Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

APR 2 2002

Docket Nos. 50-336
50-423
818621

RE: 10 CFR 50.54(f)

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

Millstone Nuclear Power Station, Unit Nos. 2 and 3
Response to NRC Bulletin 2002-01
Reactor Pressure Vessel Head Degradation and
Reactor Coolant Pressure Boundary Integrity

This submittal is the Dominion Nuclear Connecticut, Inc. (DNC) response to the Nuclear Regulatory Commission (NRC) Bulletin 2002-01, dated March 18, 2002.[1] Attachment 2 provides the information for both the 15 day required response and the 30 day required response for Millstone Unit No. 2. The information for the Unit No. 3 15 day required response is included as Attachment 3.

The regulatory commitments contained in this letter are located in Attachment 1.

Should there be any questions regarding this submittal, please contact Mr. Ravi G. Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

Attachments (3)

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

---

(1) Nuclear Regulatory Commission Bulletin from D. B. Matthews to the industry, "NRC Bulletin 2002-01: Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002.

Affirmation

I, J. Alan Price, being duly sworn, state that I am Site Vice President of Dominion Nuclear Connecticut, Inc., that I am authorized to sign and file this information with the Nuclear Regulatory Commission on behalf of Dominion Nuclear Connecticut, Inc., and that the statements made and the matters set forth herein pertaining to Dominion Nuclear Connecticut, Inc. are true and correct to the best of my knowledge, information and belief.

Dominion Nuclear Connecticut, Inc.

J. Alan Price
Site Vice President - Millstone

STATE OF Connecticut

COUNTY OF New London

Subscribed and sworn to before me, a Notary Public, in and for the County and State above named, this 2ND day of April, 2002.

My Commission Expires: __________

SANDRA J. ANTON
NOTARY PUBLIC
COMMISSION EXPIRES
MAY 31, 2005

Attachment 1

Millstone Nuclear Power Station, Unit Nos. 2 and 3

Regulatory Commitments

List of Regulatory Commitments

The following table identifies action committed to by DNC in this document.

Number	Commitment	Due
B18621-0I	Millstone Unit No. 3 will perform a bare metal visual inspection under the insulation.	During the next refueling outage.
B18621-02	in support of the bare head visual examination, Millstone Unit No. 3 will do an analysis that demonstrates that a through-wall leak in a penetration nozzle would **be** detected by this visual inspection.	Prior to the next refueling outage.

Docket Nos. 50-336
50-423
818621

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

Required 15 and 30 Day Response to
NRC Bulletin 2002-01

Millstone Nuclear Power Station, Unit No. 2
NRC Bulletin 2002-01 Response

On March 18, 2002, the Nuclear Regulatory Commission (NRC) issued Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity" to all holders of operating licenses for pressurized-water nuclear power reactors. A 15 day response was required per question 1 of the Required Information section. Additionally, a 30 day response following restart once a reactor pressure vessel head inspection has been performed is required. Below is the Millstone Unit No. 2 response, including inspection scope and findings to cover the 30 day response.

Question 1A

A summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant,

DNC Response

Millstone Unit No. 2 conducts inspections to comply with Generic Letter 88-05. These inspections cover two groups of components: the reactor coolant pressure boundary and the supporting systems/components that could be exposed to boric acid. The inspection of the reactor coolant pressure boundary is performed shortly after the plant reaches Mode 5 each refueling outage. The inspection of the supporting systems and components is performed any time leakage is detected. This inspection is proceduralized with increased attention drawn to the joints and areas of particular importance. The reactor vessel penetrations are listed in the procedure as an area of interest. These inspections are performed by VT-2 qualified personnel.

Millstone Unit No. 2 has recently completed ultrasonic (UT) inspections of all 78 penetrations in the reactor vessel head in accordance with NRC Bulletin 2001-01. This inspection was conducted from underneath the reactor vessel head and was capable of finding both axial and circumferential cracks in the penetrations as well as finding evidence of a leak path between a nozzle and the reactor vessel head. No through-wall cracking or evidence of leakage was found. Additional examinations and investigations were completed as a result of the Davis-Besse findings, which occurred while Millstone Unit No. 2 was shutdown for refueling.

First, a thorough visual examination of all exposed external surfaces above the top of the reactor vessel head was performed and documented on video tape. The insulation on Millstone Unit No. 2 is fitted in such close proximity to the vessel head that this visual examination focused primarily on the top of the insulation. No evidence of a build-up of boric acid was found. Trace amounts of boric acid were detected, but nothing more than a light dusting. Evidence of past leakage from control element drive mechanism (CEDM) vent valves was also noted on the sides of two CEDM housings. In addition, access was obtained under the insulation to facilitate a visual inspection around several incore instrumentation (ICI) penetrations and CEDM penetration No. 21. These inspections did not reveal any evidence of boric acid damage on top of the head.

Following the video inspection, the configuration of the Millstone Unit No. 2 insulation was verified by boroscope. This inspection showed that the insulation is completely encapsulated in stainless steel.

Second, selected areas of the reactor vessel head were ultrasonically scanned to verify the condition of the low alloy steel material. The UT scans were performed from underneath the reactor vessel head on the clad surface. These areas include an area which displayed the most significant surface staining, between CEDM nozzle No. 19 and CEDM nozzle No. 34, and an area which presents the smallest incline (the area around CEDM nozzle No. 1). All thickness measurements were greater than eight inches. The minimum design thickness of the head plus the cladding is seven and 11/16 inches.

