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August 31, 2001

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 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"

On August 3, 2001 the NRC issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," requesting plant specific information regarding the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles including the extent of VHP nozzle leakage and cracking that has been found to date. Information is also requested regarding inspections and repairs that have been completed and those planned in the future to satisfy regulatory requirements; and the basis for concluding that those plans will ensure compliance with the applicable regulatory requirements.

Prior to issuance of the NRC Bulletin, AmerGen/Exelon had initiated plans to address the VHP cracking issue. In addition, we have been active in the Materials Reliability Program Alloy 600 Task Group sponsored by the Electric Power Research Institute and owners groups.

Pursuant to 10 CFR 50.54, "Conditions of licenses," paragraph (f), Attachment 1 to this letter provides the AmerGen response for Three Mile Island, Unit 1 and Attachments 2 and 3 provide the Exelon Response for Braidwood Station Units 1 and 2, and Byron Station Units 1 and 2, respectively. Attachment 4 provides a summary of the commitments contained in this letter.

A088

If any further information is needed please feel free to contact me at (630) 657-2809.

Respectfully,

Jeffrey A. Benjamin
Vice President
Licensing and Regulatory Affairs

Attachments: Attachment 1, Response to NRC Bulletin 2001-01, Three Mile Island, Unit 1
Attachment 2, Response to NRC Bulletin 2001-01, Braidwood Station, Unit 1 and 2
Attachment 3, Response to NRC Bulletin 2001-01, 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

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: **Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"**

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 31st day of

August, 2001.

Timothy Byam
Notary Public

"OFFICIAL SEAL"
Timothy A. Byam
Notary Public, State of Illinois
My Commission Expires 11/24/2001

ATTACHMENT 1

Response to NRC Bulletin 2001-01 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

Three Mile Island, Unit 1

AmerGen Energy Company, LLC (AmerGen)

Attachment 1

Response to NRC Bulletin 2001-01

Three Mile Island, Unit 1

NRC requested information

1. *All addressees are requested to provide the following information:*

   a. *the plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;*

   Response:

   TMI Unit 1 has been ranked for the potential for primary water stress corrosion cracking (PWSCC) of the reactor pressure vessel (RPV) top head penetration (VHP) nozzles using the time-at-temperature model and plant-specific input data reported in MRP-48 (Reference 1). The methodology used in MRP-48 is the same as was described previously in MRP-44, Part 2. As shown in Table 2-1 of MRP-48, this evaluation indicates that it will take TMI Unit 1 approximately 4.1 effective full power years (EFPY) of additional operation from March 1, 2001, to reach the same time-at-temperature that Oconee Nuclear Station Unit 3 (ONS3) had at the time that its leaking VHP nozzles were discovered in February 2001.

   Therefore, TMI Unit 1 is in the NRC category of plants which can be considered as having high susceptibility based on a susceptibility ranking of less than 5 EFPY from the ONS3 condition.

   b. *a description of the VHP nozzles in your plant(s) including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles;*

   Response:

   The requested nozzle information is provided in Table 2-3 of MRP-48. There are 69 CRDM nozzles and 8 thermocouple (TC) nozzles. One TC nozzle is being used for the Reactor Coolant Inventory Tracking System (RCITS), one TC nozzle is being used for the reactor head vent, and the other six TC nozzles are not being used.

   The typical Babcock and Wilcox design reactor vessel head is depicted in Figure 1-1 of MRP-48. All the vessel head penetrations are SB-167 material. The J-groove weld was fabricated with Alloy 182 weld metal. The center to center distance between most penetrations is approximately 12 inches. Accounting for the diameters of the penetrations, the minimum outer surface to outer surface distance between the VHPs is approximately 8 inches.

c. *a description of the RPV head insulation type and configuration;*

Response:

As reported in Table 2-1 of MRP-48, TMI Unit 1 has reflective horizontal RPV head insulation. The minimum clearance between the bottom of the insulation and the dome of the reactor vessel head surface is approximately 2 inches. In addition, TMI Unit 1 has eight 12 inch diameter access ports in the service structure. Accessibility for visual inspection of the bare metal RPV at the VHP's is greatly enhanced by the large insulation clearances as the dome tapers away from the insulation and the access ports allowing thorough visual inspection underneath the insulation of the head surface surrounding the VHPs.

d. *a description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;*

Response:

As reported in Table 2-1 of MRP-48, TMI Unit 1 has performed RPV head and nozzle inspections within the past four years.

Within the last four years, during refueling outages 1R12 (October 1997) and 1R13 (October 1999), TMI Unit 1 has performed visual examinations of the bare metal (VT-3) VHP nozzle/reactor vessel head interface areas below the vessel head insulation. The examinations were performed by American Society of Mechanical Engineers (ASME) certified visual examiner(s) and were intended to identify any evidence of leakage including boric acid deposits. The examinations were performed by a remote visual technique. The VHP nozzle/reactor vessel head interface area is accessible through eight separate 12 inch diameter access openings in the reactor head service structure. In addition, ASME Code required VT-2 examinations were performed prior to return to service following each refueling outage. A summary of those inspections performed within the last four years are listed in the table below.

