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

Dominion

DEC 18 2002

Docket No. 50-336
B18811

RE: 10 CFR 50.54(f)

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

Millstone Power Station, Unit No. 2
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,[1] "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," to the industry. The Bulletin required a response with information on supplemental head and nozzle inspections using non-visual non-destructive examination (NDE) methods. Dominion Nuclear Connecticut, Inc. (DNC) response to the Bulletin, dated August 23, 2002,[2] stated that additional information would be submitted for Millstone Unit No. 2 by December 31, 2002. The information required by Bulletin 2002-02 is included as attachment 1 to this letter.

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): Response to NRC Bulletin 2002-02

cc: See next page

(1) U. S. Nuclear Regulatory Commission Bulletin from D. B. Matthews to the industry, "NRC Bulletin 2002-02: Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," dated August 9, 2002.

(2) W. R. Matthews letter to U. S. Nuclear Regulatory Commission, "15 Day Response to NRC Bulletin 2002-02, Reactor Pressure Vessel head and Vessel Head Penetration Nozzle Inspection Programs," dated August 23, 2002.

A096

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

B18811

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 18th day of December, 2002.

Elena L. Lockett

My Commission Expires: June 30, 2005

Attachment 1

Millstone Power Station, Unit No. 2

Response to NRC Bulletin 2002-02

Attachment 1
Response to NRC Bulletin 2002-02

Dominion Nuclear Connecticut, Inc. (DNC) submitted its response to U. S. Nuclear Regulatory Commission (NRC) Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," on August 23, 2002.(1) At that time DNC chose to delay submittal of the plans for reactor vessel head penetration and base material inspections on Millstone Unit No. 2 until December 31, 2002, to allow time for evaluation of the Fall 2002 inspections performed at other plants. Accordingly, DNC has evaluated the results and is incorporating them into the plans for the Millstone Unit No. 2 inspection as discussed below.

Requested Information:

(1) Within 30 days of this Bulletin:

A. PWR addressees who plan to supplement their inspection programs with non-visual NDE methods are requested to provide a summary discussion of the supplemental inspections to be implemented. The summary discussion should include EDY, methods, scope, coverage, frequencies, qualification requirements, and acceptance criteria.

DNC Response:

In lieu of a bare metal visual inspection, DNC intends to perform an ultrasonic (UT) inspection of 100% of the reactor pressure vessel (RPV) head nozzles and the vent line during the upcoming Millstone Unit No. 2 refueling outage, scheduled for October, 2003. The volumetric examination technique to be utilized is similar to the inspection performed during the last Unit No. 2 refueling outage as described in DNC letter dated February 18, 2002.(2) The UT inspection is capable of detecting both axial and circumferential cracks in the nozzle base material as well as finding potential leakage paths between a nozzle and the reactor vessel head.

DNC has evaluated the status of Millstone Unit No. 2 with regard to accrued Effective Full Power Years (EFPY) and Effective Degradation Years (EDY), calculated in accordance with equation 2.2 of EPRI Document MRP-48, "PWR Materials Reliability Program Response to NRC Bulletin 2001-01", dated August, 2001. As of the end of cycle 15, on or before October 11, 2003, Millstone Unit No. 2 will have accrued 12.74 EDY.

---

(1) W. R. Matthews letter to U. S. Nuclear Regulatory Commission, "15 Day Response to NRC Bulletin 2002-02, Reactor Pressure Vessel head and Vessel Head Penetration Nozzle Inspection Programs," dated August 23, 2002.

(2) J. A. Price letter to U. S. Nuclear Regulatory Commission, "Supplemental Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated February 18, 2002.

As discussed in DNC's letter dated December 28, 2001,[3] the vessel head insulation at Millstone Unit No. 2 follows the contour of the head. In order to perform a bare metal visual inspection of the RPV head nozzles this insulation would need to be removed and then replaced. At that time, DNC concluded that a bare metal visual examination was not prudent and instead proposed a UT inspection of 100% of the RPV head nozzles, including the vent line. This inspection was performed at Millstone Unit No. 2 during the refueling outage completed on April 1, 2002. The results of the inspection were discussed in DNC's letter to the NRC dated April 30, 2002.[4]

The UT inspections will be performed using a demonstrated volumetric examination technique, involving multiple transducers at varying angles to support an examination of the low alloy steel directly adjacent to the nozzles above the weld. As was shown during the last Millstone Unit No. 2 refueling outage, the absence of thermal sleeves allows the use of a probe with multiple transducers during the inspection to obtain the maximum amount of information. No obstructions were encountered during the February, 2001, refueling outage inspection that precluded gaining access to any penetration with the UT probe. Personnel performing the UT examinations will be qualified in accordance with the vendors written practices. Similar to the previous inspection, DNC will review and approve all non-destructive examination (NDE) personnel certifications and procedures prior to the examinations being performed.