Third, the UT data that was taken on the reactor vessel head penetration nozzles at Millstone Unit No. 2 was reviewed a second time to verify that solid material existed around all penetrations and to determine if the “ghost pattern” observed at Davis-Besse could be seen in any of the Unit No. 2 data. The reviews found no evidence of the Davis-Besse pattern.

Fourth, a review was conducted of the continuous air monitors in containment to determine if any unusual particulates had been found during the last cycle. This review showed no pick-up of iron on the filters which could be indicative of large scale damage to a carbon steel component.

Question 1B

An evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse.

DNC Response

The Generic Letter 88-05 walkdowns have proven a very effective technique for identifying leakage especially from the reactor vessel head area. If leakage is detected, the boric acid crystals are removed and the parts examined for damage. The procedure for these inspections includes criteria for various courses of action depending upon the severity of the damage. This approach has proven effective in maintaining the leak tightness of the reactor coolant system.

Question IC

A description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions,

DNC Response

Historically, Millstone Unit No. 2 has had three conditions where leakage has lead to boric acid crystals on the head or nearby equipment. In 1988, both reactor vessel head O-rings failed creating a leak path that existed for several months. In the refueling outage following this leakage, an extensive effort was made to clean up the boric acid crystals from all around the reactor vessel. Nine reactor vessel studs were replaced as a result of this leakage. The cold leg nozzles were cleaned and no measurable damage found.

The second condition was a chronic issue that resulted in a design change to eliminate its occurrence. The ICI nozzles all have flanges where the instruments are inserted. These flanges had been sources of minor leakage. Damage to the bolting on these flanges was a recurring problem. No damage to the reactor vessel head was found and the boric acid residue was removed. The design of these flanges was changed in the early 1980's to one that sealed better and the incidence of leakage has been significantly reduced. If any leakage is noted, it is cleaned up and the parts examined for corrosion damage.

A third source of leakage has been the vent valves on the top of the CEDM housings. These valves are used during start-up to vent the CEDM housings. Occasionally, when the valves were closed, some minor leakage persisted. Because of this, the vent valves were modified in 1992 with an improved design. These valves are reconditioned during each refuel in order to ensure leak tightness.

Question 1D

Your schedule, plans, and basis for future inspections of the reactor pressure **vessel** head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria,

DNC Response

As previously noted, Millstone Unit No. 2 has just completed the inspection of all the penetration nozzles in the reactor vessel head required per bulletin 2001-01. These inspections included both a demonstrated volumetric exam utilizing ten transducers at varying angles and an examination of the low alloy steel directly adjacent to the nozzles above the weld. The latter exam has been demonstrated to be effective in detecting leak paths. The technique was presented to the NRC Staff prior to the Millstone Unit

No. 2 outage and received concurrence from the NRC per letter dated February 21, 2002.[1]

Millstone Unit No. 2 will continue to perform boric acid walkdowns and inspections each refueling outage. The procedure for compliance with Generic Letter 88-05 was revised and made common to both Millstone Unit Nos. 2 and 3 in November 2000. This procedure will be evaluated in light of the recent experience at Davis-Besse to determine if any changes are needed. Our 60 day response to this bulletin will provide the results of our evaluation.

Question 1E

Your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in response to Item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. Include the following specific information in the discussion:

(1) If your evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met, discuss your plans for plant shutdown and inspection.

(2) If your evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

DNC Response

Part (1) - Not Applicable

Part (2) - There has not been a 100% bare head visual inspection conducted at Millstone Unit No. 2 due to its current insulation configuration. However, there is reasonable assurance that regulatory requirements are currently being met. The following discussion provides the basis that all regulatory requirements discussed in the applicable Regulatory Requirements section will continue to be met.

---

(1) J. T. Harrison, Nuclear Regulatory Commission, letter to J. A. Price, "Millstone Nuclear Power Station, Unit No. 2 (Millstone 2) - Response to DNC's February 7, 2002, Supplemental Response to Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated February 21, 2002.

Design Requirements: 10 CFR 50, Appendix A – General Design Criteria

The Bulletin states:

> "The applicable GDC include GDC 14 (Reactor Coolant Pressure Boundary), GDC 31 (Fracture Prevention of Reactor Coolant Pressure Boundary), and GDC 32 (Inspection of Reactor Coolant Pressure Boundary). GDC 14 specifies that the reactor coolant pressure boundary (RCPB) has an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized. GDC 32 specifies that components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leak tight integrity; inspection practices that do not permit reliable detection of degradation are not consistent with this GDC."

DNC Response

The three referenced General Design Criteria (GDC) state the following:

- Criterion 14 – Reactor Coolant Pressure Boundary

  "The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

- Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

  "The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration *of* service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient thermal stresses, and **(4)** size of flaws."

- Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

  "Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel."