TMI Unit 1 VHP Nozzle/RPV Head Inspections

Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
10/97	VT-2 of the RV closure for evidence of leakage of CRD and thermocouple nozzles. Note 1.	No evidence of leakage or boric acid residue.	No Recordable Indications (NRI)
10/97	VT-3 of the Reactor Vessel Head under the insulation	No evidence of leakage or boron residue. No degradation due to corrosion.	Summary: No active leakage was found. Boron was observed at several CRDM locations. As indicated by boron trails, the source of the leakage was determined to be from CRDM flange leakage above. All identified flange leakage was repaired.
10/99	VT-2 of the RV closure for evidence of leakage of CRD and thermocouple nozzles. Note 1	No evidence of leakage or boric acid residue.	NRI
10/99	VT-3 of the Reactor Vessel Head under the insulation	No evidence of leakage or boron residue. No degradation due to corrosion.	Summary: No active leakage was found. Boron was observed at several CRDM locations. As indicated by boron trails, the source of the leakage was determined to be from CRDM flange leakage above. All identified flange leakage was repaired.

Notes:
(1) Inspections performed in accordance with commitments in response to Generic Letter 88-05 and ASME Section XI, Category B-E, Item B.4.12.

e. *A description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.*

Response:

Missile Shield

The reactor vessel missile shield consists of three separate 8 x 29 x 1.5 feet concrete blocks, weighing 26 tons each. These blocks rest beside each other on the D-ring ledge directly above the reactor. The bottom of the missile shield is located at the 358' elevation, and the reactor vessel head flange is at 321' elevation. The shield prevents any missiles that may be ejected from atop the head from penetrating other reactor coolant system pressure boundaries and/or the containment structure. The missile shields are removed each refueling to provide access to the CRDMs and reactor vessel head removal. There is no direct contact between the missile shields and the reactor head service structure.

CRDM Housings

The full-length Control Rod Drive Mechanism (CRDM) consists of seven major subassemblies:

1. Motor Tube Assembly
2. Stator-Water Jacket Assembly
3. Leadscrew Assembly
4. Torque Tube Assembly
5. Position Indicator Assembly
6. Thermal Barrier
7. Rotor Assembly

The motor tube assembly forms part of the primary pressure boundary when bolted (eight bolt double gasket flange assembly) to the reactor head flange. This connection between the motor tube assembly and the reactor head flange is at the 329' elevation. The motor tube houses the rotor assembly, leadscrew assembly, torque tube assembly, and the thermal barrier assembly. The thermal barrier assembly is held within the base of the motor tube and restricts the thermal circulation of hot primary coolant between the reactor vessel and the drive.

The stator-water jacket assembly provides cooling for the stator windings and drive assembly. The cooling water is supplied from the Intermediate Closed Cooling System from piping connections on top of the service structure at the 348' elevation.

The position indicator assembly is used to determine the absolute position of the leadscrew within the drive. The position indicator assembly is attached to the motor tube.

Reactor Head Service Structure

The CRDMs are mounted inside a protective cylinder (service structure) bolted to and projecting up from the reactor vessel head. Twelve small diameter fans are mounted externally on the cylinder for cooling of the CRDM electrical components. The top ends of the CRDMs are restrained for seismic loading purposes. A grid of steel channel and perforated metal plates securely attached to the service structure at the top of the CRDMs provide a working floor. No other safety related components are mounted within the service structure.

The Reactor Coolant Inventory Tracking System and Reactor Vessel Head Vent System are attached to two thermocouple nozzles and their piping/tubing lines run up the side of the service structure.

Cabling

The power and instrumentation cables are located at approximately the 346' elevation. The cables extend from the top of reactor service structure across to various trays on the D-ring wall. These consist of power cables for the CRDMs, for cooling fans, and instrumentation cables for the Control Rod Position Indicators. No safety related components lie between the upper ends of the CRDMs and the bottom of the missile shields.

2. *If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide the following information:*

   a. *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

      Response:

      This item is not applicable to TMI Unit 1.

   b. *a description of the additional or supplemental inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken in response to identified cracking to satisfy applicable regulatory requirements;*

      Response:

      This item is not applicable to TMI Unit 1.

   c. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

      Response:

      This item is not applicable to TMI Unit 1.

   d. *your basis for concluding that the inspections identified in 2.c will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

      (1) If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspection are performed.

      Response:

      This item is not applicable to TMI Unit 1.

      (2) If your future inspection plans do not include volumetric examination of all VHP nozzles, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will be satisfied.

      Response:

      This item is not applicable to TMI Unit 1.

3. *If the susceptibility ranking for your plant is within 5 EFPY of ONS3, addressees are requested to provide the following information:*

   a. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

Response:

As discussed in previous AmerGen submittals (i.e., References 2, 3, and 4), visual inspections to detect leakage/boric acid deposits on all Reactor Vessel Head Thermocouple (TC) nozzles and Control Rod Drive Mechanism (CRDM) nozzles will be conducted during the upcoming TMI Unit 1 refueling outage (1R14), which is currently scheduled to begin on October 9, 2001. Visual inspections will be performed by an ASME certified Level III visual inspector. Any through-wall cracking will have sufficient ability to leak to the head surface surrounding the VHPs for visual detection since the area is open and not masked by insulation or existing deposits. Thus, the visual inspections at TMI Unit 1 will provide high confidence that any leaks will be detected. 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 will be categorized as suspect. These VHPs will be inspected with ultrasonic techniques using the best available technology. Any relevant indications in the VHPs will be evaluated to the industry developed (i.e., Nuclear Management and Resources Council) flaw acceptance criteria. All VHPs identified with unacceptable flaws will be repaired prior to return to service.

