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April 1, 2002

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Three Mile Island, Unit 1 (TMI Unit 1)
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Exelon/AmerGen Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

On March 18, 2002, the NRC issued NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." This bulletin requires that the following information be submitted to the NRC within 15 days:

- plant specific information regarding a summary of the reactor pressure vessel (RPV) head inspection and maintenance programs;
- an evaluation for the ability of the inspection and maintenance programs to identify degradation of the RPV head;
- a description of any conditions identified that could have led to degradation and the corrective actions taken;
- the schedule, plans, and basis for future inspection of the RPV head and penetration nozzles; and
- a conclusion regarding whether there is reasonable assurance the applicable regulatory requirements are currently being met.

In addition, within 30 days after plant restart following the next inspection of the RPV head, information regarding the inspection scope, results, and corrective actions taken, must be submitted to the NRC. Finally, within 60 days of the date of this bulletin, information regarding the basis for concluding that the boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," and this bulletin must be submitted to the NRC.

Pursuant to 10 CFR 50.54, "Conditions of licenses," paragraph (f), Attachment 1 to this letter provides the AmerGen 15 day response for Three Mile Island, Unit 1, and Attachments 2 and 3 provide the Exelon Generation Company, LLC 15 day response for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. This response is due to the NRC by April 2, 2002.

If you have any questions or desire additional information regarding this letter, please contact me at (630) 657-2809.

Respectfully,

Jeffrey A. Benjamin
Vice President
Licensing and Regulatory Affairs

Attachments: Attachment 1, Response to NRC Bulletin 2002-01, Three Mile Island, Unit 1
Attachment 2, Response to NRC Bulletin 2002-01, Braidwood Station, Unit 1 and 2
Attachment 3, Response to NRC Bulletin 2002-01, Byron Station, Unit 1 and 2

cc: Regional Administrator – NRC Region I
Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
NRC Senior Resident Inspector – TMI

bcc: NRC Project Manager – NRR – Braidwood Station
NRC Project Manager – NRR – Byron Station
NRC Senior Project Manager – TMI Unit 1
Office of Nuclear Facility Safety – IDNS
Site Vice President – Braidwood Station
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TMI Unit 1 Nuclear Oversight Manager
TMI Unit 1 EDMS
TMI File No. 02048

STATE OF ILLINOIS	**)**	
COUNTY OF DUPAGE	**)**	
IN THE MATTER OF	**)**	
EXELON GENERATION COMPANY, LLC	**)**	**Docket Numbers**
BRAIDWOOD STATION - UNITS 1 AND 2	**)**	**STN 50-456 AND STN 50-457**
BYRON STATION - UNITS 1 AND 2	**)**	**STN 50-454 AND STN 50-455**
AMERGEN ENERGY COMPANY, LLC	**)**	**Docket Number**
THREE MILE ISLAND, UNIT 1	**)**	**50-289**

SUBJECT: **Exelon/AmerGen Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

____
Jeffrey A. Benjamin
Vice President
Licensing and Regulatory Affairs

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this ____________ day of

______________________, 20____.

____

Notary Public

ATTACHMENT 1

Response to NRC Bulletin 2002-01
“Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

Three Mile Island, Unit 1

AmerGen Energy Company, LLC (AmerGen)

Attachment 1

Response to NRC Bulletin 2002-01

Three Mile Island, Unit 1

On March 18, 2002, the NRC issued NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The below information was required within 15 days of the date of the bulletin.

"1. Within 15 days of the date of this bulletin, all PWR addressees are required to provide the following:

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

B. 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,

C. 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,

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, and

E. 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 this 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.

The information required in Item 1.A, 1.B, and 1.C, should address:

- *the material condition of the reactor pressure vessel head as determined through direct visual examinations dating back to the last time the entire reactor pressure vessel head* *was visually inspected to the bare metal. Include the date of the last 100 percent bare metal inspection, the results of that examination, and the extent and results of visual examinations conducted since the last 100 percent bare metal inspection. If no 100 percent bare metal inspection has ever been conducted, indicate so in your response.*

- *any leaks of boric acid or any other corrosive material onto the reactor pressure vessel head or insulation since the last 100 percent bare metal inspection (the results of which were provided in responding to 1.C). Include the extent to which boric acid deposits or other corrosive materials were removed from the reactor pressure vessel head, the length of time this material was left on the reactor pressure vessel head (and whether it is still on the reactor pressure vessel head), and the condition of the head following removal of the deposits. Also include a discussion of your program for preventing corrosion of the reactor pressure vessel head and the location of the leaks relative to any nozzle with through-wall cracks. If leakage was onto the insulation, discuss whether the leakage could have permeated the insulation or flowed through gaps in the insulation (e.g., around nozzles) such that deposits accumulated on the reactor pressure vessel head.*

- *the leakage integrity of the reactor pressure vessel head penetration nozzles. Include a summary of inspections performed (including scope and extent) to detect cracking and/or degradation of the vessel penetration weld or nozzle base metal, whether the inspection plan included any examination that could identify a potential cavity behind the reactor pressure vessel head nozzle, and if so, the potential for the inspection method used to accurately and reliably detect a cavity in the reactor pressure vessel head near the penetration nozzles (including the basis for this conclusion), particularly in cases where a leakage path has existed (i.e., even if the nozzle has been repaired). For repaired nozzles, the description should include the scope and results from the post-repair inspections."*

TMI Response

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

<u>Response</u>

The last visual inspection of the TMI Unit 1 reactor pressure vessel (RPV) head was a qualified bare metal visual inspection performed following the October 9, 2001, shutdown in support of the planned Refueling Outage 1R14, as described below.

Following entry into cold shutdown and removal of the RPV insulation, a qualified bare metal visual inspection of the 69 control rod drive mechanism (CRDM) nozzles and eight Thermocouple (TC) nozzle interfaces was performed. The inspection was performed in accordance with procedure ES-NDE-07T, "Visual Inspection of TMI-1 Reactor Vessel Head Penetrations," Revision 0. The inspectors were certified Level III visual examiners and specifically trained on Vessel Head Penetration (VHP) leakage observations. The special training used industry operating experience and images of leaking nozzles to demonstrate the type and quantity of boric acid crystal deposits indicative of CRDM through-wall leaks experienced at Oconee Nuclear Station (ONS), Crystal River Unit 3 (CR-3), and Arkansas Nuclear One (ANO).

In accordance with ES-NDE-07T, the initial results of the visual inspection classified the “as-found” condition of the VHP nozzle penetrations into three categories:

1. Acceptable: Those in the Acceptable category showed no evidence of leakage at the base of the nozzle and the outer RPV head surface.

2 Masked: This was an interim category. Those nozzles in the Masked category had loose debris or obstructions around the nozzle that prevented an entire 360 degree inspection. The obstruction or loose debris was vacuumed (while videotaping the area) to allow for complete inspection. The boric acid residue from leaking RPV nozzle penetrations at other stations was characterized as tightly adhering to the nozzle/head interface area. Vacuuming would not remove this type of boric acid residue. After vacuuming, the nozzle was classified as either Acceptable or Suspect. Any nozzle that remained “masked” in the area of interest (i.e., in the annular gap) was classified as Suspect and subject to subsequent Ultrasonic (UT) and Penetrant (PT) inspections.

3. Suspect: Those in the Suspect category showed signs of boric acid residue at the nozzle base. The Suspect CRDM locations were examined using a visible dye penetrant method at the surface of the J-groove weld, the outer diameter (OD) of the CRDM nozzle protruding into the RPV, and at the end of the CRDM nozzle. All Suspect CRDMs had the drives removed and a top-down ultrasonic examination was performed utilizing the Babcock & Wilcox Owners Group (B&WOG) top-down tooling. The ultrasonic examination consisted of two complete scans of each Suspect nozzle. One axial scan was used to identify circumferential flaws, and one circumferential scan to identify any axial flaws.

