination for the inlet nozzle is every 6 years per Table 6-1 of EPRI Report MRP-139. Based on this volumetric examination schedule, the Millstone Unit 3 inlet nozzle would be inspected every four refueling outages. Per Table 6-2 of MRP-139, deterministic analysis can be used as a basis to allow the inlet nozzle DM welds to be visually examined at a frequency less than once every three refueling outages. Based on the flaw tolerance analysis performed, the results shown in Figures 3-1 and 3-2 demonstrated that there is adequate margin to support deviation from the MRP-139 visual inspection schedule for the inlet nozzle.

In summary, since no indications were detected during the Spring 2007 refueling outage, crack growth due to PWSCC for the maximum undetected flaw size would not reach the end-of-evaluation period allowable flaw size per IWB-3640 or result in leakage for continued plant operation duration of at least 36 months for the inlet and outlet nozzles. Based on the results of the flaw tolerance analysis, it is technically justified to seek a less frequent visual inspection schedule than those required in MRP-139 for both the reactor vessel inlet and outlet nozzle dissimilar metal weld regions.

5.0 References

1. Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline (MRP-139), EPRI, Palo Alto, CA: 2005. 1010087. (EPRI Proprietary Document)
2. Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler & Pressure Vessel Code, Section XI, 1998 Edition through 2000 Addenda.
3. Dominion Document 25212-ER-08-0024 dated 5/14/2008, "Reactor Vessel Nozzle Evaluation Input for NEU-08-29" (Dominion Proprietary Document)
4. Stone and Webster Engineering Corporation (SWEC) Document No. 12179-NP(B)-x7001A – "Pipe Stress Analysis: Reactor Coolant System Piping and Associated Branch Connections loops 1 and 3 ASME III Code Class 1 and 2" – Revision 0 – Boston, MA.
5. Stone and Webster Engineering Corporation (SWEC) Document No. 12179-NP(B)-x7002A – "Pipe Stress Analysis: Reactor Coolant System Piping and Associated Branch Connections loops 2 and 4 ASME III Code Class 1 and 2" – Revision 0 – Boston, MA.
6. NUREG/CR-6964 ANL-07/12, Crack Growth Rates and Metallographic Examinations of Alloy 600 and Alloy 82/182 from Field Components and Laboratory Materials Tested in PWR Environments.
7. Materials Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assessment for US PWR Plant Designs (MRP-113), EPRI, Palo Alto, CA: 2005. 1009549. (EPRI Proprietary Document)
8. Westinghouse Letter LTR-PCAM-07-21, "Millstone Unit 3 – PWROG PA-MSC-0233 – Task 2 Customer Deliverable – Primary Pressure Boundary Alloy 600/82/182 Fabrication Detail," February 23, 2007. (Westinghouse Proprietary Document)
9. Raju, I. S. and Newman, J. C., "Stress Intensity Factor Influence Coefficients for Internal and External Surface Cracks in Cylindrical Vessels," ASME Publication PVP. Volume 58, 1982, pp. 37-48.
10. Mettu, S. R., Raju, I. S., and Forman, R. G., NASA Lyndon B. Johnson Space Center report no. NASA-TM-111707, "Stress Intensity Factors for Part-through Surface Cracks in Hollow Cylinders," in Structures and Mechanics Division, July 1992.

Appendix A

Millstone Unit 3 Reactor Vessel Inlet and Outlet Nozzle Loads

Table A-1

Inlet Nozzle Loads – Table 1 of 3

	Loading	Reference	Forces (kips)	Moments (in-Kips)		
			Fx	Mx	My	Mz
Inlet Nozzle	Thermal	3, 4	+28 -35	0 -4905	+1621 -921	0 -8105
	Operating Pressure	3	1374	113	-316	-69
	Deadweight	3, 4	3	-401	1	-1108
	OBE Inertia	4	40	1354	2345	2526
	OBE SAM	4	39	71	112	181
	SSE Inertia	4	49	1350	2409	2356
	SSE SAM	4	61	108	177	274