During the initial plant licensing of Millstone Unit No. 2, it was demonstrated that the design of the reactor coolant pressure boundary met the regulatory requirements in place at that time. The GDC included in Appendix A to 10 CFR Part 50 did not become effective until May 21, 1971. The Construction Permit for Millstone Unit No. 2 was issued prior to May 21, 1971; consequently, this unit was not subject to GDC requirements. (Reference SECY-92-223 dated September 18, 1992.) However, the following information demonstrates compliance with the design criteria relative to the cracking of reactor pressure vessel (RPV) top head nozzles and the potential for subsequent wastage of the vessel head:

- Pressurized water reactors licensed both before and after issuance of Appendix A to 10 CFR Part 50 (1971) complied with these criteria in part by: 1) selecting Alloy 600 or other austenitic materials with excellent corrosion resistance and extremely high fracture toughness, for reactor coolant pressure boundary materials, and 2) following ASME Codes and Standards and other applicable requirements for fabrication, erection, and testing of the pressure boundary parts. NRC reviews of operating license submittals subsequent to issuance of Appendix A included evaluating designs for compliance with the General Design Criteria. The standard review plans (SRPs) in effect at the time of licensing did not address the selection of Alloy 600. They required that ASME Code requirements be satisfied.

- Although stress corrosion cracking of primary coolant system penetrations was not originally anticipated during plant design, it has occurred in the RPV top head nozzles at some plants. The robustness of the design has been demonstrated by the small amounts of the leakage that has occurred and by the fact that none of the cracks in Alloy 600 reactor coolant pressure boundary materials has rapidly propagated or resulted in catastrophic failure or gross rupture. The suitability of the originally selected materials has been confirmed. Given the inherently high fracture toughness and flaw tolerance of the Alloy 600 material, there is in fact an extremely low probability of a rapidly propagating failure and gross rupture. It should be noted that earlier versions of the GDCs are in terms of extremely low probability of gross rupture or significant leakage throughout design life.

- Recent events at the Davis-Besse plant have demonstrated that the design of the reactor vessel head is very robust and that it can tolerate significant degradation without rapidly propagating failure or gross rupture. The inspection program at Millstone Unit No. 2 is capable of discovering leakage that could lead to wastage or degradation of the reactor vessel head and ensures that continued structural and leak tight integrity is maintained.

As described above, the requirements established for design, fracture toughness, and inspectability in GDC 14, 31, and 32 respectively were satisfied during Millstone Unit No. 2's initial licensing review, and continue to be satisfied during operation, even in the presence of a potential for stress corrosion cracking of the reactor vessel head penetrations and/or subsequent wastage of the reactor vessel head.

Inspection Requirements: 10 CFR 50.55a and ASME Section XI

The Bulletin states:

"NRC regulations contained in 10 CFR 50.55a state that American Society of Mechanical Engineers (ASME) Class 1 components (which includes the reactor coolant pressure boundary) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Table IWA-2500-1 [IWB-2500-1[(2)]] of Section XI of the ASME Code provides examination requirements for VHP nozzles and references IWB-3522 for acceptance standards. IWB-3522.1(c) and (d) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as 'the through-wall leakage that penetrates the pressure retaining membrane.' Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall degradation of the reactor coolant pressure vessel head penetration nozzles.

For through-wall leakage identified by visual examinations in accordance with the ASME Code, acceptance standards for the identified degradation are provided in IWB-3142. Specifically, supplemental examination (by surface or volumetric examination), corrective measures or repairs, analytical evaluation, and replacement provide methods for determining the acceptability of degraded components."

DNC Response

Title 10 of the Code of Federal Regulations, Part 50.55a requires that inservice inspection and testing be performed per the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, 'Inservice Inspection of Nuclear Plant Components." Section XI contains applicable rules for examination, evaluation and repair of code class components, including the reactor coolant pressure boundary.

Requirements for partial penetration welds attaching control rod drive (CRD) housings to the reactor vessel head are contained in Table IWB-2500-1, Examination Category B-E, "Pressure Retaining Partial Penetration Welds in Vessels," Item Numbers: B4.10, "Partial Penetration Welds;" B4.11, "Vessel Nozzles;" B4.12, "CRD Nozzles;" and B4.13, "Instrumentation Nozzles.'' The Code requires a VT-2 visual examination of 25% of the CRD nozzles from the external surface. Since the head is insulated, and the nozzles do not represent a bolted flange, paragraph IWA-5242(b) permits these inspections to be performed with the insulation left in place.

Millstone Unit No. 2 will continue to perform visual inspections for evidence of leakage on top of the insulation and above the insulation pursuant to our commitments to NRC

(2) An erratum appears to exist in the Bulletin. Table IWA-2500-1 is cited, but does not exist. It appears that the citation should have been IWB-2500-1.

Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." Millstone Unit No. 2 will follow industry experience to develop inspections of the reactor vessel head penetration nozzles in accordance with any re-inspection criteria developed from the Bulletin 2001-01 inspections. Future inspections planned for the reactor vessel heads meet the requirements of ASME Section XI both in scope and frequency.

The acceptance standard for the visual examination is found in paragraphs IWA-5250, "Corrective Measures'' and IWB-3522, "Standards for Examination Category B-E, Pressure Retaining Partial Penetration Welds in Vessels, and Examination Category B-P, All Pressure Retaining Components." Paragraph IWA-5250 requires repair or replacement of the affected part if a through-wall leak is found and requires an assessment of damage, if any, associated with corrosion of steel components by boric acid. Millstone Unit No. 2 will not return to service after finding a leak from a RPV top head nozzle without first having repaired the nozzle and having assessed any wastage of the head the leakage may have caused.

In addition, flaws identified by Non-Destructive Examination (NDE) methods, which are not addressed by specific ASME Section XI acceptance criteria are evaluated in accordance with the flaw evaluation rules for piping contained in Section XI of the ASME Code. Any flaw not meeting requirements for the intended service period would be repaired before returning it to service.

