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Millstone Power Station
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Dominion

JAN 24 2003

Docket Nos. 50-336
50-423
B18815

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

Millstone Power Station, Unit Nos. 2 and 3
Reply to Request for Additional Information
Related to NRC Bulletin 2002-01

In letters dated April 2, 2002,[1] and May 16, 2002,[2] Dominion Nuclear Connecticut, Inc. (DNC) submitted information in response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." On November 19, 2002,[3] the NRC requested additional information related to NRC Bulletin 2002-01. This additional information was due January 21, 2003, however an extension to January 24, 2003, was granted by the NRC to ensure information was included to fully answer all questions. The purpose of this letter is to provide that additional information.

There are no regulatory commitments contained within this letter.

Should there be any questions regarding this submittal, please contact Mr. Paul R. Willoughby at (860) 447-1791, extension 3655.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

cc: See next page

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(1) J. A. Price, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated April 2, 2002.

(2) J. A. Price letter to U.S. Nuclear Regulatory Commission, "Response to NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated May 16, 2002.

(3) R. B. Ennis, U.S. Nuclear Regulatory Commission, letter to J. A. Price, "Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," 60-day Response for Millstone Power Station, Unit Nos. 2 and 3, Request for Additional Information (TAC Nos. MB4555 and MB4556)," dated November 19, 2002.

A095

Attachments (2)

cc: H. J. Miller, Region I Administrator
R. B. Ennis, NRC Senior Project Manager, Millstone Unit No. 2
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
Millstone Senior Resident Inspector

Docket Nos. 50-336
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Attachment 1

Millstone Power Station, Unit Nos. 2 and 3

Reply to Request for Additional Information
Related to NRC Bulletin 2002-01
Millstone Unit No. 2

Reply to Request for Additional Information
Related to NRC Bulletin 2002-01
<u>Millstone Unit No. 2</u>

NRC Question 1

1) *Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, frequency of inspections, personnel qualifications, and degree of insulation removal for the examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel (RPV) bottom head).*

DNC Response:

Currently the site procedure for the inspection of components exposed to boric acid, C EN 109, "Inspection of Components Exposed to Boric Acid," has a table that lists the places to visually inspect during every refueling outage. This list, developed as a result of Generic Letter 88-05, primarily contains bolted connections. The alloy 600 components currently included in this list are the reactor vessel head penetrations, the pressurizer instrument nozzles and the pressurizer heater sleeves. The insulation has been left in place for these examinations in accordance with the ASME code. No Alloy 82/182 welds are currently included in C EN 109 to be specifically examined. There are currently no personnel qualification requirements in C EN 109 other than the examiner must be an Inservice Inspection Technician, which implies that they possess non-destructive examination (NDE) certifications. The boric acid inspection program at Millstone is currently being revised. Specific training and qualifications will become mandatory for the personnel performing the inspections as part of the revision.

Based upon industry experience, Millstone Unit No. 2 did a bare metal examination of all the pressurizer heater sleeves and pressurizer instrument nozzles during the last refueling outage in February, 2002. Two leaking heater sleeves were identified and sealed with Mechanical Nozzle Seal Assembly (MNSA) clamps. In addition Millstone Unit No. 2 performed an ultrasonic (UT) inspection of 100% of the reactor vessel head penetrations during the February, 2002, refueling outage. Three penetrations with shallow outer diameter (OD) cracking below the J-weld were repaired as described in DNC letters dated April 30, 2002,[4] and May 30, 2002.[5] In light of these cracks and leaks and other recent industry events, the Millstone Station procedures for boric acid corrosion control (BACC) are being revised. The locations of all the Alloy 600

(4) Letter from J. A. Price to U.S. Nuclear Regulatory Commission, "Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated April 30, 2002.

(5) Letter from J. A. Price to U.S. Nuclear Regulatory Commission, "Reply to Request for Additional Information Related to NRC Bulletin 2001-01," dated May 30, 2002.

penetrations and Alloy 82/182 welds at Millstone Unit No. 2 have been identified through two CE Owners Group (CEOG) programs (Tasks 1191 and 1142) and those that could possibly cause degradation of carbon steel are being added to the list of components to be examined. The frequency of inspection of the alloy 600 components and welds will be determined based upon an evaluation of the temperature to which the component or weld is exposed.

