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

Dominion

SEP 11 2002

Docket No. 50-423
B18735
RE: 10 CFR 50.54(f)

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

Millstone Power Station, Unit No. 3
30 Day Response to NRC Bulletin 2002-02
Reactor Pressure Vessel Head and
Vessel Head Penetration Nozzle Inspection Programs

On August 9, 2002, the Nuclear Regulatory Commission (NRC) issued Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," to the industry. Attachment 1 to this letter provides a 30 day response with justification for continued reliance on visual examinations as the primary method for inspecting the Millstone Unit No. 3 vessel head and penetrations. Upon NRC disposition of the Electric Power Research Institute Materials Reliability Program (MRP) inspection plan, Dominion Nuclear Connecticut, Inc. (DNC) will provide additional information on future inspection plans for the Millstone Unit No. 3 vessel head.

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

Attachment (1)

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

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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 11th day of September, 2002.

Edward L Annino

My Commission Expires: Jan 31, 2006

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Attachment 1

Millstone Power Station Unit No. 3

30 Day Response to NRC Bulletin 2002-02

Attachment 1

30 Day Response to NRC Bulletin 2002-02

Dominion Nuclear Connecticut, Inc. (DNC) has evaluated the current status of Millstone Unit No. 3 with regard to the accrued effective full power years (EFPY) and the resulting effective degradation years (EDY) calculated in accordance with equation 2.2 of Electric Power Research Institute (EPRI) Document MRP-48, "PWR Materials Reliability Program Response to Nuclear Regulatory Commission (NRC) Bulletin 2001-01," dated August, 2001. An EDY is one year at 600°F while an EFPY, for the Millstone Unit 3 reactor vessel head penetrations, is one year at 558°F. Equation 2.2 of MRP-48 uses the Arrhenius relationship to equate EFPY to EDY. Section 2 of MRP-48 includes a discussion of how the Arrhenius relationship can be utilized with stress corrosion cracking issues. As of September 7, 2002, Millstone Unit No. 3 has accrued 1.88 EDY.

In a letter dated April 2, 2002,[1] DNC committed to perform a bare metal visual examination of the Millstone Unit No. 3 reactor vessel head during the next refueling outage scheduled to start on September 7, 2002. The bare metal visual examination of the reactor vessel head will be performed under the insulation using a crawler equipped with a video camera. For areas of the head that are inaccessible to the crawler, fiber optics will be used. All examination results will be documented on video tape. Hence, Millstone Unit No. 3 will be performing a bare metal visual examination of the reactor vessel head within the three years suggested in NRC Bulletin 2002-02 for a plant with less than 8 EDY.

EPRI Document MRP-75, "PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan, Revision 1," dated September 2002, provides an inspection plan for the reactor vessel head penetrations. This plan has been approved by the PWR utilities who make up the Materials Reliability Program (MRP). It presents a technically credible inspection technique that assures, to a high degree of certainty, that leaks in penetrations will be detected at an early stage long before wastage or circumferential cracking can challenge the structural integrity of the reactor coolant system (RCS) pressure boundary. Furthermore, implementation of the MRP inspection plan will assure continued compliance with the regulatory requirements cited within NRC Bulletin 2002-02. Assuming NRC approval, DNC plans to implement the MRP inspection plan and comply with its requirements for Millstone Unit No. 3 inspections which follow the September, 2002 outage.

DNC will also monitor the development of technologies for performing non-destructive examination (NDE) inspections of the reactor vessel head penetrations and the associated J-groove welds. These technologies have undergone significant change since the discovery of primary water stress corrosion cracking (PWSCC) in reactor

(1) J Alan 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.

vessel head penetrations at Oconee almost two years ago. DNC will determine which technology is best suited to Millstone Unit No. 3.

As DNC plans to follow the MRP inspection plan for Millstone Unit No. 3, the following responses are provided as justification for continued reliance on visual examinations as the primary method for inspecting the vessel head and penetrations. Included in these responses is a discussion on the reliability and effectiveness of using a visual examination as it relates to the six concerns cited in NRC Bulletin 2002-02 and the basis for concluding that unacceptable wastage will not occur between refueling outages. The Millstone Unit No. 3 response to NRC Bulletin 2002-01, dated April 2, 2002, addressed the adequacy of visual examination for compliance with the design and licensing basis of the unit. This response is still applicable.