Repairs to RPV top head nozzles would be performed in accordance with Section XI requirements, NRC approved ASME Code Case requirements, or an alternative repair or replacement method approved by the NRC.

Millstone Unit No. 2 complies with these ASME Code requirements through implementation of its inservice inspection program. If a VT-2 examination detects the conditions described by IWB-3522.1(c) and (d), then corrective actions per IWB-3142 will be performed in accordance with the plant's corrective action program. No new plant actions are necessary to satisfy the cited regulatory criteria.

<u>Quality Assurance Requirements: 10 CFR 50, Appendix B</u>

<u>The Bulletin states:</u>

> "Criterion V (Instructions, Procedures, and Drawings) of Appendix B to 10 CFR Part 50 states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Criterion V further states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual and volumetric examinations of the reactor coolant pressure boundary are activities that should be documented in accordance with these requirements."

DNC Response

Any of the work undertaken to inspect, evaluate, and/or repair the Millstone Unit No. 2 reactor vessel head penetrations has been and will be conducted and documented in accordance with existing or new procedures which comply with the Company's Quality Assurance (QA) Topical Report, the QA program, and Criterion V of Appendix B to 10 CFR Part 50.

The Bulletin states:

> "Criterion IX (Control of Special Processes) of Appendix B to 10 CFR Part 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of the reactor coolant pressure boundary for the degradation observed at Davis-Besse, special requirements for visual examination and/or ultrasonic testing would generally require the use of a qualified visual and ultrasonic testing methods. Such methods are ones that a plant-specific analysis has demonstrated would result in the reliable detection of degradation prior to a loss of specified reactor coolant pressure boundary margins of safety. The analysis would have to consider, for example, the as-built configuration of the system and the capability to reliably detect and accurately characterize flaws or degradation, and contributing factors such as the presence of insulation, preexisting deposits on the RPV head, and other factors that could interfere with the detection of degradation."

DNC Response

The designed range of interference fit of the vessel head penetration (VHP) nozzles in the Millstone Unit No. 2 vessel head has been shown to result in gaps between the penetration tube and hole in the vessel head at operating pressure and temperature. Consequently, flaws breaching the reactor head penetration will result in discernable leakage. The UT inspection technology that was just applied to Millstone Unit No. 2 reactor head penetrations has been demonstrated to be effective in identifying a leakage path.

Additionally, qualification of any other NDE techniques we would use for the inspections have been or will be demonstrated prior to use.

The last Appendix B criterion cited in the bulletin is:

> "Criterion XVI (Corrective Action) of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For degradation of the reactor coolant

> pressure boundary, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future degradation. These actions could include proactive inspections and repair of degraded portions of the reactor coolant pressure boundary."

DNC Response

Criterion XVI contains two important attributes pertinent to the potential for reactor vessel head penetration cracking.

The first of these is "...that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected." This criterion infers a licensee's responsibility to be aware of industry experience, and has been implemented in this manner in Millstone's corrective action program. The Millstone Operating Experience Program determines if industry experience applies and what, if any, corrective actions are appropriate.

The second attribute of Criterion XVI that should be considered is that for "... significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions." The bulletin suggests that for cracking of vessel head penetrations and degradation of the vessel head, the root cause determination is important in understanding the nature of the degradation and the required actions to mitigate future cracking and degradation. As part of its corrective action program, Millstone personnel would determine the cause of cracks in the vessel head penetration and/or degradation of the head.

Operating Requirement: 10 CFR 50.36 - Plant Technical Specifications

The Bulletin states:

> "Plant technical specifications pertain to the issue insofar as they do not allow operation with known reactor coolant system pressure boundary leakage."

DNC Response

Title 10 of the Code of Federal Regulations, Part 50.36 (10 CFR 50.36) contains requirements for Plant Technical Specifications. Paragraphs 2 and 3 of 10 CFR 50.36 are particularly relevant:

- 10 CFR 50.36(2) Limiting Conditions for Operation

> "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications

until the condition can be met. A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one of the following criteria:

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion **4:** A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

- 10 CFR 50.36 (3) Surveillance Requirements

  "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions will be met."

The reactor coolant pressure boundary is one of the three primary physical barriers to the release of radioactivity to the environment. Therefore, our plant Technical Specifications (TS) include a requirement and associated action statements addressing reactor coolant pressure boundary leakage.

Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," requested licensees to provide assurance that a program was implemented at their facility to ensure that boric acid corrosion due to leakage will not lead to degradation of the Reactor Coolant System Pressure Boundary. The program was to include the following attributes:

- Determination of the principal locations where leaks may occur and cause significant boric acid corrosion of the primary pressures boundary.

- Procedures for the location of small coolant leaks (i.e., leakage rates at less than technical specification limits).

- Methods for conducting examinations and performing engineering evaluations to establish the impacts on the RCS pressure boundary when leakage is located.

- Corrective actions to prevent recurrence of this type of corrosion

Millstone Unit No. 2 has a program in place that meets the required attributes of Generic Letter 88-05.

In summary, the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CEDM nozzle cracking and reactor vessel head degradation is clearly in compliance with regulatory objectives.