NRC Question 2

2) *Provide the technical basis for determining whether or not insulation is removed to examine all 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 the limitations to removal of insulation. Also include in your response action involving removal of insulation required in your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.*

DNC Response:

Currently, insulation is only identified to be removed to examine the MNSA clamps on the two pressurizer heater sleeves. In all other locations the insulation has remained in place consistent with the ASME Code. However, as noted in the response to Question 1, this is being changed with the revision of the inspection procedure. Either the insulation will be removed or a method of getting under the insulation for an examination will be devised. Millstone Unit No. 2 has a mix of mirror and fiber insulation on components and piping that carry borated water. Combustion Engineering (CE) Reactor Coolant Systems (RCS) utilize Alloy 600 for a number of small bore nozzles for pressure and temperature measurement along with the pressurizer heater sleeves. The leakage found on the pressurizer heater sleeves demonstrates when a visual examination is performed, it must be a bare metal examination to find very small leaks. The exception to the above is the reactor vessel head penetrations, where Millstone Unit No. 2 has employed a UT technique demonstrated to be able to detect erosion as a result of leakage. This volumetric examination was selected in lieu of a bare metal visual of the reactor vessel head because of the close fitting insulation package, as described in a DNC letter dated September 4, 2001.[6]

Should indications be found that are indicative of coolant leakage, a condition report (CR) will be written. The investigation of such a CR will identify the reason for the leakage. This will include the removal of insulation, should the situation warrant, to determine the cause for the staining or boric acid deposits.

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[6] J. A. Price letter to U.S. Nuclear Regulatory Commission, "Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated September 4, 2001.

NRC Question 3

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.*

DNC Response:

The current procedure for boric acid corrosion inspection requires that the reactor coolant system and supporting systems that convey borated water be inspected at every refueling and cold shutdown. The locations and frequency of inspections are based upon the guidance from Generic Letter 88-05. Equipment located inside of containment is inspected as described above. Parts of these systems that are not in containment are also inspected during a refueling outage or cold shutdown, and when the plant is on line by operations personnel during their normal rounds.

The components inside containment that become inaccessible during power operation are monitored during normal operation by the RCS leak detection system. The Millstone Unit No. 2 leak detection system consists of containment sump level monitoring (one channel), containment atmosphere particulate radioactivity monitoring (two channels), and containment atmosphere gaseous radioactivity monitoring (two channels). The requirements for these systems are found in the Millstone Unit No. 2 Technical Specification section 3.4.6.1.

The containment sump level monitoring systems are capable of detecting a leak of one gallon per minute (gpm) in a period of one hour or less in accordance with the recommendations of Reg. Guide 1.45. The sensitivity and response time of the containment atmosphere radiation monitors are dependent upon a number of factors. Experience has shown that the monitors are sensitive to small leak rates. As an example on Millstone Unit No. 2, from July 12th through July 14th, 1999, a small leak of .05 - .15 gpm (based on plant process computer (PPC) calculations from containment sump levels) was identified on a tubing connection on a reactor vessel differential pressure transmitter. Operators were first alerted to the situation by an increase in containment atmosphere particulate and gaseous activity. This upward trend correlated with a slight increase in containment sump level. A containment entry verified the source of the leakage as a tube fitting on the pressure transmitter. Isolating the transmitter stopped the leak.

The containment atmosphere radiation monitors are very useful as an early warning of increased RCS leakage. On Millstone Unit No. 2, PPC alarms are provided based on the rate of change of the containment particulate and gaseous radiation monitor channels ("RCS leak rate rising" alarms) which prompts operators to check other indications for the possibility of an RCS leak. Abnormal operating procedure 2568, "Reactor Coolant Leak," provides a structured method for diagnosing and responding to RCS leaks.

Operations personnel are assisted by the PPC in tracking RCS leakage. The PPC performs a reactor coolant system water inventory balance on water added and lost from the reactor coolant system to determine identified and unidentified leakage rates. The reactor coolant system water inventory balance is performed at least every 72 hours during steady state operation as required by Technical Specification Surveillance Requirement 4.4.6.2.1. In addition, the PPC can calculate a leak rate based upon the rate of change of the containment sump level. The RCS identified and unidentified leakage rates are recorded daily on the shift turnover report. Operations personnel are very sensitive to changes in RCS leakage rate.