Boric acid deposits were located at the base of all eight TC nozzles. After reviewing tapes of the last TC nozzle inspection, all TC nozzles were deemed to be leaking and were repaired. No evidence of wastage was observed.

The initial visual inspection categorized two CRDM nozzles as Suspect; and forty-five were categorized as Masked. The Masked locations were videotaped as the loose debris was vacuumed to allow for complete inspection of the base of the CRDM nozzles. Subsequently, 10 additional CRDM nozzles were deemed Suspect. This brought the total number of Suspect CRDM nozzles requiring additional PT and UT examinations to 12. Six of the 12 Suspect CRDM nozzles required repair.

After CRDM and TC nozzle repairs and water-wash cleaning activities, a video inspection of the RPV head surface was completed to provide a baseline for future RPV head visual inspections. All boric acid residue was removed from the RPV head and no wastage to the head was observed. Videotape of the post-head cleaning was made for future reference. An in-service leakage test was performed in accordance with plant procedure 1303-8.1, “Reactor Coolant System.” The plant conditions were nominal operating pressure and temperature. No evidence of leakage was noted following a four-hour hold. Operability of the CRDM was confirmed during plant start-up in accordance with plant procedures.

This information, including the details of the PT and UT examinations, was previously submitted to NRC in a letter from AmerGen to the NRC, "AmerGen Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, Item No. 5," dated January 7, 2002.

B. *Provide 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.*

Response

A 100% bare metal head inspection was performed in 2001 during 1R14 in the as-found condition and after repairs and cleaning prior to returning the unit to service. Exelon believes the inspections performed would have identified significantly smaller degradation than discovered at Davis-Besse. In addition, during repair of the six CRDMs and eight thermocouple penetrations described in response to Item 1.C, no anomalies were identified during the work process.

The repair methodology selected for the TMI CRDM penetrations would reveal any wastage at the interface between the nozzle and vessel head material. The basis for this conclusion lies in the repair process and inspections. A portion of the nozzle is removed by machining, which then exposes the vessel material above the J-groove weld that was exposed to reactor coolant. Prior to welding, the local area of the vessel head (i.e., ID of the bored hole) was inspected with dye penetrant as part of the pre-welding inspection. Any evidence of degradation to the reactor head due to primary coolant leakage from the J-groove weld flaw would have been apparent during the pre-weld PT examinations. No problems were noted on any of the six repaired CRDMs. The cleaning activities and subsequent visual inspections on the top side of the vessel head would have revealed any degradation or wastage from above.

During the repair of two thermocouples (i.e., by replacement of the nozzle with Inconel 690 material) and the plugging of the remaining six thermocouples, no evidence of degradation or wastage to the reactor head was observed. In addition, the thermocouple nozzle repairs required PT of the vessel head prior to welding. This would have also shown degradation of the surface; however, none was observed.

TMI Unit 1 has not identified any leaks of boric acid or other corrosive material onto the RPV head since the 100% bare metal inspection performed following the October 9, 2001, refueling outage 1R14 shutdown, as described in the response to Item A above. Consequently, TMI Unit 1 has not removed any boric acid deposits or other corrosive materials from the RPV head since the 1R14 outage 100% bare metal inspection.

With respect to NRC Generic Letter 88-05,"Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," TMI Unit 1 established a program to examine all boric acid leaks discovered in the containment building and to perform an engineering evaluation of the impact of identified leaks on carbon steel or low alloy steel components. Any evidence of leakage, including dry boric acid crystals or residue, is examined and evaluated regardless of whether the leak was discovered at power or during an outage. The following issues are considered in the examination and evaluation.

1. Evidence of corrosion or metal degradation (e.g., thinning and pitting).
2. Effect the leakage may have on the pressure boundary.
3. Possibility of boric acid traveling along the inside of insulation on piping.
4. Possibility of dripping or spraying on other components.

Based on this evaluation, appropriate corrective actions are initiated to preclude recurrence of the leakage, and to repair or replace, if necessary, any degraded materials or components.

The TMI Unit 1 Augmented Inservice Inspection (ISI) Program specifies a complete RPV head inspection each refueling outage. Any boric acid residue that is observed is dispositioned in this inspection.

C. Provide 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.

<u>Response</u>

Subsequent to the 100% bare metal visual inspection described in the response to Item 1.A above, PT and UT examinations were performed on twelve suspect CRDM nozzles during refueling outage 1R14. The final engineering evaluation of the visual inspection, PT and UT data identified that five of the CRDM nozzles were leaking. The five leaking nozzles were TMI nozzles #29, #35, #37, #44, and #64. One additional nozzle, TMI CRDM Nozzle #51, was analyzed by fracture mechanics to be unacceptable for the next operating cycle. This brought the total number of CRDM nozzles requiring repair to six.

<u>PT Examinations</u>

The results of the PT examination of the J-groove weld surface identified four CRDM locations with indications. All CRDM locations with PT indications were repaired. The other eight nozzles did not exhibit any PT indication.

<u>UT Examinations</u>

Five of the 12 nozzles exhibited no flaws based on UT. The results of the UT examinations identified seven CRDM nozzles with indications. No circumferential flaws were detected in the nozzles either above or below the J-groove weld. Three of the CRDM nozzles were determined to require repair. Flaws in the other four CRDM nozzles were evaluated as acceptable.

Six CRDM and eight TC nozzles were repaired. Repairs were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, “Requirements for the Inservice Inspection of Nuclear Power Plant Components," with relief from Code requirements as approved by the NRC. All repaired CRDM nozzles were UT and PT tested with no flaws reported. A video inspection of the RPV head surface was completed after cleaning activities to provide a baseline for future visual inspections. An in-service leakage test was performed in accordance with plant procedure 1303-8.1, “Reactor Coolant System.” The plant conditions were nominal operating pressure and temperature. No evidence of leakage was noted following a four-hour hold. Operability of the CRDM was confirmed during plant start-up in accordance with plant procedures. This information, including the details of the PT and UT examinations, was previously submitted to NRC in an AmerGen letter dated January 7, 2002.

The result of these inspections and repairs provide adequate assurance of the current leakage integrity of the TMI Unit 1 RPV head penetration nozzles and assures that the type of damage seen at Davis-Besse has not occurred at TMI Unit 1.

D. Provide 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.

Response

TMI Unit 1 is currently planning to replace the RPV head in the next refueling outage, 1R15, in Fall 2003. The new RPV head will contain Alloy 690 nozzles and equivalent weld metal that will significantly reduce susceptibility to PWSCC in the head penetrations. TMI Unit 1 will continue to utilize the Augmented ISI Program that specifies a complete RPV head inspection each refueling outage. Any boric acid residue that is observed is dispositioned in this inspection.

E. Provide 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 this 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.

Response

Not Applicable. At TMI Unit 1 there is reasonable assurance that regulatory requirements are being met. See the response to part E(2).

(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.

Response

Based upon AmerGen's evaluation of the TMI Unit 1, 1R14 refueling outage 100% bare metal RPV head inspections, VHP nozzle inspections and repair, and plans for future inspections, AmerGen concludes that there is reasonable assurance that regulatory requirements will continue to be met at TMI Unit 1. The following discusses each of the criteria addressed in Bulletin 2002-01 and demonstrates that the criteria will continue to be met by TMI Unit 1. This information was previously addressed in a letter from Exelon/AmerGen to the NRC, "Exelon/AmerGen Response to NRC Bulletin 2001-01, 'Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,'" dated August 31, 2001.