NRC Concern 1

Circumferential cracking of CRDM nozzles was identified by the presence of relatively small amounts of boric acid deposits. This finding increases the need for more effective visual and non-visual NDE inspection methods to detect the presence of degradation in CRDM nozzles before nozzle integrity is compromised.

DNC Response

Since the initial discovery of circumferential cracks in PWRs above the J-groove welds in 2001, visual inspection techniques and approaches used have significantly improved. In addition, there is substantial experience for the range in size and appearance of visual indications that must be further investigated. Nothing in the recent event at Davis-Besse has altered the fundamental inspection capability requirements previously established as necessary to identify the presence of PWSCC and subsequent associated boric acid wastage. Inspection techniques continue to be evaluated and improved.

EPRI Document MRP-75, "PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan, Revision 1," dated September 2002, provides detailed guidance for performing visual examination of RPV heads. This guidance and lessons learned for recent field experience, including Davis-Besse establish appropriate inspection parameters and techniques that assures operability of the reactor coolant system boundary. RPV head bare metal visual inspections at Millstone Unit No. 3 will be performed and documented in accordance with written procedures. Evaluations and corrective actions will be rigorous and thoroughly documented.

For outer diameter (OD) circumferential cracks to initiate and grow above the J-groove weld, a leak path must first be established to the Control Rod Drive Mechanism (CRDM) annulus region from the inner wetted surface of the reactor vessel head. If primary water does not leak to the annulus, the environment does not exist to cause

OD circumferential cracking. Axial cracks in the CRDM nozzles above the J-groove weld or cracks in the J-groove welds must first initiate and grow through wall. Experience has shown that through wall axial cracks will result in observable leakage at the base of the penetration on the outer surface of the vessel, even with interference fits. Alloy 600 steam generator drain pipes at Shearon Harris (1998) and pressurizer instrument nozzles at Nogent 1 and Cattenom 2 (1989) were all roll expanded but still developed leaks during operation as was documented in EPRI Technical Report 1006899, "Visual Examination for Leakage of PWR Reactor Head Penetrations on Top of the RPV Head: Revision 1," dated March, 2002. Plant specific top head gap analyses have been performed for a large number of plants, with nozzle initial interference fits ranging from 0 to .0034 inches. DNC is documenting an analysis that will establish that a leak path under normal operating pressure and temperature conditions will result in observable leakage at the base of the penetration on the outer surface of the vessel.

Appendix B of EPRI Document MRP-75 "Probability of Detecting Leaks in RPV Top Head Nozzles," dated September 2002, states, "Visual inspections of the reactor coolant system pressure boundary have proven to be an effective method for identifying leakage from primary water stress corrosion cracking (PWSCC) in Alloy 600 base metal and Alloy 82/182 weld metal. Specifically, visual inspections have detected leaks in reactor pressure vessel (RPV) head CRDM nozzles, RPV head thermocouple nozzles, pressurizer heater sleeve, pressurizer instrument nozzles, hot leg instrument nozzles, steam generator drain lines, a RPV hot leg nozzle weld, a power operated relief valve (PORV) safe end and a pressurizer manway diaphragm plate." To date, no leaking CRDM nozzles have been discovered by non-visual methods except for Davis-Besse where leakage would have been detected visually had there been good access for visual inspections and the head cleaned.

Finally, as described under Concern 3 below, detailed probabilistic fracture mechanics (PFM) analyses have been performed to demonstrate the effectiveness of visual inspections in detecting circumferential cracking in the CRDM nozzles, as described in Appendix A of EPRI Document MRP-75. Even though the above discussion illustrates that visual inspections performed in accordance with MRP recommendations have a high probability of detecting through wall leakage, a very low probability of detection was assumed in the PFM analyses. To be conservative, the PFM analyses assume only a 60% probability that leakage will be detected if a CRDM nozzles is leaking at the time a visual inspection is performed. Furthermore, if a nozzle has been inspected previously, and leakage was missed, subsequent visual inspections are assumed to have only a 12% probability of detecting the leak. Even with these conservative probability of detection assumptions, the PFM analyses show that visual inspection every outage reduces the probability of a nozzle ejection to an acceptable level for plants with 18 or more EDY. Visual inspection of plants with less than 18 EDY in accordance with the MRP inspection plan will maintain the probability of nozzle ejection for these plants more than an order of magnitude lower than that for plants with greater than 18 EDY.