NRC Question 4

4) *Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g. bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,*

   a) *If observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or*

   b) *If observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.*

DNC Response:

Should leakage of a mechanical joint occur, a CR will be written to document the situation. The corrective action program will then be used to evaluate the situation against the existing design and technical specification requirements for the components in question. A number of factors are important in determining if leakage from a bolted connection is acceptable. These factors will be used along with the guidance from Code Case N-566-1 as part of a decision tree to determine acceptability of the leakage. The major factors and examples of actions that might be taken are:

- The age of the leakage is considered. If the leakage is active, then further investigation is warranted. If the boric acid crystals are dry, i.e., an old inactive leak, and there is no damage to any of the components, then the evidence of leakage will be cleaned up.

- The rate of leakage will be a criteria. Leakage greater than a number of drops per minute will warrant further evaluation. A steady stream of either water or gas will need immediate remedial action to stop the leakage.

- The location of the leakage is also important. Leakage found inside containment during a refueling outage will be evaluated and may not be repaired provided the requirements of Code Case N-566-1 are met and other compensatory measures are taken. Leakage outside of containment found while on line may also be acceptable provided the requirements of Code Case N-566-1 are met.

- The materials of construction are important in determining whether leakage from a bolted connection is acceptable for continued operation as stipulated in Code Case N-566-1.

- If the leakage can easily be isolated and the rate is a steady stream, then the action will be to isolate the leakage and correct the problem. If the leakage is unisolable, and the requirements of Code Case N-566-1 are met, the leakage will be monitored. If the leakage is found inside containment during a refueling outage, then action will be taken to stop the leakage prior to restart of the unit.

- The impact of the leak on other components is also important as stipulated in Code Case N-566-1. Leaks that don't impact other components may be acceptable provided the requirements of Code Case N-566-1 can be met.

If leakage from a bolted connection is determined to be acceptable, the leakage will be contained in some manner. The rate of leakage will be monitored both by visual observation on a periodic basis, i.e. operator rounds, and if possible with indirect measurements such as area radiation monitors. An increase in leakage rate will be cause to re-evaluate the acceptability decision against the criteria discussed above. This review will be documented in an engineering evaluation.

Should the evaluation determine that the leakage is unacceptable then actions will be taken to stop the leakage. This would usually mean isolating the leak and tightening fasteners, or replacing gaskets, to make the joint leak tight again.

NRC Question 5

5) *Explain the capabilities of your program to detect low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore 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 incore 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.*

DNC Response:

This question is not applicable to Millstone Unit No. 2 since the reactor design does not have bottom mounted incore instrumentation nozzles.

NRC Question 6

6) *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.*

DNC Response:

As noted in the answer to Question 3, the data from sump measurements and radiation monitors has found leaks of approximately 0.1gpm. Leak rates lower than that would be difficult to detect or quantify. The leaks on the pressurizer heater sleeves were less than this threshold and were only found with a visual examination with the insulation removed. As noted in answer to Question 2, the inspection procedure is being revised so insulation is either removed or looked under to perform bare metal examinations of Alloy 600 components and welds that are susceptible to cracking.

Once a leak has been identified with the initiation of a CR, the investigation of that CR will look at the impact of the leak on surrounding components. The methodology for this investigation will follow the description in the answer to Question 4.

NRC Question 7

7) *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 or consequence models.*

DNC Response:

Millstone Unit No. 2 does not use any susceptibility models to help determine either location or frequency of inspection. The use of a consequence model (PRA) is one tool that may be used in the analysis of leakage investigations to provide further justification of a particular situation.

NRC Question 8

8) *Provide a summary of recommendations made by your reactor vendor on visual inspection 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.*

DNC Response:

The cracking of Alloy 600 pressurizer heater sleeves has been an issue at CE designed nuclear steam supply systems since the late 1980's. The CE Owner's Group (CEOG) sponsored an evaluation of this problem in 1989. The report from this work[7] recommended an initial inspection of the heater sleeves for all plants and further follow up examinations based upon the results of the initial examination and industry experience. Millstone Unit No. 2 performed the initial inspection in late 1989 and again in late 1990, with no indications of leakage detected. In 1991, following a destructive examination of a cracked heater sleeve from Calvert Cliffs Unit No. 3, CE changed this recommendation slightly to, "CE recommends visual inspection of pressurizer nozzle locations for evidence of boric acid deposits or iron oxide (rust) corrosion product at each refueling outage." In the absence of observable leakage and with the experience from other CE plants indicating that most leaks, although small, were large enough to create a drip or puddle of water that would be easily detectable, no further bare metal inspections with the insulation removed were performed at Millstone Unit No. 2 until the February, 2002, refueling outage. Inspections with the insulation in place were performed every outage. By 2002, cracking at other CE designed plants as well as other industry events emphasized the need to do a bare metal inspection in the absence of leakage. These recommendations were recently confirmed by the CEOG, which is now part of the Westinghouse Owner's Group (WOG).[8]

NRC Question 9

9) *Provide the basis for concluding that the inspections and evaluations described in your response to the above questions comply with your Plant Technical Specifications and Title 10 of the Code of Federal Regulation (10CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically address how your boric acid corrosion control program complies with Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.*

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(7) Report CEN-393-NP, "Evaluation of Pressurizer Heater Sleeve Susceptibility to Primary Water Stress Corrosion Cracking," dated November, 1989.