Question 2

Within 30 days after plant restart following the next inspection of the reactor pressure vessel head to identify any degradation, all PWR addressees are required to submit to the NRC the following information:

A. the inspection scope (if different than that provided in response to Item 1.D.) and results, including the location, size, and nature of any degradation detected,

B. the corrective actions taken and the root cause of the degradation.

DNC Response

In accordance with the Millstone 88-05 program certified Millstone Station inspection personnel conducted a thorough visual examination of all exposed external surfaces above the head for evidence of leakage and/or boron accumulations. Following this, a UT examination was conducted on all of the Reactor Vessel Head Penetration (RVHP) nozzles. The inspection scope included; 69 CEDM nozzles, eight ICI nozzles, and one vent line. The purpose of this examination was to identify any discontinuities contained within the volume of the tube material and to detect any evidence of a leak path associated with a leak between the external Alloy 600 nozzle surface and the low alloy steel vessel head penetration internal surface above the pressure boundary J-groove weld. The UT technique implemented at Millstone Unit No. 2 for the evaluation of the RVHP nozzles utilized ten transducers for the detection of axial and circumferential oriented discontinuities as well as the detection of any evidence of a leak path.

The UT examination technique was presented to the NRC Staff prior to the Millstone Unit No. 2 outage and received NRC concurrence per letter dated February 21, 2002.(1)

During the course of this inspection, three nozzles were determined to contain indications of discontinuities that could be attributed to service induced degradation. These indications required evaluation by the Millstone Station Nuclear Engineering department. Although the insulation had not been removed prior to this examination, access was obtained under the insulation for one of these nozzles. A bare metal visual examination was performed and confirmed that there was no boron residue present on the top of the vessel head near this nozzle. UT data supports the conclusion that none of the nozzles were leaking.

Liquid penetrant examinations were performed to confirm the recorded UT indications on all three of the nozzles.

Nozzle No. 21

Five axial indications and one circumferential indication were found on the outer diameter (OD) of the nozzle below the J-groove weld on the downhill side. The longest axial flaw was 2.44" and the circumferential flaw was 0.77" long. The maximum depth of any of these flaws was 0.2".

Nozzle No. 34

Four axial indications and two circumferential indications were found on the OD of the nozzle below the J-groove weld on the downhill side. The longest axial indication was 1.05" and the maximum depth of these indications was 0.15". The longest of the two circumferential indications was 0.86" and the maximum depth of these indications was 0.10".

Nozzle No. 50

Two axial indications were found on the OD of the nozzle below the J-groove weld on the downhill side. The longest of the axial indications was 1.06" and the deepest was 0.15".

The three nozzles that were evaluated by the Millstone Station Nuclear Engineering department were repaired via the temper bead weld repair process. Since all the axial flaws ran up to and possibly underneath the weld, the three nozzles were repaired by removing the lower half of the nozzle and replacing it with one made from Alloy 690. The repairs are described in detail in DNC submittals made to the NRC on February 25, 2002,[3] and March 21, 2002.[4]

After the original CEDM nozzles were machined away from the lower extent up into the volume of the vessel head, liquid penetrant examinations were conducted on the bored area of the vessel head. This surface exam interrogated the low alloy steel vessel head, the remaining J-groove weld, and the beveled portion of the original CEDM nozzle. No indications were recorded.

Following the repair, liquid penetrant and UT examinations were performed on the repair welds. These inspections recorded no indications.

---

(3) J. Alan Price letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Request to Use an Alternative to ASME Code Section XI Repair Welding Requirements by Employing Temper Bead Techniques," dated February 25, 2002.

(4) J. Alan Price letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Relief Request RR-89-34, Revision I, Response to Request for Additional Information," dated March 21, 2002.

During the course of this inspection, additional information was released by the NRC documenting severe degradation of the low alloy steel vessel head at Davis-Besse Nuclear Power Station. Following this, Millstone personnel conducted an additional thorough visual examination of all exposed external surfaces above the head, the perimeter and seams of the insulation were specifically scrutinized for signs of boric acid coming out from under the insulation or evidence of leakage from above. When the UT data from Davis-Besse became available, analysts at Millstone Station reviewed the data and then re-evaluated the Millstone data for additional assurances that a similar condition did not exist at Millstone Unit No. 2.

The benefit of using the information gathered from a UT examination of the interference fit, above the J-groove weld, is that it provides a reliable verification of the condition of the low alloy steel directly adjacent to the nozzle. This data, which was collected during the initial UT inspection, was re-evaluated to ensure the soundness *of* the low alloy steel adjacent to the nozzles.

Additional UT longitudinal wave scanning of the low alloy steel vessel head material was performed from the clad surface underside the vessel head in selected areas to provide additional assurances that this material is also in sound condition.

It is assumed that the indications found in Nozzle Nos. 21, 34 and 50 were due to primary water stress corrosion cracking of the Alloy 600. The repair process used, machined away the portion of the lower nozzle that was replaced, so it was not possible to recover any of the indications for further examination.

Docket Nos. 50-336
50-423
B18621

Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Required 15 Day Response to
NRC Bulletin 2002-01

Millstone Nuclear Power Station, Unit No. 3
NRC Bulletin 2002-01 Response

On March 18, 2002, the Nuclear Regulatory Commission (NRC) issued Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity" to all holders of operating licenses for pressurized-water nuclear power reactors. Millstone Unit No. 3 is considered as having a low susceptibility to primary water stress-corrosion cracking (PWSCC) and is ranked 56th out of 69 operating PWR's in the United States per NEI letter to the NRC dated August 21, 2001.[(1)] A 15 day response to NRC Bulletin 2002-01 was required per question 1 of the Required Information section. Below is the Millstone Unit No. 3 response.