Note: SAM = Seismic Anchor Motion

Table A-2

Inlet Nozzle Loads – Table 2 of 3

	Loading	Reference	Forces (kips)	Moments (in-Kips)		
			Fx	Mx	My	Mz
Inlet Nozzle	Thermal	3, 5	+34 -61	0 -4909	+1711 -1371	+7277 0
	Operating Pressure	3	1376	-305	533	-113
	Deadweight	3, 5	3	69	-37	-90
	OBE Inertia	5	42	1377	2150	2113
	OBE SAM	5	39	71	213	162
	SSE Inertia	5	53	1369	2189	1972
	SSE SAM	5	61	108	335	245

Note: SAM = Seismic Anchor Motion

Table A-3

Inlet Nozzle Loads – Table 3 of 3

	Loading	Reference	Forces (kips)	Moments (in-Kips)		
			Fx	Mx	My	Mz
Inlet Nozzle	Thermal	3, 5	+34 -61	0 -4909	+1711 -1371	+7277 0
	Operating Pressure	3	1376	-305	533	-113
	Deadweight	3, 5	3	69	-37	-90
	OBE Inertia	5	42	1377	2150	2113
	OBE SAM	5	39	71	213	162
	SSE Inertia	5	53	1369	2189	1972
	SSE SAM	5	61	108	335	245
	Break 9 – RHR	3	+245 -533	+913 -1132	+3909 -6865	+1705 -1003
	Break 10 - SI	3	+387 -500	+7756 0	+52699 -24678	0 -14413
	Break 11 - Surge	3	+74 -441	+927 -1101	+3630 -4790	+2164 -1436

Note: SAM = Seismic Anchor Motion

Table A-4

Outlet Nozzle Loads – Table 1 of 3

	Loading	Reference	Forces (kips)	Moments (in-Kips)		
			Fx	Mx	My	Mz
Outlet Nozzle	Thermal	3, 4	+35 -41	+43 -348	+3813 -2570	0 -13444
	Operating Pressure	3	1511	-27	341	-1095
	Deadweight	3, 4	2	29	-114	-728
	OBE Inertia	4	194	547	3813	3670
	OBE SAM	4	45	166	225	366
	SSE Inertia	4	254	630	3544	3351
	SSE SAM	4	71	251	358	554

Note: SAM = Seismic Anchor Motion

Table A-5

Outlet Nozzle Loads – Table 2 of 3

	Loading	Reference	Forces (kips)	Moments (in-Kips)		
			Fx	Mx	My	Mz
Outlet Nozzle	Thermal	3, 5	-27 +90	+738 0	+2071 -4889	0 +12545
	Operating Pressure	3	1509	70	-649	-237
	Deadweight	3, 5	0	-27	+50	-2550
	OBE Inertia	5	182	862	3488	3239
	OBE SAM	5	50	154	227	319
	SSE Inertia	5	236	965	3214	2956
	SSE SAM	5	80	234	360	483

Note: SAM = Seismic Anchor Motion

Table A-6

Outlet Nozzle Loads – Table 3 of 3

	Loading	Reference	Forces (kips)	Moments (in-Kips)		
			Fx	Mx	My	Mz
Outlet Nozzle	Thermal	3, 5	+90 -27	0 -738	+2071 -4889	+12545 0
	Operating Pressure	3	1509	70	-649	-237
	Deadweight	3, 5	0	-27	+50	-2550
	OBE Inertia	5	182	862	3488	3239
	OBE SAM	5	50	154	227	319
	SSE Inertia	5	236	965	3214	2956
	SSE SAM	5	80	234	360	483
	Break 9 – RHR	3	+803 -14	+3810 -3428	+34494 0	0 -48087
	Break 10 - SI	3	+492 -55	+1238 -1661	+2538 -4114	+5391 -5296
	Break 11 - Surge	3	+567 -274	+4086 -4250	+38733 0	+42348 0

Note: SAM = Seismic Anchor Motion

Independent Materials Expert Concurrence

W. McBrine, Altran Solutions

(5 pages follow)

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September 25, 2008
08-0419-L-001

Mr. Steven D. Janes
Dominion Nuclear Connecticut
Millstone Power Station
Rope Ferry Road (Route 156)
Waterford, CT 06385

SUBJECT: Transmittal of Altran Design Verification Report 08-0419-VR-001, Rev. 0, "Third Party Review of the Technical Justification for Deviation from MRP-139 Visual Inspection Schedule, Millstone Point Unit 3"

REF: Dominion Purchase Order 70187510, dated 9/11/2008.