Any CRDM nozzle found leaking will be characterized using an Ultrasonic Testing (UT) technique (i.e., best available technology) in the nozzle and Penetrant Testing (PT) on the associated J-groove welds. The repair plans seek to significantly reduce exposures by instituting remote machine processes for CRDM nozzle repair similar to that used at the Oconee Nuclear Station – Unit 2. The repairs will be performed in accordance with ASME Code requirements, NRC approved Code Cases, or specific relief from Code requirements requested to be approved by the NRC in References 2 and 4.

Reference 3 informed the NRC of detailed inspection plans for the 1R14 outage at TMI Unit 1. We are continuing to work with the industry organizations and other owners to incorporate Oconee lessons learned and this bulletin response provides an update on our inspection plans for the TMI Unit 1 outage 1R14, currently scheduled to begin October 9, 2001.

All 15 leaks found to date have been in Babcock and Wilcox designed plants (TMI Unit 1 type reactor designs) and have been attributed to PWSCC in Alloy 600 CRDM nozzles. All of these 15 leaking nozzles have been discovered by visual inspection for boric acid crystal deposits.

TMI Unit 1 fabrication records were reviewed and it was determined that there are as-built clearances between the CRDM nozzles and the reactor head penetrations. Framatome review of the final QA inspection reports found measurement data for CRDM bores which were used to calculate the actual top and bottom CRDM dimensional fits by subtracting the CRDM nozzle shaft diameter from the CRDM bore diameter. The clearances calculated for TMI Unit 1 ranged from approximately 0.0005 inches to 0.0070 inches as documented in Reference 5. During plant operation, this clearance increases due to temperature and pressure dilation, as noted in Reference 7, which provides a leak path from a through-wall crack to the surface of the vessel head and subsequent detection by visual inspection. Therefore, the original fabrication clearances along with the increase in the clearance during operation provides a leakage path from a through-wall crack to the surface of the RPV head surface around the VHP.

Plans for the TMI Unit 1 1R14 Outage include:

- Both direct and remote visual examinations will be performed using ASME Section XI IWA-2210 as a guideline utilizing industry experience gained from the Oconee and ANO inspections. Access ports in the service structure provide adequate access to the top of the head. Each of the 69 CRDM and 8 Thermocouple nozzle locations will be accessed through these ports under the insulation to perform the visual examinations.

- Personnel performing this task will be trained on the type of unacceptable conditions found at Oconee and ANO. The examination crew will consist of AmerGen/Exelon personnel with at least one person Certified as an ASME Level III Visual examiner along with the cognizant engineer.

- Inspections will be performed in accordance with a procedure developed specifically for these examinations that will meet the basic requirements of an ASME VT-3 inspection. The EPRI MRP Visual Training Package will also be used to further enhance the execution of the inspection.

- Inspections from the TMI Unit 1 outage 1R13 were recorded on videotape. Those videotapes will be used to help identify changes in conditions.

- Any CRDM VHPs that are categorized as suspect leakers will have an ultrasonic test (i.e., using the best available technology) performed from the inside diameter (ID) surface of the nozzle. All defects detected that do not meet ASME Section XI code acceptance criteria will be repaired.

- All CRDM nozzles requiring repair will have the cracks characterized. If any circumferential cracks are detected above the J-groove weld then an expansion of ID ultrasonic inspection (i.e., using the best available technology) will be performed. The CRDM VHPs with leakage will require removal of adjacent CRDMs to accommodate repairs. The CRDMs removed as a result of the repairs define the extent of the expansion.

- The top of the reactor head will be cleaned to remove existing boron deposits and videotaped prior to return to service to re-establish a baseline for future inspections.

AmerGen will continue to perform visual examinations, as described in this document, on the existing RPV head for the evidence of leaking CRDM nozzles during subsequent refueling outages. In addition, new technology developed in the future will be evaluated for suitability for application. TMI Unit 1 type reactor design experience, coupled with this inspection plan, provides a high confidence that any potential leakage will be identified.

b. *your basis for concluding that the inspections identified in 3.a. will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

   (1) If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

   (2) If your future inspection plans include only visual inspections, discuss the corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.

Response:

TMI Unit 1 will perform inspections of the reactor vessel head nozzles during the next refueling outage, which is currently scheduled to begin on October 9, 2001 as discussed in response to item 3.a above.

The visual inspection of the TMI Unit 1 VHPs is the proven technique that provides a qualified examination for PWSCC degradation mechanism. VHPs identified with leakage will be characterized using the best available Ultrasonic Testing (UT) technique and Penetrant Testing (PT) methods. In addition, if a circumferential crack above the J-groove weld is detected in CRDM nozzles found to be leaking, those CRDMs that are made accessible (i.e., as a result of the visual inspection findings or to permit repairs) will be examined using the best available UT technique. If an unacceptable flaw is found in a non-leaking CRDM nozzle, the extent of condition will be evaluated in accordance with the TMI Unit 1corrective action program. All defects found that do not meet the acceptance criteria established by Nuclear Management and Resources Council, will be repaired.

The visual inspection described in this submittal will detect cracks that develop before they become structurally significant. MRP 2001-050 (i.e., Reference 7) indicates that the Oconee nozzles would have taken more than 4-5 EPFY to reach the structural margin. TMI Unit 1 is approximately 4 EFPY from reaching the Oconee Unit 3 time-at-temperature when the cracks were detected. Therefore, TMI Unit 1 is not expected to have any structurally significant flaws.