In summary, the industry has responded to the need to detect indications of leakage by increased visual inspection sensitivity, increased inspection frequencies, and improved inspection capabilities. Indications of leakage can be detected visually and it has been shown that timely detection by visual methods will ensure the structural integrity of the RPV head penetrations with respect to circumferential cracking.

NRC Concern 2

Cracking of 82/182 weld metal has been identified in the CRDM nozzle J-groove welds for the first time and can precede cracking of the base metal. This finding raises concerns because examination of weld material is more difficult than base material.

DNC Response

Cracks in the J-groove weld do not pose an increased risk regarding nozzle ejection as compared to penetration base metal cracks. J-groove weld cracks that initiate and grow through-wall will leak the same as cracks in the penetration base metal. Therefore, weld cracks pose a similar risk as cracks in the base material and are equally detectable by visual examination. Although higher crack growth rates have been observed in laboratory testing of weld metal, the industry model of time-to-leakage include plants that have had weld metal cracking as well as base metal cracking. The visual examination frequencies from the MRP inspection plan have been conservatively established based on the risk informed analyses considering leakage due to both weld metal and base metal cracking.

NRC Concern 3

Through-wall circumferential cracking from the outside diameter of the CRDM nozzle has been identified for the first time. This raises concerns about the potential for failure of CRDM nozzles and control rod ejection, causing a LOCA.

DNC Response

PFM analyses using a Monte-Carlo simulation algorithm were performed, as described in Appendix A of EPRI Document MRP-75, to estimate the probability of nozzle failure and control rod ejection due to through wall circumferential cracking. The PFM analyses conservatively assume that, once a leak path has extended to the annulus region, an OD circumferential crack develops instantaneously, with a length encompassing 30° of the nozzle circumference. Fracture mechanics crack growth calculations are then performed using material crack growth rate data from EPRI Document MRP-55, "Crack Growth Rates for Evaluating Primary Water Stress

Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," dated July 2002. The parameters used in the PFM model were benchmarked against the most severe cracking found to date in the industry (B&W Plants) and produced results that are in agreement with this experience to date. The analyses were used to determine probability of nozzle failure versus EFPY for various head operating temperatures. Analyses were then performed to estimate the effect of visual and NDE inspections of the plants in the most critical inspection category, using the conservative assumption discussed above (see Concern #1 response) for probability of leakage detection by visual inspection. These analyses demonstrate that performing visual inspections significantly reduces the probability of nozzle ejection, and that performing such examinations on a regular basis (in accordance with the inspection schedule prescribed in the MRP inspection plan) effectively maintains the probability of nozzle ejection at an acceptably low level indefinitely.

In the extremely unlikely event that nozzle failure and rod ejection were to occur due to an undetected circumferential crack, an acceptable margin of safety to the public would still be maintained as was discussed in the Davis-Besse Nuclear Power Station Report CR2002-0891, "Root Cause Analysis Report – Significant Degradation of the Reactor Pressure Vessel Head," dated April 2002. The consequences of such an event are similar to that of a small-break loss of coolant accident (LOCA), which is a design-basis event. The probability of core damage given a nozzle failure (assuming that failure leads to ejection of the nozzle from the head) has been estimated to be 1×10^{-3}. The PFM analyses demonstrate that periodic visual inspections are capable of maintaining the probability of nozzle failure, due to circumferential cracking, well below 1×10^{-3}. Therefore, the PFM analyses demonstrate that the resulting incremental change in core damage frequency due to CRDM nozzle cracking can be maintained at less than 1×10^{-6} (i.e., 1×10^{-3} times 1×10^{-3} equals 1×10^{-6}) per plant year, through a program of periodic visual examinations performed in accordance with the MRP inspection plan. This result is consistent with NRC Regulatory Guide 1.174 that defines an acceptable change in core damage frequency (1×10^{-6} per plant year) for changes in plant design parameters, technical specifications, etc.