Dear Mr. Janes:

Please find enclosed the original copy of Altran Design Verification Report 08-0419-VR-001, Rev. 0. This report documents the third party review that Altran performed on Dominion Nuclear Connecticut Technical Evaluation M3-EV-08-0018, Rev. 0, "Technical Evaluation for Technical Justification for Deviation from Mandatory Requirements of MRP-139, Millstone Unit Three."

Altran appreciates the opportunity to be of service to Dominion Nuclear. If you have any questions or comments, please do not hesitate to call Bill McBrine at (617) 204-1000.

Very truly yours,
ALTRAN CORPORATION

Alfred W. Chock, Jr.
Technical Lead – Mechanical Engineering

William J. McBrine
Technical Manager – Materials Engineering

Enclosure

ALTRAN

VERIFICATION REPORT

VR No.: 08-0419-VR-001 Project No. 08-0419 Page 1 of 4

Design, Analysis, Test, or Examination Verified: <n/a>

Document Verified:

Dominion Nuclear Connecticut Technical Evaluation M3-EV-08-0018, Rev. 0, "Technical Evaluation for Technical Justification for Deviation from Mandatory Requirements of MRP-139, Millstone Unit Three."

Method of Verification:

_x_Independent Review ___Alternate Calculation ___Testing Qualification

Summary of Verification

At the request of Dominion Nuclear Connecticut, Altran Solutions Corporation performed a third-party review of a technical justification for deviation from the MRP-139 visual inspection schedules for Millstone Point Unit 3 (MP3) reactor vessel input and output nozzles. The results of this review are discussed in the following sections.

Documents Reviewed

1. Dominion Nuclear Connecticut, "Technical Evaluation for Technical Justification for Deviation from Mandatory Requirements of MRP-139, Millstone Unit Three". Tech. Eval. No. M3-EV-08-0018, Rev. 0, September, 2008.
2. Westinghouse Letter LTR-PAFM-08-127 (Non-Proprietary), "Technical Justification for Deviation from MRP-139 Visual Inspection Schedules for Millstone Unit 3 Reactor Vessel Inlet and Outlet Nozzles", Rev. 2, September, 2008.

Background

Recent field experiences and the potential for Primary Water Stress Corrosion Cracking (PWSCC) at the Alloy 82/182 dissimilar metal (DM) butt welds require reassessment of the examination frequency and the overall examination strategy for these butt welds. EPRI MRP-139 provides the inspection and evaluation guidelines for the primary system piping dissimilar metal butt welds. For butt welds at the outlet nozzles, that are exposed to hot leg temperatures, are not made of PWSCC resistant material and also have not been mitigated, MRP-139 requires visual inspection is required at every outage when volumetric examinations are not being performed, until these welds are replaced or mitigated. A less frequent visual inspection schedule is required

for the inlet nozzles per MRP-139 due to the lower normal operating temperature at these nozzles.

MP3 performed a volumetric examination of the reactor vessel inlet and outlet nozzle to safe end dissimilar metal butt welds during the Spring 2007 outage. At that time, no indications were detected. To provide relief from the MRP-139 visual inspection schedule requirements, Westinghouse performed a flaw tolerance analysis (see Document 1) of the MP3 RV inlet and outlet nozzle DM welds. This analysis demonstrated that the next visual inspection of the reactor vessel inlet and outlet nozzle butt welds would not be necessary for at least two operating cycles (36 months).

Technical Approach

Altran's review of the two documents assessed the adequacy and presentation of the following:

- Criteria (i.e., applicability to the requirements of MRP-139)
- Methodology
- Selection of suitable input
- Tabulated results
- Conclusions of the evaluation.

Reference Documents

As part of the review process, the following documents were examined. These documents are either commercially available or comprise the design basis of Millstone Unit 3.