The Bulletin section entitled Applicable Regulatory Requirements cites the following regulatory requirements as providing the basis for the bulletin assessment:

- Appendix A to 10 CFR Part 50, *General Design Criteria for Nuclear Power Plants*

  Criteria 14 – *Reactor Coolant Pressure Boundary*

  Criteria 31 – *Fracture Prevention of Reactor Coolant Pressure Boundary, and*

  Criteria 32 – *Inspection of Reactor Coolant Pressure Boundary*

- Plant Technical Specifications
- 10 CFR 50.55a, Codes and Standards, which incorporates by reference Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code*

- Appendix B of 10 CFR Part 50, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Criteria V, IX, and XVI*

The following discusses each of these criteria and demonstrates that the criteria will be met by TMI Unit 1.

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

The Bulletin states:

"The applicable GDC include GDC 14, GDC 31, and GDC 32. GDC 14 specifies that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 32 specifies that components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leak tight integrity; inspection practices that do not permit reliable detection of VHP nozzle cracking are not consistent with this 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 as follows: "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 mentioned, 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 General Design Criteria. The SRPs (standard review plans) in effect at the time of licensing do not address the selection of Alloy 600. They only require that ASME code requirements be satisfied.
- Although stress corrosion cracking of primary coolant system penetrations was not originally anticipated during plant design, it has occurred in the RPV top head nozzles at some plants. The robustness of the design has been demonstrated by the small amounts of the leakage that has occurred and by the fact that none of the cracks in Alloy 600 reactor coolant pressure boundary materials has rapidly propagated or resulted in catastrophic failure or gross rupture. 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. A reactor pressure vessel material surveillance program conforming to this criterion has been established as described in TMI Unit 1 Updated Final Safety Analysis Report (UFSAR) Section 4.4.5. The present reactor vessel surveillance program is described in B&WOG Topical Report BAW-1543.

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 PWSCC of the CRDM nozzle penetrations of the reactor vessel 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 reactor vessel head to improve inspector capabilities during ASME Code required visual examinations.

Operating Requirement: 10 C.F.R. § 50.36 - Plant Technical Specifications

The Bulletin states:

"Plant technical specifications pertain to the issue of VHP nozzle cracking insofar as they require no through-wall reactor coolant system leakage."

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

- 10CFR 50.36 (2) Limiting Conditions for Operation

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

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

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

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

- 10 CFR 50.36 (3) Surveillance Requirements

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

The reactor coolant pressure boundary 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 pressure boundary leakage are 1 gallon per minute (gpm) for unidentified leakage, 10 gpm for total leakage (identified plus unidentified leakage), and no leakage from a non-isolable fault in the reactor coolant system pressure boundary (Reference: TMI Unit 1 Technical Specifications Section 3.1.6). 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 matter of hours, 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 reactor vessel 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 reactor vessel 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 form the VHPs prior to plant startup. If leakage or unacceptable indications are found, then the defect must be repaired before the plant goes back on line. 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 reactor coolant system pressure boundary fault.

<u>Inspection Requirements: 10 C.F.R. § 50.55a and ASME Section XI</u>

The Bulletin states:

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

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

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

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

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 response to AmerGen request for relief from Code requirements (i.e., References 2 and 4).

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* (i.e., Reference 6).

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 will be evaluated in accordance with the Nuclear Management and Resources Council criteria will be repaired. This approach has been accepted by the NRC. Any flaw not meeting requirements for the intended service period will be repaired before returning it to service.

AmerGen intends to perform any 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 ONS2 in accordance with relief requests that have been submitted to the NRC for approval (i.e., References 2 and 4).

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 C.F.R. § 50, Appendix B</u>

The Bulletin states:

> *"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. Within the context of providing assurance of the structural integrity of VHP nozzles, special requirements for visual examination would generally require the use of a qualified visual examination method. Such a method is one that a plant-specific analysis has demonstrated will result in sufficient leakage to the RPV head surface for a through-wall crack in a VHP nozzle, and that the resultant leakage provides a detectable deposit on the RPV head. The analysis would have to consider, for example, the as-built configuration of the VHPs and the capability to reliably detect and accurately characterize the source of the leakage, considering the presence of insulation, preexisting deposits on the RPV head, and other factors that could interfere with the detection of leakage. Similarly, special requirements for volumetric examination would generally require the use of a qualified volumetric examination method, for*

example, one that has a demonstrated capability to reliably detect cracking on the OD of the VHP nozzle above the J-groove weld."

The design interference fit of TMI Unit 1 reactor vessel head VHP nozzles was designed with the same nominal interference shrink fit as the Oconee and ANO units, 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 Oconee 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 MRP-50 (see Reference 7), 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 inspectors 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 inspection will be 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 will be categorized as suspect. These VHPs will be inspected with ultrasonic techniques using the best available technology.

Activities related to inspection and repair of the CRDM nozzles will be controlled as required by the AmerGen Operational Quality Assurance Program for TMI Unit 1. Personnel, processes and procedures will be used as required. The visual inspections of the CRDM nozzle reactor vessel head interface will be conducted by qualified inspectors using approved procedures. The inspectors will be 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 will be controlled in accordance with the QA program.

The Bulletin further states:

"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.

The last Appendix B criterion cited in the bulletin is:

"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. For cracking of VHP nozzles, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future cracking. These actions could include proactive inspections and repair of degraded VHP nozzles."

The identification and confirmation of a leaking VHP nozzle 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 will be adequately addressed.

In summary, the TMI Unit 1 and industry approach to inspection, monitoring, cause determination, and resolution of the identified VHP nozzle cracking is consistent with the performance-based objectives of Appendix B.