Design Requirements: 10 CFR § 50, Appendix A – General Design Criteria (GDC)

The three referenced design criteria 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 TMI Unit 1, it was demonstrated that the design of the reactor coolant pressure boundary met the regulatory requirements in place at that time, as documented in the safety evaluation by the Atomic Energy Commission (AEC) dated July 11, 1973. The safety evaluation stated:

> “The Three Mile Island Unit 1 was designed and constructed to meet the intent of the AEC’s General Design Criteria, as originally proposed in July 1967. Construction of the plant was about 60% complete and the Final Safety Analysis Report (FSAR) had been filed as Amendment 12 with the Commission before publication of the revised General Design Criteria in February 1971 and the present version of the criteria in July 1971. As a result, we did not require the applicant to reanalyze the plant on the basis of the revised criteria. However, our technical review did assess the plant against the General Design Criteria now in effect and we conclude that the plant design conforms to the intent of these newer criteria.”

This demonstrates that although TMI Unit 1 was not originally designed to the present GDC, including the three GDC noted, the NRC did review and conclude that TMI Unit 1 met the intent of these criteria.

The following information demonstrates compliance with design criteria relative to the cracking of RPV top head nozzles.

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 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 GDC. The Standard Review Plans (SRPs) in effect at the time of licensing do not address the selection of Alloy 600. They only 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 have rapidly propagated or resulted in catastrophic failure or gross rupture. It should be noted that the proposed Appendix A was written in terms of extremely low probability of gross rupture or significant leakage throughout the design life.

The reactor coolant pressure boundary components at TMI Unit 1 meet this criterion. Access is provided for non-destructive examination during plant shutdown. An RPV material surveillance program conforming to this criterion has been established as described in the TMI Unit 1 Updated Final Safety Analysis Report (UFSAR), Section 4.4.5, "Material Irradiation Surveillance." The present RPV surveillance program is described in the Babcock & Wilcox Owners Group Topical Report, BAW-1543, "Master Integrated Reactor Vessel Surveillance Program," Revision 4.

As described above, the intent of the requirements established for design, fracture toughness, and inspectability in GDC 14, 31, and 32 were satisfied during the initial licensing review of TMI Unit 1, and continue to be satisfied during operation, even in the presence of the potential for primary water stress corrosion cracking (PWSCC) of the CRDM nozzle penetrations of the RPV head. In part, the selection of Alloy 600 materials provide excellent corrosion resistance and extremely high fracture toughness of the reactor coolant pressure boundary. TMI Unit 1 in the original design of the reactor head service structure had the capability to perform required ASME Code visual examinations. The TMI Unit 1 reactor head service structure provides additional access to the bare metal interface of the VHP nozzles and the RPV head to improve inspector capabilities during ASME Code required visual examinations.

<u>Operating Requirement: 10 CFR § 50.36 - Plant Technical Specifications</u>

The reactor coolant pressure boundary provides one of the critical barriers that guard against the uncontrolled release of radioactivity. Therefore, TMI Unit 1 Technical Specifications include requirements and associated action statements addressing reactor coolant pressure boundary leakage. The TMI Unit 1 Technical Specification limits for reactor coolant leakage are one gallon per minute (gpm) for unidentified leakage, 10 gpm for total leakage (i.e., identified plus unidentified leakage), and no leakage from a non-isolable fault in the reactor coolant system (RCS) pressure boundary (reference TMI Unit 1 Technical Specifications Section 3.1.6, "Leakage").

Compliance with the zero non-isolable leakage criteria is met by conducting inspections and repairs in accordance with ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Plant Components," and 10 CFR 50.55a, "Codes and standards," as described below. Specifically, during TMI Unit 1 refueling outage 1R14, the inspections performed identified all leaking penetrations which were subsequently repaired. Further, a nozzle identified with a flaw that potentially would not be acceptable for one full additional fuel cycle was also repaired. Finally, the TMI Unit 1 RPV head is planned to be replaced in refueling outage 1R15 in Fall 2003.

In addition, the unidentified leakage limit of 1 gpm is established as a quantity which can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a short time, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

Leaks from Alloy 600 RPV head penetrations due to PWSCC have been well below the sensitivity of on-line leakage detection systems. AmerGen/Exelon has evaluated this condition and have determined that the appropriate inspection for the TMI Unit 1 plant is bare-metal (VT-3) visual inspections of the reactor head for boric acid deposits during plant refueling outages. TMI Unit 1 has gaps between the CRDM nozzles and the RPV head which provides a leak path should a through-wall crack develop. This provides the ability for visual detection. The leak path coupled with the visual inspection assures that TMI Unit 1 will not have leakage from the VHPs prior to plant startup. If leakage or unacceptable indications are found, then the defect must be repaired before the plant returns to service. If through-wall boundary leaks of CRDM nozzles increase to the point where they are detected by the containment radiation monitor, mass balance calculations, reactor building sump level reading, or containment humidity monitors, then the leak must be evaluated per the specified acceptance criteria, and the plant shut down if the leak is determined to be a non-isolable RCS pressure boundary fault.

Inspection Requirements: 10 CFR § 50.55a and ASME Section XI

Title 10 of the Code of Federal Regulations, Part 50.55a, "Codes and standards," 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. On April 20, 2001, TMI Unit 1 began its third ten-year ISI interval and was required by NRC regulations to update the ISI program to meet the 1995 Code Edition with Addenda through 1996 for its third ten-year interval. The 1995 Code Edition, which applies to all third interval exams and any repairs and replacement, no longer includes Category B-E. The 1995 Code includes Category B-P, Item B15.10, "Reactor Vessel Pressure Retaining Boundary," which contains requirements for system leakage tests in accordance with IWB-5220 with visual (VT-2) examinations of the reactor pressure boundary using the acceptance standard in IWB-3522. Examinations performed as a result of the repair of any CRDM nozzles found leaking will be performed as third interval examinations in accordance with the 1995 Code with Addenda through 1996 or in accordance with relief from Code requirements granted by the NRC.

In addition to ASME Code inspections, TMI Unit 1 performs (VT-3) visual examinations of 100% of the bare metal surfaces of the reactor head in conjunction with the procedures put in place as a result of commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

The acceptance standard for the TMI Unit 1 visual examination is found in paragraph IWA-5250, "Corrective Measures." 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.

Flaws identified by nondestructive examination (NDE) methods at TMI Unit 1 which do not meet requirements have been evaluated in accordance with the Nuclear Management and Resources Council criteria and repaired. This approach has been accepted by the NRC. Any flaw not meeting requirements for the intended service period has been repaired before returning it to service.

AmerGen has performed the necessary repairs to the RPV head nozzles. The repair plans include significant reduction in exposure by instituting remote machine processes for CRDM nozzle repair(s) similar to that used at ONS Unit 2 in accordance with relief requests that have been approved by the NRC.

If a VT-2 examination detects the conditions described by IWB-3522.1(c) and (e), then corrective actions per IWB-3142 would be performed in accordance with the TMI Unit 1 corrective action program.

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

Criterion V of Appendix B to 10 CFR Part 50

Criterion V 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 VHP nozzles are activities that should be documented in accordance with these requirements.

Activities for visual inspection, NDE and repair of VHP nozzles are performed in accordance with the AmerGen Operational Quality Assurance Plan (1000-PLN-7200.01). The procedures, instructions and drawings are subject to preparation, review and approval requirements imposed through the QA Program. The QA Program meets the requirements of Appendix B.

Criterion IX of Appendix B to 10 CFR Part 50

Criterion IX 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.

The design interference fit of TMI Unit 1 RPV head VHP nozzles was designed with the same nominal interference shrink fit as the ONS and ANO units, i.e., 0.0005 to 0.0015 inches. All four plants have demonstrated that leakage does occur and can be detected. In general, TMI Unit 1 interference fits were fabricated larger than the ONS and ANO units and larger than any B&W designed unit. In actuality, TMI Unit 1 VHP nozzles were not fabricated with an interference fit. The as-built gaps open further during operation, as discussed in Material Reliability Program (MRP) Report, MRP-50 indicating that through-wall cracking of the housings will produce visually detectable evidence of leakage on the RPV head. The design of the insulation for TMI Unit 1 is such that it will not interfere with the inspector's ability to gain access for inspection of the area of interest. There are eight separate 12 inch ports in the reactor head service structure which provide access under the insulation. The insulation has adequate clearance from the bare metal surface of the head to allow unobstructed inspections.