(8) Westinghouse Owner's Group letter to U.S. Nuclear Regulatory Commission, WOG-02-223, "Transmittal of Response to NRC Request for Information, Bulletin 2002-01: Vendor Recommendations for Visual Inspection of Alloy 600/82/182 Component Locations," dated December 13, 2002.

DNC Response:

Both procedures that cover inspections for leakage, SP 21238 and C EN 109, meet the requirements of ASME XI Paragraph 5250. Specifically, SP 21238 requires that an RCS leak inspection be considered UNSATISFACTORY and a CR be written if any leakage is found. The investigation of this CR would then include an engineering evaluation of the affected system, structure or component to determine acceptability for continued use or if a repair/replacement was needed.

Also both procedures, SP-21238 and C EN 109, require that leakage or boric acid residue found be evaluated in accordance with Code Case N-566-1 to determine if any bolting is damaged and requires replacement. In addition, other components that may have boric acid residue as a result of the leakage are examined per IWA 5250(b). Bolting that could have been damaged is removed and a VT-3 inspection by a qualified inspector is performed.

Millstone Procedure WC-3, "ASME Section XI Repair and Replacement Program," controls the repair/replacement activities.

The basis for compliance to TS and Code requirements is in the use of the procedures described above, the NRC approved leakage detection systems described in response to Question 3, and the results of inspections that are conducted post shutdown, prior to reactor startup at normal operating pressure and temperature, and during operation of accessible systems.

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Attachment 2

Millstone Power Station, Unit Nos. 2 and 3

Reply to Request for Additional Information
Related to NRC Bulletin 2002-01
Millstone Unit No. 3

Reply to Request for Additional Information
Related to NRC Bulletin 2002-01
Millstone Unit No. 3

NRC Question 1

1) *Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, frequency of inspections, personnel qualifications, and degree of insulation removal for the examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel (RPV) bottom head).*

DNC Response:

Currently the site procedure for the inspection of components exposed to boric acid, C EN 109 "Inspection of Components Exposed to Boric Acid," has a table that lists the places to visually inspect during every refueling outage. This list, developed as a result of Generic Letter 88-05, primarily contains bolted connections. The only Alloy 600 components currently included in this list are the reactor vessel head penetrations. The insulation has been left in place for these examinations in accordance with the ASME Code. No Alloy 82/182 welds are currently included in procedure C EN 109 to be specifically examined. There are currently no personnel qualification requirements in C EN 109 other than the examiner must be an Inservice Inspection Technician, which implies that they possess non-destructive examination (NDE) certifications. The boric acid inspection program at Millstone is currently being revised. Specific training and qualifications will become mandatory for the personnel performing the inspections as part of the revision.

Based upon industry experience, Millstone Unit No. 3 performed a bare metal visual examination of the reactor vessel head penetrations under the insulation during the September 2002 refueling outage.[9] In light of recent industry events, the Millstone Station procedures for boric acid control are being revised. A preliminary list of Alloy 600 penetrations and Alloy 82/182 welds has been developed by the Westinghouse Owners Group (WOG). Those locations where a leak could possibly cause degradation of carbon steel will be added to the list of components to be examined. The frequency of inspection of the Alloy 600 components and welds will be determined based upon a number of factors including an evaluation of the temperature that the component or weld is exposed.

[9] J. A. Price letter to U.S. Nuclear Regulatory Commission, "Response to NRC Bulletins Regarding Reactor Pressure Vessel Head Inspections," dated November 1, 2002.