4. *If the susceptibility ranking of your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information;*

   a. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

   Response:

   This item is not applicable to TMI Unit 1

   b. *your basis for concluding that the inspections identified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

      (1) *If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

      Response:

      This item is not applicable to TMI Unit 1

(2) The corrective actions that will be taken , including alternative inspection methods (for example, volumetric examination), if leakage is detected.

Response:

This item is not applicable to TMI Unit 1

5. *Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage.*

   a. *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

   b. *if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.*

   Response:

   AmerGen will provide the information requested by item 5 of NRC Bulletin 2001-01, or indicate that no leakage was identified, within 30 days after restart following the next refueling outage at TMI Unit 1, which is currently scheduled to begin on October 9, 2001.

References

1. *PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48)*, EPRI, Palo Alto, CA: 2001. 1006284.

2. AmerGen Letter, J. A. Hutton to NRC, *ASME Section XI Relief Requests Associated with Reactor Vessel Head Repair*, dated July 2, 2001 (5928-01-20188).

3. AmerGen Letter, J. A. Benjamin to NRC, *Reactor Vessel Head Inspection and Repair*, dated August 1, 2001 (5928-01-20217).

4. AmerGen Letter, M. P. Gallagher to NRC, *ASME Section XI Relief Requests Associated with Reactor Vessel Head Repair*, dated August 9, 2001 (5928-01-20214).

5. *CRDM Nozzle/Bore Dimensional Analysis*, Framatome ANP Engineering Information Record, Document Identifier 51-5013435-02, July 24, 2001.

6. GPU Nuclear Letter, H. D. Hukill to NRC, *GPUN Response to Generic Letter (GL) 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*, dated June 8, 1988 (C311-88-2075).

7. Response to NRC Review Comments Transmitted by letter dated June 22, 2001 to the Nuclear Energy Institute related to PWR Materials Reliability Program Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44) Part 2; Reactor Vessel Top Head Penetrations EPRI TP-1001491, Part 2, May 2001 (MRP 2001-050), dated June 29, 2001.

ATTACHMENT 2

Response to NRC Bulletin 2001-01
Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

Braidwood Station, Units 1 and 2

Exelon Generation Company, LLC

Attachment 2

Response to NRC Bulletin 2001-01

Braidwood Station Units 1 and 2

NRC requested information

1. *All addressees are requested to provide the following information:*

   a. *the plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;*

   Response:

   Braidwood Units 1 and 2 have been ranked for the potential for primary water stress corrosion cracking (PWSCC) of the reactor pressure vessel (RPV) top head nozzles using the time-at-temperature model and plant-specific input data reported in MRP-48 (i.e., Reference 1). As shown in Table 2-1 of MRP-48, this evaluation indicates that it will take Braidwood Units 1 and 2 129.5 and 154.8 effective full power years (EFPY), respectively, of additional operation from March 1, 2001, to reach the same time at temperature that Oconee Nuclear Station Unit 3 (ONS3) had at the time that its leaking nozzles were discovered in February 2001.

   Therefore, Braidwood Units 1 and 2 are in the NRC category of plants which can be considered as having low susceptibility based on a susceptibility ranking of more than 30 EFPY from the ONS3 condition.

   b. *a description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles;*

   Response:

   Braidwood Units 1 and 2 each have 79 total RPV head penetration nozzles. Figure 1 shows the Braidwood Unit 1 layout. Braidwood Unit 2 is a mirror image of the Unit 1 layout. The requested nozzle information is provided in Table 2-3 of MRP-48.

   The Braidwood configuration has 53 CRDM nozzles, 18 spare CRDM nozzles, 5 in-core thermocouple nozzles, 2 reactor vessel level indication system (RVLIS) nozzles, all equally sized, plus 1 smaller reactor head vent nozzle. The head vent has an outside diameter of 1.315 inches and an inside diameter of 0.815 inches (1 inch schedule 160 piping). The head vent is a J-groove design as shown in Figure A-12 of the MRP-44, Part 2, report. The other 78 nozzles have approximately 4 inch outside diameter housings with an inside diameter of 2.75 inches. All the vessel head penetrations are SB-167 material. The J-groove weld was fabricated with Alloy 182 weld material. The center to center distance between most penetrations is approximately 12 inches. Accounting for the diameters of the penetrations, the minimum outer surface to outer surface distance between the VHP is about 8 inches.