The TMI Unit 1 visual inspections are performed by a certified ASME Level III visual examiner and trained to a site-specific procedure. CRDM nozzles for which previous leakage cannot be attributed to other sources (e.g., CRDM mechanical joints) and that could mask leakage from VHP cracking are categorized as suspect. These VHPs are then inspected with ultrasonic techniques using the best available technology.

Activities related to inspection and repair of the CRDM nozzles are controlled as required by the AmerGen Operational Quality Assurance Program for TMI Unit 1. Personnel, processes and procedures are used as required. The visual inspections of the CRDM nozzle RPV head interface are conducted by qualified inspectors using approved procedures. The inspectors are specifically trained for VHP nozzle leakage observations. Additional processes and procedures required for nondestructive examination (NDE) and other repair activities such as machining and welding are controlled in accordance with the QA program.

Criterion XVI of Appendix B to 10 CFR Part 50

Criterion XVI 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.

The identification and confirmation of a leaking VHP nozzle and/or presence of boric acid requires that the issue be appropriately identified and entered into the TMI Unit 1 Corrective Action Program (CAP). In the case of a significant adverse condition, the CAP requires determination of the cause of the failure, evaluation of the extent of condition, and assignment of appropriate corrective actions to preclude recurrence. The CAP implemented at TMI Unit 1 meets the requirements of Appendix B, Criterion XVI.

The repair and inspection approach outlined in this response provides assurance that the extent of conditions discovered are adequately addressed.

In summary, the TMI Unit 1 approach to inspection, monitoring, cause determination, and resolution of degradation of the reactor coolant pressure boundary is consistent with the performance-based objectives of Appendix B.

ATTACHMENT 2

Response to NRC Bulletin 2002-01
“Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

Braidwood Station, Units 1 and 2

Exelon Generation Company, LLC

Attachment 2

Response to NRC Bulletin 2002-01

Braidwood Station Units 1 and 2

On March 18, 2002, the NRC issued NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The below information was required within 15 days of the date of the bulletin.

"1. *Within 15 days of the date of this bulletin, all PWR addressees are required to provide the following:*

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

   B. *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,*

   C. *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,*

   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, and*

   E. *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 this 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.*

The information required in Item 1.A, 1.B, and 1.C, should address:

- *the material condition of the reactor pressure vessel head as determined through direct visual examinations dating back to the last time the entire reactor pressure vessel head* *was visually inspected to the bare metal. Include the date of the last 100 percent bare metal inspection, the results of that examination, and the extent and results of visual examinations conducted since the last 100 percent bare metal inspection. If no 100 percent bare metal inspection has ever been conducted, indicate so in your response.*

- *any leaks of boric acid or any other corrosive material onto the reactor pressure vessel head or insulation since the last 100 percent bare metal inspection (the results of which were provided in responding to 1.C). Include the extent to which boric acid deposits or other corrosive materials were removed from the reactor pressure vessel head, the length of time this material was left on the reactor pressure vessel head (and whether it is still on the reactor pressure vessel head), and the condition of the head following removal of the deposits. Also include a discussion of your program for preventing corrosion of the reactor pressure vessel head and the location of the leaks relative to any nozzle with through-wall cracks. If leakage was onto the insulation, discuss whether the leakage could have permeated the insulation or flowed through gaps in the insulation (e.g., around nozzles) such that deposits accumulated on the reactor pressure vessel head.*

- *the leakage integrity of the reactor pressure vessel head penetration nozzles. Include a summary of inspections performed (including scope and extent) to detect cracking and/or degradation of the vessel penetration weld or nozzle base metal, whether the inspection plan included any examination that could identify a potential cavity behind the reactor pressure vessel head nozzle, and if so, the potential for the inspection method used to accurately and reliably detect a cavity in the reactor pressure vessel head near the penetration nozzles (including the basis for this conclusion), particularly in cases where a leakage path has existed (i.e., even if the nozzle has been repaired). For repaired nozzles, the description should include the scope and results from the post-repair inspections."*

Braidwood Station Response

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

<u>Response</u>

Table 1 provided below lists the examinations performed in and around the reactor pressure vessel (RPV) head since the startup of the Braidwood Units. The table identifies three types of examinations: VT-2 examinations performed at normal Reactor Coolant System (RCS) pressure, VT-1 examinations of bolted connections on the RPV head, and Non Destructive Examinations (NDE) examinations (i.e., liquid dye penetrant test) performed on selected peripheral control rod drive mechanism (CRDM) housings. In all cases, these exams have not identified any evidence of leakage, boric acid deposits or boric acid corrosion on RPV components.

Review of documentation dating back to plant startup has identified two instances when RCS water was sprayed above the head area during outage fill and vent evolutions. The first incident occurred during the first Braidwood Station, Unit 1 refueling outage (i.e., A1R01) in September 1989. The second event occurred on Unit 2 during a forced outage in Spring 1994, (i.e., A2F27). In both cases the affected areas were cleaned and determined to be acceptable.

There have been 25 boric acid surveillance walkdowns performed during the refueling and forced outages at Braidwood Station. These walkdowns have utilized certified VT-2 examiners and the procedures utilized contained explicit instructions for the detection of boric acid. The current Braidwood Station procedure requires examination personnel to quantify and record all locations of boric acid residue or evidence of borated water and evaluate surface areas for degradation and wastage. Special attention is given to the RPV head canopy seal area, the reactor coolant pump studs, steam generators, pressurizer, and reactor head vent tail piece. All leakage from system components is identified, quantified and documented.

In all cases, recordable indications include evidence of borated water leakage or boric acid residue as well as any degradation of pressure boundary due to corrosion. Recordable indications require review and disposition by a VT-2 Level III examiner and if the indications are determined to be outside the procedural acceptance criteria, a Condition Report is initiated along with an associated Work Request to address the issue.

There have been 14 examinations performed on the bolted connections for the RPV head since the startup of the Braidwood Units. There are seven bolted connection vessel head penetrations (VHPs) on each RPV, two of these are for the Reactor Vessel Level Indication System (RVLIS) penetration connections and five are for the Core Exit Thermocouple (TC) penetration connections. Since all the CRDM housing to VHP connections are welded connections, the RVLIS and TC connections are the only bolted connections on the vessels. These bolted connections are currently classified as American Society of Mechanical Engineers (ASME) Section XI, 1989 Edition, Category B-G-2, Item B7.10, “Pressure Retaining Bolting, 2 inch and less in Diameter,” and require a VT-1 visual examination. In all exams performed since startup of the respective units, there has not been any evidence of erosion, corrosion, wear or boric acid residue on the bolting material and there have not been any instances of bolting material degradation due to corrosion. Also, these connections which are disassembled each refueling outage are specifically examined for leakage by VT-2 personnel prior to unit startup.

As part of the normal ISI weld inspection program, Braidwood Station examined 10 VHP housing welds (i.e., five per unit) using dye penetrant. The VHP housing welds are full penetration welds joining Inconel to stainless steel. These examinations were performed per the requirements of ASME Section XI, 1983 Edition, Summer 83 Addenda, Code Category B-O, Code Item B14.10. There were no recordable indications.