Concern 4

The environment in the CRDM housing/RPV head annulus will likely be more aggressive after any through-wall leakage because potentially highly concentrated borated primary water may become oxygenated. This raises concerns about the technical basis for current crack growth rate models.

DNC Response

The MRP panel of international experts on stress corrosion cracking (SCC) (including representatives from ANL/NRC Research), prior to the Davis-Besse incident, gave extensive consideration to the likely environment in the annulus between a leaking

CRDM nozzle and the RPV head. They have subsequently revisited this issue through EPRI Document MRP-55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," dated July 2002. When revisited, the relevant arguments remain valid for leak rates that are less than 1 liter/h or 0.004 gpm, which plant experience has shown to be the usual case. The conclusions were:

1. An oxygenated crevice environment is highly unlikely because:

   - Back diffusion of oxygen is too low compared to counterflow of escaping steam (two independent assessments based on molecular diffusion models were examined).

   - Oxygen consumption by the metal walls would further reduce its concentration.

   - Presence of hydrogen from leaking water and diffusion through the upper head results in a reducing environment.

   - Even if the concentration of hydrogen was depleted by local boiling, coupling between low alloy steel and Alloy 600 would keep the electrochemical potential low.

   - Corrosion potential will be close to the Ni/NiO equilibrium, resulting in PWSCC susceptibility similar to normal primary water.

2. The most likely crevice environments are either hydrogenated steam or PWR primary water within normal specifications and both would result in similar, i.e. non-accelerated, susceptibility of the Alloy 600 penetration material to PWSCC.

3. If the boiling interface happens to be close to the topside of the J-weld, itself a low probability occurrence, concentration of PWR primary water solutes, lithium hydroxide and boric acid, can in principle occur. Of most concern here would be the accelerating effect of elevated pH on SCC, but calculations and experiments show that any changes are expected to be small, in part because of the buffering effects of precipitates. A factor of two (2) times on the crack growth rate (CGR) conservatively covers possible acceleration of PWSCC, even up to a high-temperature pH of around 9.

For higher leakage rates, which could lead to local cooling of the head, concentration of boric acid, and development of a sizeable wastage cavity adjacent to the penetration, the above arguments no longer directly apply. However, limited data (Berge et al., 1997) on SCC in concentrated boric acid solutions indicate that:

- Alloy 600 is very resistant to transgranular SCC (material design basis).

- High levels of oxygen and chloride are necessary for intergranular cracking to occur at all.

- The effects are then worse at intermediate temperatures, suggesting that the mechanism is different from PWSCC.

The above considerations show that there is no basis for assuming that any post-leakage, crevice environment in the CRDM housing/RPV head annulus would be significantly more aggressive with regard to SCC of the Alloy 600 penetration material than normal PWR primary water, irrespective of the assumed leakage rate and/or annulus geometry. The current industry model, EPRI Document MRP-55, which includes a factor of two (2) times on CGR to cover residual uncertainty in the composition of the annulus environment, remains valid.

NRC Concern 5

The presence of boron deposits or residue on the RPV head, due to leakage from mechanical joints, could mask pressure boundary leakage. This raises concerns that a through-wall crack may go undetected for years.

DNC Response

The experience at Davis-Besse has clearly demonstrated that effective visual inspection for leakage from CRDM nozzle and weld PWSCC requires unobstructed inspection access and that the head surface be free of pre-existing boric acid deposits. Accumulations of debris and boric acid deposits from other sources can interfere with a determination as to the presence or absence of boric acid deposits extruding from the tube-to-head annulus. Therefore, to effectively perform a visual examination of the RPV head outer surface for penetration leakage, such deposits and debris accumulations must be carefully inspected, removed, and the area re-inspected. Evaluation may show that it is necessary to perform additional non-visual examination(s) to establish the source of the leakage.