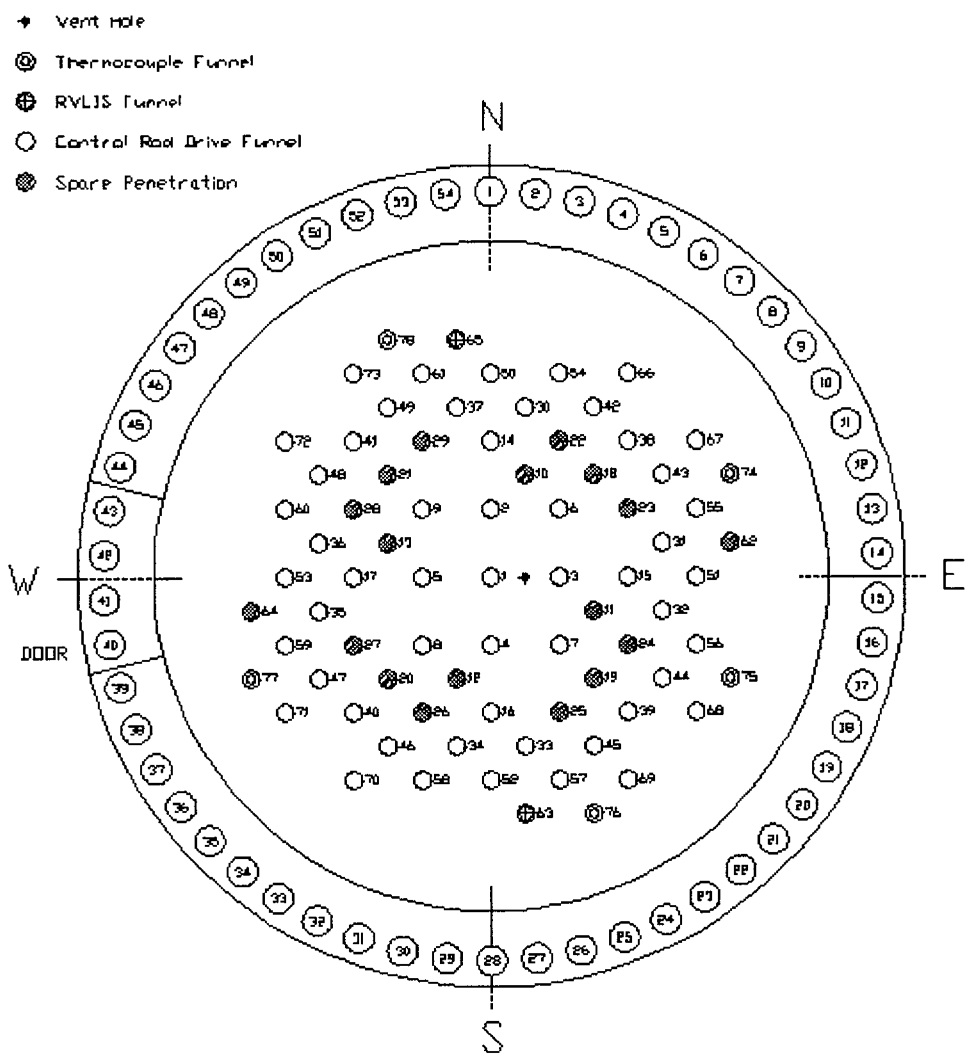

Figure 1
Braidwood Unit 1 VHP Layout

c. a description of the RPV head insulation type and configuration;

Response:

As reported in Table 2-1 of MRP-48, Braidwood Units 1 and 2 have reflective horizontal RPV head insulation.

Braidwood Units 1 and 2 RPV heads have 3 inch mirror insulation installed with overlapping joints in an interwoven pattern. The insulation is installed in a flat field across the top of the RPV head and is stepped down as it approaches the outer perimeter of the RPV head. No inspection ports or other access exists at this time to view the bare metal of the RPV head.

The insulation is designed for removal, however, with the configuration and close quarters on the RPV head, the removal would require significant dose.

The minimum clearance between the vessel surface and the bottom of the insulation is 1.5 inches at the top of the dome (see Figure 2).

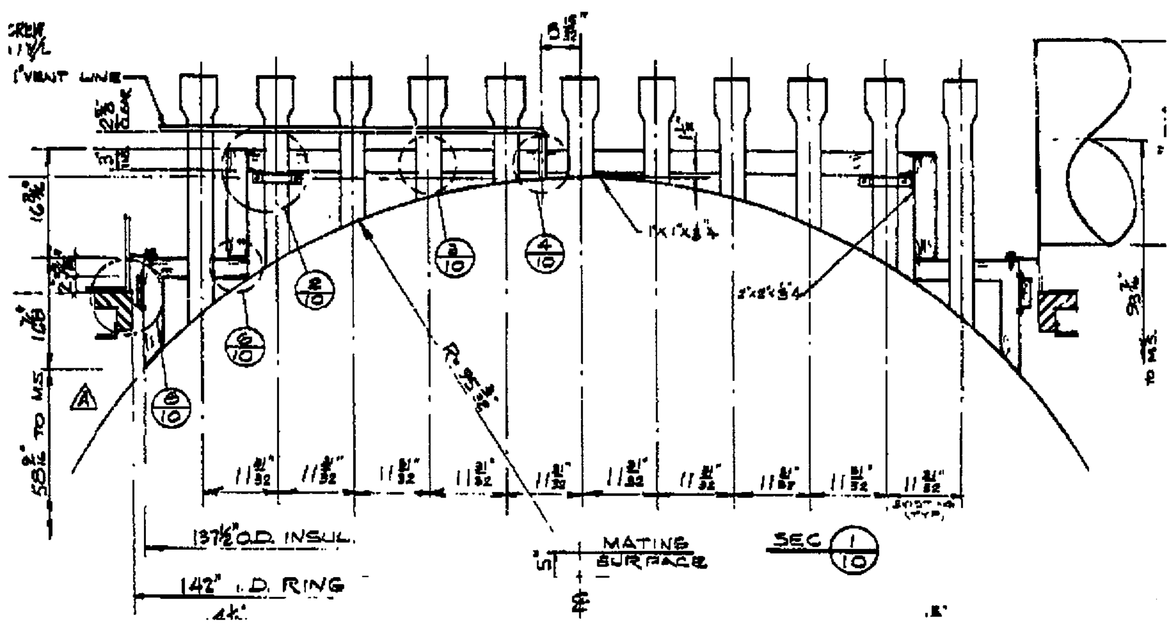

Figure 2
Reactor Vessel Dome Insulation Layout

d. *a description of the VHP nozzle head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;*

Response:

Braidwood performs a visual examination of the CRDM housings and VHP housing areas above the vessel head insulation each refueling outage. This examination is performed at both shutdown and startup. The examination performed at shutdown in response to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants", as documented in Reference 2. The examination performed prior to startup is required by ASME Section XI, Table IWB-2500-1.