Table 1
Braidwood Station Reactor Pressure Vessel Head Examinations

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
1	9/89	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	No Recordable Indications (NRI)
1	10/89	VT-1 of the Thermocouple Bolted Connections.	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
1	10/89	VT-1 of the Reactor Vessel Level Indication bolted connections (2).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	5/91	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	11/91	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	9/92	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	1/93	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	5/93	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	10/93	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	3/94	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	4/94	VT-1 of the Thermocouple Bolted Connections.	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	4/94	VT-1 of the Reactor Vessel Level Indication bolted connections (2).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	3/95	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	9/95	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	10/96	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	3/97	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	4/97	VT-1 of the Thermocouple Bolted Connections.	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	4/97	Surface NDE of selected peripheral CRDM Housing welds.	No crack or linear indications	NRI

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
1	4/97	VT-1 of the Reactor Vessel Level Indication bolted connections (2).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	9/98	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	10/98	VT-1 of the Thermocouple Bolted Connections.	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	10/98	VT-1 of the Reactor Vessel Level Indication bolted connections (2).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	3/00	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	9/01	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	5/90	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	8/91	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	9/91	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	3/93	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	4/93	VT-1 of the Reactor Vessel Level Indication bolted connections (2).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
2	4/93	VT-1 of the Thermocouple Bolted Connections.	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
2	4/94	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	10/94	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	3/96	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	4/96	VT-1 of the Thermocouple Bolted Connections.	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
2	4/96	VT-1 of the Reactor Vessel Level Indication bolted connections (2).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
2	9/97	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	10/97	Surface NDE of selected peripheral CRDM Housing welds.	No crack or linear indications.	NRI
2	4/99	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	5/99	VT-1 of the Reactor Vessel Level Indication bolted connections (2).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
2	5/99	VT-1 of the Thermocouple Bolted Connections.	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
2	10/00	VT-2 of the accessible areas on top of the head – CRDM housings. Note 1.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI

Note 1: Exam performed per commitments in response to GL 88-05 and ASME Section XI, IWB 2500-1, Category B-P, Item B15.10 during shutdown and startup.

B. Provide 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,

Response

Braidwood Station has a thorough boric acid inspection program and has conducted all past VT-2 examinations of the Reactor Coolant Pressure Boundary (RCPB) with Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," requirements in place. To ensure compliance with this program, the RCPB, as defined by UFSAR Section 5.2, "Integrity of Reactor Coolant Pressure Boundary," has a VT-2 inspection performed by certified VT-2 examiners every refueling outage consisting of a pre-outage visual examination as well as a visual examination conducted prior to startup. These examinations are conducted to identify evidence of boric acid crystallization and residue accumulations. Additionally, Generic Letter 88-05 requirements are incorporated through the completion of normal station operator walkdowns, Maintenance and System Engineering training, the normal Inservice Inspection Program, and the Section XI System Pressure Testing Program. In addition, a heightened level of awareness to this issue was communicated to the site by Corporate Engineering during the first quarter of 2001.

Braidwood Station has established a program to inspect all boric acid leaks discovered in the containment building and to evaluate the impact of those leaks on carbon steel or low alloy steel components. Any evidence of leakage, including dry boric acid crystals or residue, is inspected and evaluated regardless of whether the leak was discovered at power or during an outage. Issues such as the following are considered in the inspection and evaluation.

1. Evidence of corrosion or metal degradation (e.g., thinning and pitting).
2. Effect the leak may have on the pressure boundary.
3. Possibility of boric acid traveling along the inside of insulation on piping.
4. Possibility of dripping or spraying on other components.

Based on this evaluation, Braidwood Station Engineering initiates appropriate corrective actions to prevent recurrence of the leak and to repair, if necessary, any degraded materials or components. In addition, work requests written on components and/or equipment with either wet or dried boric acid are uniquely coded and, after completion, are routed for Engineering review.

At Braidwood Station there have not been any examinations performed to date under the RPV head insulation for Unit 1 or Unit 2. However, considering leakage from VHPs, Braidwood Station, Units 1 and 2 are in the NRC category of plants which can be considered as having low susceptibility to VHP cracking. As reported in the Braidwood response to NRC Bulletin 2001-01, Braidwood Station, Units 1 and 2 have been ranked for the potential for primary water stress corrosion cracking (PWSCC) of the RPV top head nozzles using the time-at-temperature model and plant-specific input data reported in Material Reliability Program (MRP) Report, MRP-48. This evaluation indicates that it will take Braidwood Station, Units 1 and 2, 129.5 and 154.8 effective full power years, respectively, of additional operation from March 1, 2001, to reach the same time-at-temperature that Oconee Nuclear Station, Unit 3 had at the time that its leaking nozzles were discovered in February 2001. Because of this low susceptibility, leakage from the VHPs and subsequent accumulation of boric acid on the vessel head around the VHP is very unlikely.

C. Provide 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.

<u>Response</u>

As discussed in the response to Bulletin Question A, Braidwood has not identified leakage that could lead to degradation of the RCS pressure boundary.

D. Provide 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.

<u>Response</u>

Braidwood Station will continue to perform RPV head inspections consistent with the program discussed in Question 1.A. We are currently evaluating the extent of the reactor head examination which will be implemented in the upcoming Unit 2 refueling outage, A2R09, in

April 2002. Prior to the A2R09 outage, Braidwood Station will communicate the details of the reactor head inspection scope and will subsequently inform the NRC of the results of the inspection prior to unit startup. In addition, consistent with Item 2 of Bulletin 2002-01, within 30 days after plant restart following the next inspection of the RPV heads, we will submit information to the NRC regarding the inspection scope, the results of the inspection, any corrective actions taken and the root cause of any degradation detected. Braidwood Station will also monitor MRP recommendations and factor them into future outage planning.

E. *Provide 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 this 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.*

<u>Response</u>

Not Applicable. Braidwood Station believes there is reasonable assurance that regulatory requirements are being met. See the response to part E(2).

(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.*

<u>Response</u>

Braidwood Station has high confidence that given the current reliability (i.e., low susceptibility ranking in MRP-48) of the Braidwood Station, Unit 1 and Unit 2 VHPs, the absence of any past RCS leakage on the vessel head, the limited potential sources of boric acid leakage on the RPV, and the level of detail in current visual exams regarding detection and reporting of boric acid, there is reasonable assurance that there are not any significant amounts of boric acid deposits or corrosion on the Unit 1 or Unit 2 RPV Heads.

<u>Design Requirements: 10 CFR § 50, Appendix A – General Design Criteria (GDC)</u>

The three referenced design criteria 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."

Braidwood Station continues to be in compliance with the requirements of GDC 14, 31 and 32. Braidwood Station, Units 1 and 2 have been ranked 64th and 65th, respectively, out of 69 plants for the potential for PWSCC of the RPV top head nozzles. This ranking used the time-at-temperature model and plant-specific input data reported in MRP-48. This provides reasonable assurance the Braidwood RCPB maintains a low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture due to VHP cracking. Therefore, head wastage from leaking Inconel head penetrations is not credible concern at this time.

By design, the Braidwood Station components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leaktight integrity. These inspections are typically performed under the provisions of the ASME Boiler and Pressure Vessel Code, Section XI, "Requirements for the Inservice Inspection of Nuclear Power Part Components," as modified by the requirements of, or alternatives approved by, the NRC. While direct visual examinations of the VHP, while accessible under insulation, have not been performed in the past, they will be examined as part of an augmented program.

<u>Operating Requirement: 10 CFR § 50.36 - Plant Technical Specifications</u>

Braidwood Station Technical Specifications include requirements and associated action statements addressing reactor coolant pressure boundary leakage. The Braidwood Station Technical Specification limits for reactor coolant operational leakage are one gallon per minute (gpm) for unidentified leakage, 10 gpm for identified leakage, and no pressure boundary leakage (reference: Braidwood Station Technical Specifications Section 3.4.13, "RCS Operational Leakage).