Accordingly, each inspection at Millstone Unit No. 3 will be conducted with a questioning attitude and any boric acid deposit on the vessel head will be evaluated to determine its source in accordance with existing industry guidance, supplemented by the most recent industry experience at the time of the inspection. These requirements are incorporated in the visual inspection guidance contained in the MRP inspection plan. Implementation of these requirements will preclude the cited condition of a through-wall crack remaining undetected for years. Since the bare metal visual inspection to be done during September, 2002 refueling outage is the first for Millstone Unit No. 3, this inspection will establish a baseline condition. Should there be any debris found that would impair the quality of the current or future visual inspections this debris will be cleaned up prior to returning to power operation in accordance with procedures.

NRC Concern 6

The causative conditions surrounding the degradation of the RPV head at Davis-Besse have not been definitively determined. The staff is unaware of any data applicable to the geometries of interest that support accurate predictions of corrosion mechanisms and rates.

DNC Response

The causes of the Davis-Besse degradation are sufficiently well known to avoid significant wastage. The root cause evaluation performed by the utility clearly identifies the root cause as PWSCC of CRDM nozzles followed by boric acid corrosion. The large extent of degradation has been attributed to failure to address evidence of leakage that had been accumulating over a five year period of time.

The industry has provided utilities with guidance for performing visual inspections of the vessel head to ensure that conditions which existed at Davis-Besse will not occur. Visual inspection guidelines have been provided, and industry meetings have been conducted to thoroughly review industry experience, regulatory requirements, leakage detection, and analytical work performed to understand the causes of high wastage rates.

Subsequent to significant wastage being discovered on the Davis-Besse RPV head, the industry has performed analytical work to determine how a small leak such as seen at several plants can progress to the significant amounts of wastage discovered at Davis-Besse. This work is referenced within the basis for the MRP inspection plan and has been presented to the NRC as Appendix C to EPRI Document MRP-75 "Supplemental Visual Inspection Intervals to Ensure RPV Closure Head Structural Integrity," dated September 2002.

The analytical work shows that the corrosion rate is a strong function of the leakage rate. Finite element thermal analyses show that leak rates must reach approximately 0.1 gpm for there to be sufficient cooling of the RPV top head surface to support concentrated liquid boric acid that will produce high corrosion rates. The leak rate is in turn a strong function of the crack length. The effect of crack length above the J-groove weld on crack opening displacement and area has been confirmed by finite element modeling of nozzles including the effects of welding residual stresses and axial cracks. Leak rates have been calculated using crack opening displacements and areas determined by the finite element analyses and leak rate models based on PWSCC cracks in steam generator tubes.

Cracks that just reach the annulus through the base metal or weld metal will result in small leaks such as those that produced small volumes of boric acid deposits on several vessel heads, where the CRDM nozzles penetrate the RPV head outside surface. These leaks are typically on the order of 10^{-6} to 10^{-4} gpm. There is no report of any of these leaks resulting in significant corrosion. A leak rate of 10^{-3} gpm will result in the release of about 500 in^3 of boric acid deposits in an 18-month operating cycle, which will be detectable by visual inspections, as described in EPRI document, MRP-75.

The time for a crack to grow from a length that will produce a leak rate of 10^{-3} gpm to a leak rate of 0.1 gpm has been estimated by deterministic analyses based on the MRP crack growth models to be 1.7 years for plants with 602°F head temperatures. Probabilistic analyses show that there is less than a 1×10^{-3} probability that corrosion will proceed to the point that the inside surface cladding of the head would be uncovered over a significant area before the wastage would be detected by supplemental visual inspections as required under the MRP inspection plan. During the transition from leak rates of 10^{-3} gpm to 0.1 gpm, loss of material will be a relatively slow process as described in Appendix C of EPRI Document MRP-75.

The ability to detect leakage prior to the risk of structural failure is illustrated by Figure 26 of the Davis-Besse root cause analysis report. There was visual evidence of boric acid deposits on the vessel head for five years prior to the degradation being detected. Guidance provided in the MRP inspection plan would not permit these conditions to exist without determining the source of the leak, including non-visual NDE examinations if necessary.

Therefore, while the exact timing of the event progression at Davis-Besse cannot be definitively established, the probable duration can be predicted with sufficient certainty to conclude that a visual inspection regimen can ensure continued structural integrity of the RCS pressure boundary.