Both of these examinations are performed by certified ASME Section XI VT-2 examiners and are intended to identify any evidence of leakage including boric acid deposits. The examinations are performed by direct VT-2 method through the access doors in the cooling shroud assembly. The current mirror insulation configuration (insulation is not normally removed) does not allow for a bare metal examination of the head. The insulation is designed for removal, however, with the configuration and close quarters on the RPV head, the removal would require significant dose.

Also, each ASME Section XI inspection period, a qualified remote VT-3 visual examination is performed on the underside of the reactor vessel head looking at the interior surfaces including the cladding and the areas around the VHPs. These examinations are required per ASME Section XI, Category B-N-1. There have not been any bare metal exams performed under the reactor vessel head insulation.

All individuals performing the VT-2 examinations are certified to corporate procedure SPPM 2-1-0, which meets the requirements of the ASME Section XI 1989 and 1992 Editions. In accordance with the current acceptance criteria for boric acid deposits, any amount detectable is considered a recordable indication and must be dispositioned.

There have been no recordable indications identified in the Generic Letter 88-05 examinations conducted on the reactor vessel head. Section XI Category B-N-1 examinations performed under the vessel head have resulted in no recordable indications.

Braidwood VHP Nozzle/RPV Head Inspections

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
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.	No Recordable Indications (NRI)
1	4/97	VT-3 of the interior surfaces of the Reactor Vessel. Note 2.	No cracks, linear indications. No erosion, corrosion, wear, or blocked CRD motion.	NRI
1	5/97	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	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-3 of the interior surfaces of Reactor Vessel Head. Note 2.	No cracks, linear indications. No erosion, corrosion, wear, or blocked CRD motion.	NRI
1	11/98	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	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	4/00	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
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	VT-3 of the interior surfaces of Reactor Vessel Head. Note 2.	No cracks, linear indications. No erosion, corrosion, wear, or blocked CRD motion.	NRI
2	11/97	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	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-3 of the interior surfaces of Reactor Vessel Head. Note 2.	No cracks, linear indications. No erosion, corrosion, wear, or blocked CRD motion.	NRI
2	5/99	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	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
2	11/00	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

Notes:
(1) Exam performed per commitments in response to Generic Letter 88-05 and ASME Section XI, IWB 2500-1, Category B-P, Item B15.10 during shutdown and startup.
(2) Exam performed using underhead robotic inspection camera per the requirements of ASME Category B-N-1, Item B13.10.

e. *a description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.*

Response:

Mounted directly on the reactor vessel head, the Integrated Reactor Vessel Head Assembly (see Figure 3) combines the head lifting rig, seismic platform, lift columns, reactor vessel missile shield, control rod drive mechanism (CRDM), forced air cooling system, and electrical and instrumentation cable routing into one system.

Vessel Missile Shield

The reactor vessel missile shield is a 4 inch thick flat circular steel plate mounted above the head cooling assembly. The bottom of the missile shield is located 338 inches up from the mating surface of the reactor vessel head. The shield prevents any missiles that may be ejected from atop the head from penetrating other reactor coolant system pressure boundaries and/or the containment structure. The missile shield also provides seismic support for the CRDMs. Extensions on the CRDM rod travel housings protrude through holes in the missile shield plate, limiting the lateral displacement of the housings during a seismic occurrence. The missile shield is in turn attached to 3 seismic tie rods, which are attached to clevises on the cavity walls. In addition, the missile shield serves as a spreader for the head lift rig, transferring the reactor vessel head load to the lift rig during the lift of the head. Supported by the three lift rods that extend down to the head, the missile shield may be leveled as well as removed to provide access to the CRDMs. The lift rods, missile shield, and seismic tie rods comprise the CRDM seismic support system. They are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF.

CRDM Housings

The full-length Control Rod Drive Mechanism (CRDM) consists of five basic assemblies:

1. Drive Rod Assembly
2. Latch Assembly
3. Pressure Vessel Assembly
4. Seismic Sleeve Assembly
5. Operating Coil Stack Assembly

The pressure vessel assembly and the seismic sleeve assembly were designed to meet the design requirements of the 1974 Edition, Summer 1974 Addenda, ASME Code Section III. The pressure vessel assembly is classified as safety Class 1. The seismic sleeve is designed as a safety Class 1 component support and provides seismic support for the CRDM pressure boundary.

The pressure vessel assembly encloses the drive rod and latch assemblies to provide a pressure barrier for the primary reactor coolant water in the CRDM. The latch housing portion of the pressure vessel assembly has an internal thread at the bottom which mates with the external thread on the reactor head adapter. This connection between the CRDM pressure vessel assembly and the reactor head adapter is located approximately 230 inches down from the missile shield. The vessel head penetration housing to CRDM head adapter weld is located approximately 241.5 inches down from the missile shield.

The seismic sleeve assembly is supported by the top cap portion of the drive rod assembly via a close tolerance sliding fit. A reduced diameter at the top of the sleeve protrudes through a hole in the missile shield to provide the sleeve-to-missile shield interface. In this way CRDM inertia loads that may result from a seismic event are transferred from the CRDM to the sleeve which reacts the loading onto the seismically supported missile shield. The seismic sleeve to CRDM top cap interface is located approximately 22 inches down from the missile shield.