Compliance with the zero non-isolable leakage criteria is met by conducting inspections and repairs in accordance with ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Plant Components," and 10 CFR 50.55a, "Codes and standards," as described below. In addition, the unidentified leakage limit of one gpm is established as a quantity which can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a short time, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

Leaks from Alloy 600 RPV head penetrations due to PWSCC have been well below the sensitivity of on-line leakage detection systems, however, because Braidwood Station is predicted to have very low susceptibility to VHP degradation, leakage is not expected.

Inspection Requirements: 10 CFR § 50.55a and ASME Section XI

Title 10 of the Code of Federal Regulations, Part 50.55a, "Codes and standards," 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.

Braidwood Station is currently in the 2nd ISI Inspection Interval. The 2nd ISI interval is being conducted to the 1989 Edition, no addenda, of ASME Section XI Code. Braidwood Station, Unit 1 began Interval 2 on July 29, 1998, and Unit 2 began Interval 2 on October 17, 1998. Under Code Category B-E, Code Item B4.11, the current schedule calls for the VT-2 inspection of vessel nozzles for Unit 1 in refueling outage A1R13 in Fall 2007, and in refueling outage A2R13 in Spring 2008, for Unit 2.

Quality Assurance Requirements: 10 CFR § 50, Appendix B

Criterion V of Appendix B to 10 CFR Part 50

Criterion V 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 VHP nozzles are activities that should be documented in accordance with these requirements.

ASME Code required visual and volumetric examinations including visual examination performed for Generic Letter 88-05 are performed using procedures that contain specific acceptance criteria or detailed recording criteria that are subsequently evaluated for acceptability. The visual examinations are performed using detailed instructions with a combination of qualitative and quantitative standards for the essential exam variables. Visual examination at Braidwood Station have been and currently are governed by Special Process Procedures covering both qualification of examiners and procedural requirements.

Criterion IX of Appendix B to 10 CFR Part 50

Criterion IX 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.

ASME Code required NDE and visual examinations at Braidwood Station are performed by certified Level II or Level III examiners using Level III approved procedures with additional detailed instructions as necessary.

Criterion XVI of Appendix B to 10 CFR Part 50

Criterion XVI 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."

The identification of an unacceptable NDE or visual indication requires repair, replacement or acceptance by analytical evaluation. In all these cases, these indications would be tracked by the Corrective Action Program (CAP). In the case of a significant adverse condition, the CAP requires determination of the cause of the failure, evaluation of the extent of condition, and assignment of appropriate corrective actions to preclude recurrence. The Braidwood CAP program meets the requirements of Appendix B, Criterion XVI.

ATTACHMENT 3

Response to NRC Bulletin 2002-01
“Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

Byron Station, Units 1 and 2

Exelon Generation Company, LLC

Attachment 3

Response to NRC Bulletin 2001-01

Byron Station Units 1 and 2

On March 18, 2002, the NRC issued NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The below information was required within 15 days of the date of the bulletin.

"1. *Within 15 days of the date of this bulletin, all PWR addressees are required to provide the following:*

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

   B. *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,*

   C. *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,*

   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, and*

   E. *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 this 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.*

The information required in Item 1.A, 1.B, and 1.C, should address:

- *the material condition of the reactor pressure vessel head as determined through direct visual examinations dating back to the last time the entire reactor pressure vessel head was visually inspected to the bare metal. Include the date of the last 100 percent bare metal inspection, the results of that examination, and the extent and results of visual examinations conducted since the last 100 percent bare metal inspection. If no 100* *percent bare metal inspection has ever been conducted, indicate so in your response.*

- *any leaks of boric acid or any other corrosive material onto the reactor pressure vessel head or insulation since the last 100 percent bare metal inspection (the results of which were provided in responding to 1.C). Include the extent to which boric acid deposits or other corrosive materials were removed from the reactor pressure vessel head, the length of time this material was left on the reactor pressure vessel head (and whether it is still on the reactor pressure vessel head), and the condition of the head following removal of the deposits. Also include a discussion of your program for preventing corrosion of the reactor pressure vessel head and the location of the leaks relative to any nozzle with through-wall cracks. If leakage was onto the insulation, discuss whether the leakage could have permeated the insulation or flowed through gaps in the insulation (e.g., around nozzles) such that deposits accumulated on the reactor pressure vessel head.*

- *the leakage integrity of the reactor pressure vessel head penetration nozzles. Include a summary of inspections performed (including scope and extent) to detect cracking and/or degradation of the vessel penetration weld or nozzle base metal, whether the inspection plan included any examination that could identify a potential cavity behind the reactor pressure vessel head nozzle, and if so, the potential for the inspection method used to accurately and reliably detect a cavity in the reactor pressure vessel head near the penetration nozzles (including the basis for this conclusion), particularly in cases where a leakage path has existed (i.e., even if the nozzle has been repaired). For repaired nozzles, the description should include the scope and results from the post-repair inspections."*

Byron Station Response

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

Response

At Byron Station there have been visual examinations performed under the RPV head insulation at various times for Unit 1 and Unit 2. Table 1, provided below, lists the examinations performed in and around the RPV heads since the startup of the Byron Units. The table identifies three types of examinations: VT-2 examinations performed at normal Reactor Coolant System (RCS) pressure, VT-1 examinations of bolted connections on the RPV head, and Non Destructive Examinations (NDE) exams (i.e., liquid dye penetrant) performed on selected peripheral control rod drive mechanism (CRDM) housings.

Focused inspections have been performed on the respective reactor head under the insulation, on both units following leaks that were identified to have reached the insulation and reactor head below. These leaks were discovered on January 5, 1990, during refueling outage B1R03 and on March 1, 1998, during refueling outage B1R08 on Unit 1; and on December 5, 1987, during planned outage B2P01 and on September 1, 1990, during refueling outage B2R02 on Unit 2. On Unit 1, approximately 20% of the bare metal surface was visually inspected on March 21, 2002, to confirm the inspection results of a previous leak on the reactor head vent valve discovered during refueling outage B1R03 in 1990; no boric acid accumulation or head wastage was observed.

Most recently, during the Generic Letter 88-05 RPV visual inspections, iron oxide stains on the underside of the RPV were identified. These stains were caused by water leakage from the reactor cavity boot seal during refueling operations in March 2002. No visual boric acid residue was evident and no RPV wastage was identified.

Boric acid surveillance walkdowns have been performed during the refueling outages and selected forced outages at Byron Station. These walkdowns have always used certified VT-2 examiners and the procedures utilized have always contained explicit instructions for the detection of boric acid. The current Byron Station procedure requires examination personnel to quantify and record all locations of boric acid residue or evidence of borated water and evaluate surface areas for degradation and wastage. Special attention is given to the RPV head canopy seal area, the reactor coolant pump studs, steam generators, pressurizer, and reactor head vent tail piece. All leakage from system components is identified, quantified and documented.

In all cases, recordable indications include evidence of borated water leakage or boric acid residue as well as any degradation of pressure boundary due to corrosion. Recordable indications require review and disposition by a VT-2 Level III Examiner and if the indications are determined to be outside the procedural acceptance criteria, a Condition Report is written. At Byron Station, all recordable indications are resolved and documented via a Work Request and all boric acid residue is removed.

Visual examinations are performed on the bolted connections for the RPV head since the startup of the Byron Station Units. There are seven bolted connection vessel head penetrations (VHPs) on each RPV, two of these are for the Reactor Vessel Level Indication System (RVLIS) penetration connections and five are for the Core Exit Thermocouple (TC) penetration connections. Since all the CRDM housing to VHP connections are welded connections, the RVLIS and TC connections are the only bolted connections on the vessels. These bolted connections are currently classified as ASME Section XI, 1989 Edition, Category B-G-2, Item B7.10, “Pressure Retaining Bolting, 2 inch and less in Diameter” and require a VT-1 visual examination. In all examinations performed since startup of the respective units, there has not been any evidence of erosion, corrosion, or wear of the bolting material, and there has not been any instances of bolting material degradation due to corrosion. Also, these connections, which are disassembled each refueling outage, are specifically examined for leakage by VT-2 personnel during unit startup.