NRC Question 2

2) *Provide the technical basis for determining whether or not insulation is removed to examine all 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 the limitations to removal of insulation. Also include in your response action involving removal of insulation required in your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.*

DNC Response:

Previously, insulation was not removed to examine any components at Millstone Unit No. 3 as allowed by the ASME Code. However, as noted in the response to Question 1, this is being changed with the revision of the inspection procedure. Either the insulation will be removed or a method of getting under the insulation for an examination will be devised. Millstone Unit No. 3 has a mix of mirror and fiber insulation on piping and components that carry borated water. As noted in Question 1, a preliminary list of Alloy 600 components and Alloy 82/182 weldments has been developed by the WOG. In addition to the reactor vessel head penetrations, the preliminary list includes the instrument penetrations on the bottom reactor vessel head along with a number of pipe welds. This list will be incorporated into the Millstone inspection procedure, C EN 109.

Should indications be found that are indicative of coolant leakage, a condition report (CR) will be written. The investigation of such a CR will identify the reason for the leakage. This will include the removal of insulation, should the situation warrant, to determine the cause for the staining or boric acid deposits.

NRC Question 3

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.*

DNC Response:

The current procedure for boric acid corrosion inspection requires that the reactor coolant system and supporting systems that convey borated water be inspected at every refueling and cold shutdown. The locations and frequency of inspections are based upon the guidance from Generic Letter 88-05. Equipment located inside of containment is inspected as described above. Parts of these systems that are not in containment are also inspected during a refueling outage or cold shutdown, and when the plant is on line by operations personnel during their normal rounds.

The components inside containment that become inaccessible during power operation are monitored during normal operation by the RCS leak detection system. The Millstone Unit No. 3 leak detection system consists of containment sump level monitoring (one channel), containment atmosphere particulate radioactivity monitoring (one channel), and containment atmosphere gaseous radioactivity monitoring (one channel). The requirements for these systems are found in the Millstone Unit No. 3 Technical Specification section 3.4.6.1.

The containment sump level monitoring systems are capable of detecting a leak of one gallon per minute (gpm) in a period of one hour or less in accordance with the recommendations of Reg. Guide 1.45. The sensitivity and response time of the containment atmosphere radiation monitors are dependent upon a number of factors. Experience has shown that the monitors are sensitive to small leak rates, as was described in the Unit 2 response to Question 3 (Attachment 1 of this letter).

The containment atmosphere radiation monitors are very useful as an early warning of increased RCS leakage. On Millstone Unit No. 3, containment particulate and gaseous radiation monitor alarms are based on activity level and provide an input to a Main Board radiation monitoring system (RMS) Annunciator which prompts operators to check other indications for the possibility of an RCS leak. Abnormal operating procedure 3555, "Reactor Coolant System Leak," provides a structured method for diagnosing and responding to RCS leaks.

Operations personnel are assisted by the plant process computer (PPC) in tracking RCS leakage. The PPC performs a reactor coolant system water inventory balance on water added and lost from the reactor coolant system to determine identified and unidentified leakage rates. The reactor coolant system water inventory balance is performed at least every 72 hours during steady state operation as required by Technical Specification Surveillance Requirement 4.4.6.2.1. In addition, the PPC can calculate a leak rate based upon the rate of change of the containment sump level. The RCS identified and unidentified leakage rates are recorded daily on the shift turnover report. Operations personnel are very sensitive to changes in RCS leakage rate.

NRC Question 4

4) *Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g. bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,*

    a) *If observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or*

    b) *If observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.*

DNC Response:

Should leakage from a mechanical joint occur, a CR will be written to document the situation. The corrective action program will then be used to evaluate the situation against existing design and technical specification requirements for the components in question. A number of factors are important in determining if leakage from a bolted connection is acceptable. These factors will be used along with the guidance from Code Case N-566-1 as part of a decision tree to determine acceptability of the leakage. The major factors and examples of actions that might be taken are:

- The age of the leakage is considered. If the leakage is active, then further investigation is warranted. If the boric acid crystals are dry, i.e., an old inactive leak, and there is no damage to any of the components, then the evidence of leakage will be cleaned up.

- The rate of leakage will be a criteria. Leakage greater than a number of drops per minute will warrant further evaluation. A steady stream of either water or gas will need immediate remedial action to stop the leakage.

- The location of the leakage is also important. Leakage found inside containment during a refueling outage will be evaluated and may not be repaired provided the requirements of Code Case N-566-1 are met and other compensatory measures are taken. Leakage outside of containment found while on line may also be acceptable provided the requirements of Code Case N-566-1 are met.

- The materials of construction are important in determining whether leakage from a bolted connection is acceptable for continued operation as stipulated in Code Case N-566-1.

- If the leakage can easily be isolated and the rate is a steady stream, then the action will be to isolate the leakage and correct the problem. If the leakage is unisolable, and the requirements of Code Case N-566-1 are met, the leakage will be monitored. If the leakage is found inside containment during a refueling outage, then action will be taken to stop the leakage prior to restart of the unit.