1. Electric Power Research Institute, *Material Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline (MRP-139)*, EPRI Report 1010087. (EPRI Proprietary Document). Palo Alto, CA: 2005.
2. American Society of Mechanical Engineers, "Rules for Inservice Inspection of Nuclear Power Plant Components", *ASME Boiler & Pressure Vessel Code*, Section XI, 1998 Edition through 2000 Addenda.
3. Argonne National Laboratory, *Crack Growth Rates and Metallographic Examinations of Alloy 600 and Alloy 82/182 from Field Components and Laboratory Materials Tested in PWR Environments*, NUREG/CR-6964. Argonne, IL: U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, 2008.
4. Nuclear Energy Institute, "Guidelines for the Management of Materials Issues", NEI 03-08, Rev. 1. Washington, DC: April 2007.
5. Electric Power Research Institute, *Material Reliability Program: Alloy 82/182 Pipe Butt Weld Safety Assesment for US PWR Plant Designs (MRP-113)*, EPRI Report 1007029. (EPRI Proprietary Document). Palo Alto, CA: 2004.

Conclusions

As a result of the review, Altran Solutions Corporation has made the following findings:

1. Altran concurs that the deviation set forth in Technical Evaluation M3-EV-08-0018 satisfies the objective and intent of MRP-139.
2. Altran further finds that the technical arguments in support of Technical Evaluation M3-EV-08 are satisfactory, and that they accurately incorporate the basis provided in Westinghouse Document LTR-PAFM-08-127.

Qualifications of Reviewers

The third-party review was conducted by William McBrine, PE, with contributions from Edmund Dunn, Sc.D. and Bahaa Elaidi, Ph.D. A short summary of team member qualifications is provided in Attachment A. Full professional resumes are available upon request.

Statement of Concurrence

Having performed a third-party review of Domnion Technical Evaluation M3-EV-08-0018, Rev. 0 in the role of Independent Materials Expert, Altran Solutions hereby states its concurrence with the technical evaluation and the results herein.

Mr. McBrine has affixed his endorsement as Independent Materials Expert to Millstone Technical Evaluation M3-EV-08-0018.

William J. McBrine, PE, Technical Lead	9/25/08 Date
Bahaa A. Elaidi, Ph.D., Contributor	9/25/08 Date
Edmund M. Dunn, Sc.D., Contributor	9/25/08 Date

ATTACHMENT A
SUMMARY OF QUALIFICATIONS OF REVIEWERS

William J. McBrine, PE
Technical Lead

William McBrine is the Technical Manager of the Materials Engineering Group at Altran Solutions. Mr. McBrine has 30 years of experience in the nuclear power industry with particular expertise in addressing structural integrity issues. He has extensive experience in the assessment of degraded mechanical components, including failure analysis, flaw evaluations and remaining life prediction. He has led projects investigating Alloy 600 issues including the prediction of SCC crack growth rate and influencing factors. Mr. McBrine also has extensive experience in stress analysis, fracture mechanics and qualifications to ASME B&PV Sections III and XI requirements.

Bahaa A. Elaidi, Ph.D.

Dr. Elaidi is the Technical Manager of Structural Engineering and Engineering Mechanics at Altran Solutions. He has over 25 years of experience in applied mechanics, failure analysis, and root cause evaluation, with a diverse background in analysis, inspection, and repair of civil and mechanical systems and components. Previous applicable work includes investigation of cracking in steam generator tubes, establishment of critical flaw sizes welded joints of piping and spent fuel canisters, failure analyses and life assessment of nuclear plant components, and analytical modeling of flaws and crack growth.

Edmund M. Dunn, Sc.D.

Edmund M. Dunn has over 30 years of experience in Materials Science and Engineering with core expertise in solidification metallurgy, brazing and welding. His experience includes Bettis Atomic Power Laboratory, and GTE Laboratories. While at Bettis, his work included studies on factors affecting stress corrosion cracking and weld hot cracking in reactor plant materials (Ni-Cr-Fe Alloy 600).

He has served as chair of a national committee, the TMS/AIME Solidification Committee and is the author or coauthor of numerous papers and five patents. His work has included materials selection, market evaluation, process improvement, and failure analysis. He is a member of ASM and TMS/AMIE. He has been Secretary of ASM International Boston Chapter.

Dr. Dunn received an Sc.D. in Materials Science and Engineering from MIT, a B.S. in Materials Engineering from RPI, and an MBA from the University of California at Berkeley.