Additional NDE will be performed on either or both the J groove weld and the penetration in the event that the UT inspection finds an indication requiring further interrogation or the leakage path exam results indicate a potential leak path.

The acceptance criteria for any indications found will be through the use of a "Flaw Handbook" developed specifically for Millstone Unit No. 2. This handbook incorporates the ASME flaw tolerance methods with the acceptance criteria as modified by the NRC recommendation letter ("Flaw Evaluation Criteria", Jack Strosnider, NRC to Alex Marion, NEI, November 21, 2001).

In the discussion section of Bulletin 2002-02, the NRC staff noted six specific concerns about the adequacy of the current industry RPV head and vessel head penetration (VHP) inspection programs that rely solely on visual examinations. DNC has reviewed these and believes the supplemental examinations described above, in addition to information from the EPRI MRP-75 report, addresses each of the concerns. Together this information ensures that unacceptable wastage or RPV head nozzle ejection will not occur at Millstone Unit No. 2.

---

[3] J. A. Price letter to U. S. Nuclear Regulatory Commission, "Supplemental Response to NRC Bulleting 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated December 28, 2001.

[4] J. A. Price letter to U. S. Nuclear Regulatory Commission, "Response to NRC Bulletin 2002-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated April 30, 2002.

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

As discussed above, a UT examination of all the RPV head nozzles will be performed during the Fall 2003 refueling outage. This UT examination has been demonstrated to detect circumferential cracks in the RPV head nozzles. Hence, at the end of the refueling outage, any circumferential cracking detected will have been evaluated or repaired. In addition, the precursor to circumferential cracking above the J-groove weld, leakage, either through the nozzle wall or through the weld, will have been identified and repaired. Prior to the February 2001 refueling outage, Millstone Unit No. 2 confirmed that a gap exists between the nozzle and the reactor vessel head at operating conditions allowing any leakage through a potential crack in the weld or nozzle wall to create a leakage path to the top of the vessel head which will be detectable per the UT inspection.

NRC Concern 2:

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

DNC Response

Weld cracks pose a similar risk as cracks in the base material and the effects of the cracking are equally detectable by UT examination of the interference fit region. J-groove weld cracks that initiate and grow through-wall will leak the same as cracks in the penetration base metal. Cracks in the J-groove weld do not pose an increased risk regarding nozzle ejection as compared to penetration base metal cracks. 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. As discussed in the response to concern 1, any through weld cracking will be identified with the "leak path" verification technique during the UT inspection.

NRC Concern 3

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

DNC Response

As discussed above, a UT examination of all RPV head nozzles will be performed during the October 2003 refueling outage. The UT examination techniques have been demonstrated to detect circumferential cracks in reactor vessel head (RVH) nozzles. Additionally, as noted in the response to concern 1, the precursor to circumferential cracking above the J-groove weld, leakage, will also be found with the UT examination and "leak path" verification technique.

NRC 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 2x 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 control rod drive mechanism (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 2x 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 determination of whether leakage exists from RPV head nozzles at Millstone Unit No. 2 will be made based upon information from the UT examination of the interference fit region of the penetration. An inspection of the top of the insulation as well as the portion of the penetrations above the insulation will be performed to verify that no leakage is coming onto the head from above. Hence this concern is addressed by the inspection methods being used, i.e. a UT examination and visual inspection above the insulation.

NRC Concern 6

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

DNC Response

The causes of the Davis-Besse degradation have been evaluated since Bulletin 2002-02 was distributed and are sufficiently well known to avoid significant wastage. The root cause evaluation, performed by Davis-Besse, identifies the root cause as PWSCC of CRDM nozzles followed by boric acid corrosion.

The industry has provided utilities with guidance for performing inspections of the vessel head to ensure that conditions which existed at Davis-Besse will not occur. 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 August, 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 or 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. 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.

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 inspections as required under the MRP Inspection Plan. During the transition from leakrates 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 thorough vessel head UT inspection regimen can ensure continued structural integrity of the RCS pressure boundary.