- The impact of the leak on other components is also important as stipulated in Code Case N-566-1. Leaks that don't impact other components may be acceptable provided the requirements of Code Case N-566-1 can be met.

If leakage from a bolted connection is determined to be acceptable, the leakage will be contained in some manner. The rate of leakage will be monitored both by visual observation on a periodic basis, i.e. operator rounds, and if possible with indirect measurements such as area radiation monitors. An increase in leakage rate will be cause to re-evaluate the acceptability decision against the criteria discussed above. This review will be documented in an engineering evaluation.

Should the evaluation determine that the leakage is unacceptable then actions will be taken to stop the leakage. This would usually mean isolating the leak and tightening fasteners, or replacing gaskets, to make the joint leak tight again.

NRC Question 5

5) *Explain the capabilities of your program to detect low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore 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 incore 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.*

DNC Response:

Millstone Unit No. 3 has incore instrumentation nozzles that are in the bottom of the reactor vessel head. These nozzles are made from Alloy 600. As noted in Question 1, these locations are not visually inspected as part of the current inspection procedure but will be added in the next revision. These nozzles are not insulated.

Leakage from these nozzles, or the instrumentation tubes themselves, would be noted as an increase in sump level.

NRC Question 6

6) *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 addresses leakage that may impact components that are in the leak path.*

DNC Response:

As noted in the answer to Question 3, the data from sump measurements, radiation monitors, walkdowns, daily measurement of identified and unidentified leak rates and trending of containment temperatures and reactor coolant pump motor stator temperatures are used to find leakage. As noted in answer to Question 2, the inspection procedure is being revised so insulation is either removed or looked under to perform bare metal examinations of Alloy 600 components and welds that are susceptible to cracking.

Once a leak has been identified with the initiation of a CR, the investigation of that CR would look at the impact of the leak on surrounding components. The methodology for this investigation would follow the description in the answer to Question 4.

NRC Question 7

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

DNC Response:

Millstone Unit No. 3 does not use any susceptibility models to help determine either location or frequency of inspection. The use of a consequence model (PRA) is one tool that may be used in the analysis of leakage investigations to provide further justification of a particular situation.

NRC Question 8

8) *Provide a summary of recommendations made by your reactor vendor on visual inspection 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.*

DNC Response:

Westinghouse has not made any recommendations on visual inspections of Alloy 600 locations in Westinghouse nuclear steam supply systems. A review of communications by Westinghouse has recently been made to confirm this conclusion.[10] WCAP 13525 and WCAP 13525 Revision 1 submitted safety assessments of control rod drive mechanism (CRDM) nozzle cracking to the NRC and discussed possible inspection strategies, but did not include recommendations or suggestions for visual inspections.

NRC Question 9

9) *Provide the basis for concluding that the inspections and evaluations described in your response to the above questions comply with your Plant Technical Specifications and Title 10 of the Code of Federal Regulation (10CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically address how your boric acid corrosion control program complies with Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.*

DNC Response:

Both procedures that cover inspections for leakage, SP 3601F.7 and C EN 109, meet the requirements of ASME XI Paragraph 5250. Specifically, SP 3601F.7 requires that an RCS leak inspection be considered UNSATISFACTORY and a CR be written if any leakage is found. The investigation of this CR would then include an engineering evaluation of the affected system, structure or component to determine acceptability for continued use or if a repair/replacement was needed.

Also both procedures, SP 3601F.7 and C EN 109, require that leakage or boric acid residue found be evaluated in accordance with Code Case N-566-1 to determine if any bolting is damaged and requires replacement. In addition, other components that may have boric acid residue as a result of the leakage are examined per IWA 5250(b). Bolting that could have been damaged is removed and a VT-3 inspection by a qualified inspector is performed.

Millstone Procedure WC-3, "ASME Section XI Repair and Replacement Program," controls the repair/replacement activities.

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[10] WOG-02-223, "Transmittal of Response to NRC Request for Information, Bulletin 2002-01: Vendor Recommendations for Visual Inspection of Alloy 600/82/182 Component Locations," dated December 13, 2002.

The basis for compliance to TS and Code requirements is in the use of the procedures described above, the NRC approved leakage detection systems described in response to Question 3, and the results of inspections that are conducted post shutdown, prior to reactor startup at normal operating pressure and temperature, and during operation of accessible systems.