As part of the normal ISI weld inspection program, Byron Station examined 10 VHP housing welds (i.e., five per Unit) using dye penetrant. These exams were performed per the requirements of ASME Section XI, 1983 Edition, Summer 83 Addenda, Code Category B-O, Code Item B14.10. There were no recordable indications.

Table 1
Byron Station Reactor Pressure Vessel Head Examinations

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
1	B1R01 (05/87)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	No Recordable Indications (NRI)
1	B1R02 (9/88)	Surface NDE of selected peripheral CRDM Housing welds	No crack or linear indications	NRI

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
1	B1R02 (10/88)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Dry Boron on CRDM @ H2. Localized only. Not on insulation. Cleaned and inspected. No wastage.
1	B1R02 (10/88)	VT-1 of TC bolted connections (2) (Greylock Coupling).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R02 (10/88)	VT-1 of TC bolted connections (2) (Replacement Marmon Coupling).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R03 (01/90)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	RC vent valve leak. Leakage onto insulation and Rx head. Vent valve replaced. Documentation of the final inspection in the localized area not initially available. Re-inspected same area in B1R11 on bare metal. No recordable indications in B1R11 inspection. No wastage. Original confirming documentation subsequently found.
1	B1R03 (01/90)	VT-1 of TC bolted connections (2) (Replacement Articu-Clamp Coupling).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R03 (02/90)	Surface NDE of selected peripheral CRDM Housing welds	No crack or linear indications	NRI
1	B1R03 (02/90)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R04 (09/91)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Boron on canopy seal welds. Residual from a previous outage (B1R03) vent valve leak. Inspected/cleaned. No wastage.
1	B1R04(11/91)	VT-1 of TC bolted connections (2) (Articu-Clamp Coupling).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R04 (11/91)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R05 (02/93)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Boron on head. Residual from previous leak (B1R03). Inspected/cleaned. No wastage.
1	B1R05 (04/93)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R06 (09/94)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R06 (10/94)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
1	B1R07 (04/96)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R07 (04/96)	Surface NDE of selected peripheral CRDM Housing welds	No crack or linear indications	NRI
1	B1R07 (06/96)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R08 (11/97)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Core Exit Thermocouple (CETC) leak identified. Leakage onto insulation and Rx head. Cleaned and inspected. No wastage.
1	B1R08 (03/98)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R09 (03/99)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R09 (04/99)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R10 (09/00)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R10 (09/00)	Surface NDE of selected peripheral CRDM Housing welds	No crack or linear indications	NRI
1	B1R10 (10/00)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R11 (03/02)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R11(03/02)	VT-1 of TC bolted connections (2) (Articu-Clamp Coupling).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R11 (03/02)	Shutdown VT-2 per GL 88-05, 20% bare metal Under Head Inspection of area affected by B1R03 leak.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	B1R11 (03/02)	Mode 5 VT-2 per GL 88-05, Underside of Reactor Vessel.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Superficial iron oxide stains. Source was leakage from reactor cavity boot seal during B1R11. No wastage.
1	B1R11 (03/02)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
2	B2P01 (12/87) Unit Surveillance Outage	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Active leak at head vent valve inside the Rx head area. Leakage onto insulation and Rx head. Cleaned and inspected with insulation removed. Inspection indicated three minor indications of localized corrosion on the head surface. All indications dispositioned as "use as is" by Westinghouse. Documented in Byron On Site Review 87-284.
2	B2R01 (02/89)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Dry boric acid residue in canopy seal area in two locations from previous leak; head vent valve area and at a CRDM. Areas were cleaned/inspected. No wastage.
2	B2R01(02/89)	Surface NDE of selected peripheral CRDM Housing welds	No crack or linear indications	NRI
2	B2R01(02/89)	VT-1 of TC bolted connections (2) (Articu-Clamp Coupling).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R01 (03/89)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R02 (09/90)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Port column assembly Articu-clamp leaking from flanged joint. WR initiated. Inspected/cleaned Rx head with insulation removed in local area. No wastage.
2	B2R02 (11/90)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R03 (02/92)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R03 (04/92)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Active Leak on southeast conoseal TC Column #5. The unit startup was stopped and the leaking flange torqued. Boric acid not on Insulation or Rx Head area. Conoseal cleaned/ inspected.
2	B2R04 (09/93)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R04(09/93)	VT-1 of TC bolted connections (2) (Articu-Clamp Coupling).	No evidence of erosion, corrosion, wear or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R04 (10/93)	Surface NDE of selected peripheral CRDM Housing welds	No crack or linear indications	NRI
2	B2R04 (10/93)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
2	B2R05 (02/95)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R05 (03/95)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R06 (08/96)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R06 (08/96)	Surface NDE of selected peripheral CRDM Housing welds	No crack or linear indications	NRI
2	B2R06 (10/96)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R07 (04/98)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R07 (05/98)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R08 (10/99)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Pinhole Leak CRDM Middle Canopy Weld. Leak was not active and not on insulation or RX head. CRDM replaced. Column and area inspected for cleanliness/damage. No wastage.
2	B2R08 (11/99)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R09 (03/01)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	B2R09 (04/01)	Post-outage VT-2 of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI

B. Provide 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.

<u>Response</u>

Byron Station has a thorough boric acid inspection program and has conducted all past examinations with Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," requirements in place. These requirements are incorporated through the completion of normal station operator walkdowns, Maintenance and System Engineering training, the normal Inservice Inspection Program, and the Section XI System Pressure Testing Program. In addition, a heightened level of awareness to this issue was communicated to the site by Corporate Engineering during the first quarter of 2001.

To ensure compliance with this program, the Reactor Coolant Pressure Boundary (RCPB), as defined by UFSAR Section 5.2, "Integrity of Reactor Coolant Pressure Boundary," has a VT-2 inspection performed by certified VT-2 examiners every refueling outage consisting of a pre-outage visual examination as well as a visual examination conducted prior to startup. These examinations shall be conducted to identify evidence of boric acid crystallization and residue accumulations.

Byron Station has established a program for Engineering to inspect all boric acid leaks discovered in the containment building and to evaluate the impact of those leaks on carbon steel or low alloy steel components. Any evidence of leakage, including dry boric acid crystals or residue, is inspected and evaluated regardless of whether the leak was discovered at power or during an outage. Issues such as the following are considered in the inspection and evaluation.

1. Evidence of corrosion or metal degradation (e.g., thinning and pitting).
2. Effect the leak may have on the pressure boundary.
3. Possibility of boric acid traveling along the inside of insulation on piping.
4. Possibility of dripping or spraying on other components.

Based on this evaluation, Byron Station Engineering initiates appropriate corrective actions to prevent recurrence of the leak and to repair, if necessary, any degraded materials or components.

These exams are conducted on the RPV head with the shroud assembly access doors opened above the RPV head insulation. Although the head insulation is in place, Byron believes these examinations are sufficient to detect and monitor boric acid accumulation for several reasons as discussed below.

At Byron Station there have not been any 100% bare metal examinations performed to date of the RPV head for Unit 1 or Unit 2. However, considering leakage from VHPs, Byron Station, Units 1 and 2 are in the NRC category of plants which can be considered as having low susceptibility to VHP cracking. As reported in the Byron Station response to NRC Bulletin 2001-01, Byron Station, Units 1 and 2 have been ranked for the potential for PWSCC of the RPV top head nozzles using the time-at-temperature model and plant-specific input data reported in Material Reliability Program (MRP) Report, MRP-48. This evaluation indicates that it will take Byron Station, Units 1 and 2 160.6 and 165.9 effective full power years, respectively, of additional operation from March 1, 2001, to reach the same time at temperature that Oconee Nuclear Station Unit 3 had at the time that its leaking nozzles were discovered in February 2001. Because of this low susceptibility, leakage from the VHPs and subsequent accumulation of boric acid on the vessel head around the VHP is very unlikely.