Question 1A

A summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant,

DNC Response

Millstone Unit No. 3 conducts an inspection to comply with Generic Letter 88-05. This inspection covers two groups of components: the reactor coolant pressure boundary and the supporting systems/components that could be exposed to boric acid. The inspection of the reactor coolant pressure boundary is performed shortly after the plant reaches Mode 5 each refueling outage. The inspection of the supporting systems and components is done any time leakage is detected. This inspection is proceduralized with increased attention drawn to the joints and areas of particular importance. These inspections are performed by VT-2 qualified personnel.

Question 1B

An evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse.

DNC Response

The Generic Letter 88-05 walkdowns have proven a very effective technique for identifying leakage especially from the reactor vessel head area. If leakage is detected, the boric acid crystals are removed and the parts examined for damage. The procedure for these inspections includes criteria for various courses of action

---

(1) NEI letter to Dr. Brian Sheron, U.S. Nuclear Regulatory Commission, "Generic Information for Use by Licensees in Response to NRC Bulletin 2001-01," dated August 21, 2001.

depending upon the severity of the damage. This approach has proven effective in maintaining the leak tightness of the reactor coolant system.

Question 1C

A description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions,

DNC Response

Historically, Millstone Unit No. 3 has had one situation where boric acid could have been deposited on the reactor vessel head. In 1993, a canopy seal weld on a control rod drive mechanism (CRDM) housing was found to be leaking. Boric acid residue was observed on the CRDM housing and on top of the insulation. The residue was removed and the leak was sealed with a clamp. Additionally, a second clamp was installed on the weld from a different housing that was suspected of leaking. These clamps are still in place and the bolting torque is to be checked during the next refueling outage.

Question I D

Your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria,

DNC Response

Millstone Unit No. 3 will perform a bare metal visual inspection under the insulation during the next refueling outage, scheduled to begin September 2002, to provide a baseline of current conditions. The inspections will be conducted using remote video equipment and boroscopes, as required, which have been demonstrated to provide detailed high resolution images of the bare head under the insulation. Personnel responsible for the examination will be at least VT-2 qualified inspectors. The inspection will be documented on video tape.

In support of the bare head visual examination, Millstone Unit No. 3 will do an analysis that demonstrates that a through-wall leak in a penetration nozzle would be detected by this visual inspection.

Millstone Unit No. 3 is scheduled to inspect the two clamps put on the leaking canopy seal welds next refueling outage. A visual inspection above the head insulation will be performed at this time.

Millstone Unit No. 3 will perform boric acid walkdowns and inspections each refueling outage. The procedure for compliance with Generic Letter 88-05 was revised and made common to both Millstone Unit Nos. 2 and 3 in November 2000. This procedure will be evaluated in light of the recent experience at Davis-Besse to determine if any changes are needed. Our 60 day response to this bulletin will provide the results of our evaluation.

Question 1E

Your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in response to Item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. Include the following specific information in the discussion:

(1) If your evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met, discuss your plans for plant shutdown and inspection.

(2) If your evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

DNC Response

Part (1) - Not Applicable

Part (2) - There is reasonable assurance that regulatory requirements are currently being met. The following discussion provides the basis that all regulatory requirements discussed in the applicable Regulatory Requirements section will continue to be met until the additional inspections are performed.

Design Requirements: 10 CFR 50, Appendix A − General Design Criteria

The Bulletin states:

> "The applicable GDC include GDC 14 (Reactor Coolant Pressure Boundary)] GDC 31 (Fracture Prevention of Reactor Coolant Pressure Boundary)] and GDC 32 (Inspection of Reactor Coolant Pressure Boundary). GDC 14 specifies that the reactor coolant pressure boundary (RCPB) has an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized. GDC 32 specifies that components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leak tight integrity; inspection practices that do not permit reliable detection of degradation are not consistent with this GDC."

<u>DNC Response</u>

The three referenced General Design Criteria (GDC) state the following:

- Criterion 14 – Reactor Coolant Pressure Boundary

  "The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

- Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

  "The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (I) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient thermal stresses, and **(4)** size of flaws."

- Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

  "Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel."

During the initial plant licensing of Millstone Unit No. 3, it was demonstrated that the design of the reactor coolant pressure boundary met the regulatory requirements of the GDCs as discussed below:

- Criteria 14 Reactor Coolant Pressure Boundary

  The reactor coolant system boundary is designed to accommodate the system pressure and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stress within applicable stress limits. Reactor coolant pressure boundary materials, selection and fabrication techniques ensure a low probability of gross rupture or abnormal leakage.

- Criteria 31 Fracture Prevention of Reactor Coolant Pressure Boundary

  Close control was maintained over material selection and fabrication for the reactor coolant system to assure that the boundary behaves in a non-brittle manner. The reactor coolant system materials which are exposed to the coolant are corrosion resistant stainless steel or nickel based alloys. The reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR 50 Appendix G.