C. Provide 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.

<u>Response</u>

As discussed in the response to Bulletin Question A, Byron Station has identified leakage in the past on the reactor head of each unit. The corrective action in each case included the removal of insulation for cleaning and inspection to ensure no wastage had occurred. In the Byron

Station, Unit 2 planned surveillance outage, B2P01, in December 1987, minimal wastage was identified, evaluated and documented in Byron On-site Review Committee 87-284 records. The evaluation allowed the three locations to be left "as is" following cleaning. Additionally, during the Generic Letter 88-05 RPV visual inspections, iron oxide stains on the underside of the RPV were identified. These stains were caused by water leakage from the reactor cavity boot seal during refueling operations in March 2002. No visual boric acid residue was evident and no RPV wastage was identified other than superficial iron oxide. The dates and sources of the leaks that impacted the reactor head surfaces are summarized in Table 2 below.

Table 2
Byron Station Reactor Pressure Vessel Head Examinations
Identified Leaks

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results (RX Head)
1	B1R03 (01/90)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	RC Vent Valve Leak. Leakage onto insulation and RX head. Vent valve replaced. Verbal indications are that area was cleaned on top and below the insulation. Documentation not readily available of final inspection in the localized area. Re-inspected same area in B1R11 on bare metal. No recordable indications in B1R11 inspection. No wastage.
1	B1R08 (11/97)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	CETC (Core Exit Thermocouple) leak identified. Leakage onto insulation and RX head. Cleaned and inspected. No wastage.
1	B1R11 (03/02)	Mode 5 VT-2 per GL 88-05, Underside of Reactor Vessel	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Iron oxide stains. Source was leakage from reactor cavity boot seal during B1R11. No wastage.
2	B2P01 (12/87) Unit Surveillance Outage	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Active leak at Head Vent Valve inside the Rx head area. Leakage onto insulation and Rx head. Cleaned and inspected with insulation removed. Inspection indicated three minor indications of localized corrosion on the head surface. All indications dispositioned as "use as is" by Westinghouse. Documented in Byron On Site Review 87-284.
2	B2R02 (09/90)	Pre-outage VT-2 (GL 88-05) of the accessible areas on top of the head – CRDM housings.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Port Column assembly Articu-clamp leaking from flanged joint. WR initiated. Inspected/cleaned RX head with insulation removed. No wastage.

Note that all of the above leaks occurred during the course of the cycle in question, therefore, the length of time the boric acid residue was left on the RPV head was no longer than one cycle.

D. Provide 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.

Response

Byron Station will continue to perform RPV head inspections consistent with the program discussed in Question 1.A. We are currently evaluating performing an under insulation, bare metal head examination in the upcoming outages. Consistent with Item 2 of Bulletin 2002-01, within 30 days after plant restart following the next inspection of the RPV heads, we will submit information to the NRC regarding the inspection scope, the results of the inspection, any corrective actions taken and the root cause of any degradation detected. Byron Station will also monitor MRP recommendations and factor them into future outage planning.

E. Provide 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 this 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.

Response

Not Applicable. Byron Station believes there is reasonable assurance that regulatory requirements are being met. See the response to part E(2).

(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.

Response

Byron Station has high confidence that given the current reliability of the Byron Unit 1 and Unit 2 VHPs, the identification and documentation of any past RCS leakage on the RPV head, the limited potential sources of boric acid leakage on the RPV, and the level of detail in current visual exams regarding detection and reporting of boric acid, there is reasonable assurance that there are not any significant amounts of boric acid deposits or corrosion on the Unit 1 or Unit 2 RPV heads. The partial bare metal head examination on Unit 1 that was performed in March 2002 during the B1R11 refueling outage, validates this position for Unit 1.

<u>Design Requirements: 10 CFR § 50, Appendix A – General Design Criteria (GDC)</u>

The three referenced design criteria 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."

Byron Station continues to be in compliance with the requirements of GDC 14, 31 and 32. Byron Station, Units 1 and 2 have been ranked 66th and 67th respectively out of 69 plants for the potential for PWSCC of the RPV top head nozzles. This ranking used the time-at-temperature model and plant-specific input data reported in MRP-48. This provides reasonable assurance the Byron Station RCPB maintains a low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture due to VHP cracking. Therefore, head wastage from leaking Inconel head penetrations is not credible concern at this time.

By design, Byron Station components that are part of the RCPB have the capability of being periodically inspected to assess their structural and leak tight integrity. These inspections are typically performed under the provisions of the ASME Boiler and Pressure Vessel Code, Section XI, “Requirements for the Inservice Inspection of Nuclear Power Part Components,” as modified by the requirements of or alternatives approved by the NRC. Direct visual examinations of the VHP, while accessible under insulation, have not been performed in the past, they will be examined as part of an augmented inspection program.

<u>Operating Requirement: 10 CFR § 50.36 - Plant Technical Specifications</u>

Byron Station Technical Specifications include requirements and associated action statements addressing reactor coolant pressure boundary leakage. The Byron Station Technical Specification limits for reactor coolant operational leakage are one gallon per minute (gpm) for unidentified leakage, 10 gpm for identified leakage, and no pressure boundary leakage (reference Byron Station Technical Specifications Section 3.4.13, "RCS Operational Leakage").

Compliance with the zero non-isolable leakage criteria is met by conducting inspections and repairs in accordance with ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Plant Components," and 10 CFR 50.55a, "Codes and standards," as described below. In addition, the unidentified leakage limit of one gpm is established as a quantity which can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a short time, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

Leaks from Alloy 600 RPV head penetrations due to PWSCC have been well below the sensitivity of on-line leakage detection systems, however, because Byron Station is predicted to have very low susceptibility to VHP degradation, leakage is not expected.

<u>Inspection Requirements: 10 CFR § 50.55a and ASME Section XI</u>

Title 10 of the Code of Federal Regulations, Part 50.55a, "Codes and standards," 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.

Byron Station is currently in the 2nd ISI Inspection Interval on both units. The 2nd ISI interval is being conducted to the 1989 Edition, no addenda, of ASME Section XI. Byron Station Unit 1 began Interval 2 on June 30, 1996, and Unit 2 began Interval 2 on August 16, 1998. Under Code Category B-E, Code Item B4.11, the current schedule calls for the VT-2 inspection of RPV nozzles for Unit 1 in B1R13 (Spring 2005) and in B2R13 (Spring 2007) for Unit 2.

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

Criterion V of Appendix B to 10 CFR Part 50

Criterion V 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 VHP nozzles are activities that should be documented in accordance with these requirements.

ASME Code required visual and volumetric examinations including visual examination performed for Generic Letter 88-05 are performed using procedures that contain specific acceptance criteria or detailed recording criteria that are subsequently evaluated for acceptability. Visual examinations are performed using detailed instructions with a combination of qualitative and quantitative standards for the essential exam variables. Visual examinations at Byron Station have been and currently are governed by Special Process Procedures covering both qualification of examiners and procedural requirements.

Criterion IX of Appendix B to 10 CFR Part 50

Criterion IX 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.

ASME Code required NDE and visual examinations at Byron Station are performed by certified Level II or Level III examiners using Level III approved procedures with additional detailed instructions as necessary.

Criterion XVI of Appendix B to 10 CFR Part 50

Criterion XVI 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.

The identification of an unacceptable NDE or visual indication requires repair, replacement or acceptance by analytical evaluation. In all cases, these indications would be tracked by the Corrective Action Program (CAP). In the case of a significant adverse condition, the CAP requires determination of the cause of the failure, evaluation of the extent of condition, and assignment of appropriate corrective actions to preclude recurrence. The Byron CAP program meets the requirements of Appendix B, Criterion XVI.