- Criteria 32 Inspection of Reactor Coolant Pressure Boundary

  The design of the reactor coolant pressure boundary provides the capability for accessibility during service life to the entire internal surfaces of the reactor vessel, certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and the bottom heads, and external surfaces of the reactor coolant piping except for the area of pipe within the primary shield wall. The inspection capability complements the leakage detection systems in the assessing the pressure boundary component's integrity. The reactor coolant pressure boundary is periodically inspected under the provisions of ASME Boiler and Pressure Vessel Code, Section XI.

- Although stress corrosion cracking of primary coolant system penetrations was not originally anticipated during plant design, it has occurred in the reactor pressure vessel (RPV) top head nozzles at some plants. The robustness of the design has been demonstrated by the small amounts of leakage that has occurred in these plants and by the fact that none of the cracks in Alloy 600 reactor coolant pressure boundary materials has rapidly propagated or resulted in catastrophic failure or gross rupture. The suitability of the originally selected materials has been confirmed. Given the inherently high fracture toughness and flaw tolerance of the Alloy 600 material, there is in fact an extremely low probability of a rapidly propagating failure and gross rupture.

- Millstone Unit No. 3 meets the ASME requirement for the J-groove CRDM welds by performing a visual examination of 25% of the penetrations for leakage during pressure testing. The component was designed for that examination and it will continue to be performed each refueling outage. In addition, that same examination will be accompanied by a bare vessel head visual examination to be performed after cool down and depressurization in the next refueling outage.

- Recent events at the Davis-Besse plant have demonstrated that the design of the reactor vessel head is very robust and that it can tolerate significant degradation without rapidly propagating failure or gross rupture. The enhanced inspection program planned for Millstone Unit No. 3 will be capable of discovering wastage or degradation *of* the reactor vessel head and will ensure continued structural and leak tight integrity.

As described above, the requirements established for design, fracture toughness, and inspectability in GDC 14, 31, and 32 respectively were satisfied during Millstone Unit No. 3's initial licensing review, and continue to be satisfied during operation, even in the presence of a potential for stress corrosion cracking of the reactor vessel head penetrations and/or subsequent wastage of the reactor vessel head.

Inspection Requirements: 10 CFR 50.55a and ASME Section XI

The Bulletin states:

> "NRC regulations at 10 CFR 50.55a state that American Society of Mechanical Engineers (ASME) Class 1 components (which includes the reactor coolant pressure boundary) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Table IWA-2500-1 [IWB-2500-1[(2)]] of Section XI of the ASME Code provides examination requirements for VHP nozzles and references IWB-3522 for acceptance standards. IWB-3522.1(c) and (d) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as 'the through-wall leakage that penetrates the pressure retaining membrane.' Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall degradation of the reactor coolant pressure vessel head penetration nozzles.
>
> For through-wall leakage identified by visual examinations in accordance with the ASME Code, acceptance standards for the identified degradation are provided in IWB-3142. Specifically, supplemental examination (by surface or volumetric examination), corrective measures or repairs, analytical evaluation, and replacement provide methods for determining the acceptability of degraded components."

DNC Response

Title 10 of the Code of Federal Regulations, Part 50.55a requires that inservice inspection and testing be performed per the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Plant Components." Section XI contains applicable rules for examination, evaluation and repair of code class components, including the reactor coolant pressure boundary.

Requirements for partial penetration welds attaching control rod drive (CRD) housings to the reactor vessel head are contained in Table IWB-2500-1, Examination Category B-E, "Pressure Retaining Partial Penetration Welds in Vessels," Item Numbers: B4.10, "Partial Penetration Welds;" B4.11, "Vessel Nozzles;" B4.12, "CRD Nozzles;" and B4.13, "Instrumentation Nozzles." The Code requires a VT-2 visual examination of 25% of the CRD nozzles from the external surface. Since the head is insulated, and the nozzles do not represent a bolted flange, paragraph IWA-5242(b) permits these inspections to be performed with the insulation left in place.

---

(2) An erratum appears to exist in the Bulletin. Table IWA-2500-1 is cited, but does not exist. It appears that the citation should have been IWB-2500-1.

In addition to the top of the insulation and above the insulation pursuant to our commitments to NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” Millstone Unit No. 3 will perform a visual inspection for evidence of leakage by examining the RPV top head surface under the insulation during the next refueling outage.

The acceptance standard for the visual examination is found in paragraphs IWA-5250, “Corrective Measures” and IWB-3522, “Standards for Examination Category B-E, Pressure Retaining Partial Penetration Welds in Vessels, and Examination Category B-P, All Pressure Retaining Components.” Paragraph IWA-5250 requires repair or replacement of the affected part if a through-wall leak is found and requires an assessment of damage, if any, associated with corrosion of steel components by boric acid. Millstone Unit No. 3 will not return to service after finding a leak from a RPV top head nozzle without first having repaired the nozzle and having assessed any wastage of the head that the leakage may have caused.

In addition, flaws identified by Non Destructive Examination (NDE) methods, which are not addressed by specific ASME Section XI acceptance criteria are evaluated in accordance with the flaw evaluation rules for piping contained in Section XI of the ASME Code. This approach has been accepted by the NRC. Any flaw not meeting requirements for the intended service period would be repaired before returning it to service.

Repairs to RPV top head nozzles would be performed in accordance with Section XI requirements, NRC approved ASME Code Case requirements, or an alternative repair or replacement method approved by the NRC.

Millstone Unit No. 3 complies with these ASME Code requirements through implementation of the inservice inspection program. If a VT-2 examination detects the conditions described by IWB-3522.1(c) and (d), then corrective actions per IWB-3142 will be performed in accordance with the plant’s corrective action program. No new plant actions are necessary to satisfy the cited regulatory criteria.

Quality Assurance Requirements: 10 CFR 50, Appendix B

The Bulletin states:

> “Criterion V (Instructions, Procedures, and Drawings) of Appendix B to 10 CFR Part 50 states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Criterion V further states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual and volumetric examinations of the reactor coolant pressure boundary are activities that should be documented in accordance with these requirements.”

DNC Response

Any of the work undertaken to inspect, evaluate, and/or repair the Millstone Unit No. 3 reactor vessel head penetrations will be conducted and documented in accordance with existing or new procedures which comply with the Company's Quality Assurance (QA) Topical Report, the QA program, and Criterion V of Appendix B to 10 CFR Part 50.

The Bulletin states:

> "Criterion IX (Control of Special Processes) of Appendix B to 10 CFR Part 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of the reactor coolant pressure boundary for the degradation observed at Davis-Besse, special requirements for visual examination and/or ultrasonic testing would generally require the use of a qualified visual and ultrasonic testing methods. Such methods are ones that a plant-specific analysis has demonstrated would result in the reliable detection of degradation prior to a loss of specified reactor coolant pressure boundary margins of safety. The analysis would have to consider, for example, the as-built configuration of the system and the capability to reliably detect and accurately characterize flaws or degradation, and contributing factors such as the presence of insulation, preexisting deposits on the RPV head, and other factors that could interfere with the detection of degradation.

DNC Response

It is expected that the designed range of interference fit of the VHP nozzles in the Millstone Unit No. 3 reactor vessel head can be shown to result in gaps between the penetration tube and bore in the vessel head at operating pressure and temperature. Consequently, flaws breaching the reactor head penetration will result in discernable leakage. This analysis will be performed prior to 3R08 in the Autumn of 2002. The visual inspection technology that Millstone Unit No. 3 will rely on is a remote robotic video system for most of the vessel head and a boroscope with video camera for any portions of the head that cannot be accessed by the robot. This type of video technology has been demonstrated to be effective at detecting small amounts of boric acid accumulation on the vessel head with sufficient resolution and sensitivity to distinguish between leakage occurring at VHP nozzles versus leakage from other sources. The inspections will be recorded on videotape. Personnel involved with the evaluation of the inspections will be VT-2 qualified and familiar with the anticipated type of indication that leakage would cause. For inspections above the head insulation, more traditional VT-2 inspection procedures with demonstrated effectiveness will be used.

Additionally, qualification of any other NDE techniques we would use for the inspections have been or will be demonstrated prior to use.

The last Appendix B criterion cited in the bulletin is:

> "Criterion XVI (Corrective Action) of Appendix B to IO CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For degradation of the reactor coolant pressure boundary, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future degradation. These actions could include proactive inspections and repair of degraded portions of the reactor coolant pressure boundary."

DNC Response

Criterion XVI contains two important attributes pertinent to the potential for reactor vessel head penetration cracking.

The first of these is "…that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected." This criterion infers a licensee's responsibility to be aware of industry experience, and has been implemented in this manner in Millstone's corrective action program. The Millstone Operating Experience Program determines if industry experience applies and what, if any, corrective actions are appropriate.

The second attribute of Criterion XVI that should be considered is that for "… significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions." The bulletin suggests that for cracking of vessel head penetrations and degradation of the vessel head, the root cause determination is important in understanding the nature of the degradation and the required actions to mitigate future cracking and degradation. As part of its corrective action program, Millstone personnel would determine the cause of cracks in the vessel head penetration and/or degradation of the head,

Operating Requirement: 10 CFR 50.36 - Plant Technical Specifications

The Bulletin states:

> "Plant technical specifications pertain to the issue insofar as they do not allow operation with known reactor coolant system pressure boundary leakage."

DNC Response

Title 10 of the Code of Federal Regulations, Part 50.36 (10 CFR 50.36) contains requirements for Plant Technical Specifications. Paragraphs 2 and 3 of 10 CFR 50.36 are particularly relevant:

- 10 CFR 50.36 (2) Limiting Conditions for Operation

  "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one of the following criteria:

  Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

  Criterion **4:** A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

- 10 CFR 50.36 (3) Surveillance Requirements

  "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions will be met."

The reactor coolant pressure boundary is one of the three primary physical barriers to the release of radioactivity to the environment. Therefore, our plant Technical Specifications (TS) include a requirement and associated action statements addressing reactor coolant pressure boundary leakage.

Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," requested licensees to provide assurance that a program was implemented at their facility to ensure that boric acid corrosion due to leakage will not lead to degradation of the Reactor Coolant System Pressure Boundary. The program was to include the following attributes:

- Determination of the principal locations where leaks may occur and cause significant boric acid corrosion of the primary pressures boundary.

- Procedures for the location of small coolant leaks (i.e., leakage rates at less than technical specification limits).

- Methods for conducting examinations and performing engineering evaluations to establish the impacts on the RCS pressure boundary when leakage is located.

- Corrective actions to prevent recurrence of this type of corrosion.

Millstone Unit No. 3 has a program in place that meets the required attributes of Generic Letter 88-05.

In summary, the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CRDM nozzle cracking and reactor vessel head degradation is clearly in compliance with regulatory objectives.