Cabling

The power and instrumentation cables are located approximately 13 inches down from the missile shield. The cables extend from the CRDM connections, across the messenger tray assembly, around the shroud and to the connector plate on the cooling shroud. These consist of power cables for the CRDMs and cooling fans, an auxiliary power cable for the hoist assemblies, and instrumentation cables for the Digital Rod Position Indicators.

Cooling Shroud

The cooling shroud structure provides support for the CRDM cooling system fans and the stud tensioner hoists. Cooling air is directed through openings in the shroud, down along the mechanisms, back up the shroud through the CRDM cooling fans, and then upward into the containment atmosphere. Four fans are provided on the shroud to deliver the required flow. The shroud structure is bolted to a support ring on the reactor vessel head and is also attached to the three lift columns.

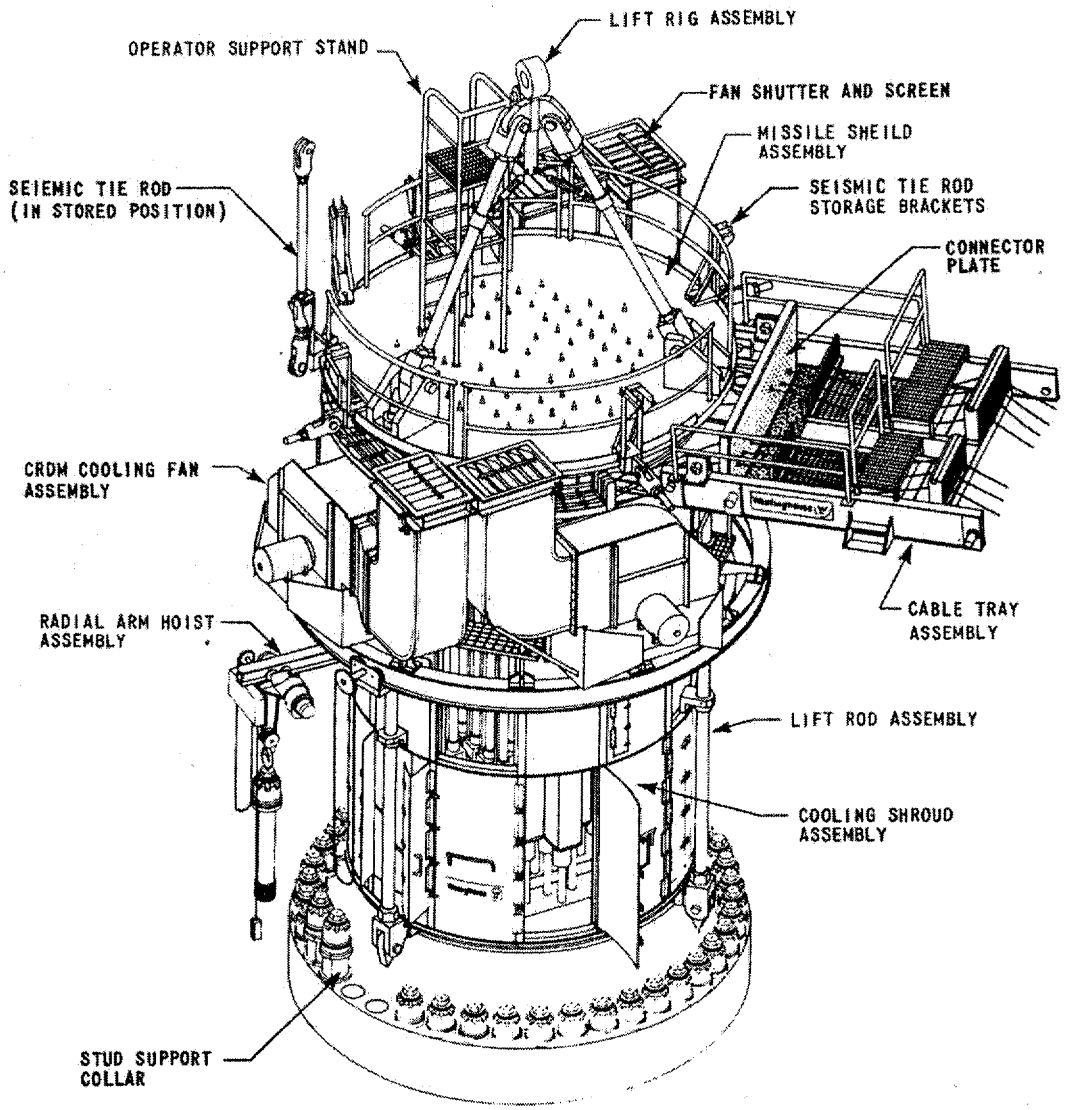

Figure 3
Integrated Reactor Vessel Head Assembly

2. *If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide the following information:*

   a. *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

   Response:

   This item is not applicable to Braidwood Station Units 1 or 2.

   b. *a description of the additional or supplemental inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken in response to identified cracking to satisfy applicable regulatory requirements;*

   Response:

   This item is not applicable to Braidwood Station Units 1 or 2.

   c. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

   Response:

   This item is not applicable to Braidwood Station Units 1 or 2.

   d. *your basis for concluding that the inspections identified in 2.c will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

      (1) *If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

   Response:

   This item is not applicable to Braidwood Station Units 1 or 2.

      (2) *If your future inspection plans do not include volumetric examination of all VHP nozzles, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will be satisfied.*

   Response:

   This item is not applicable to Braidwood Station Units 1 or 2.

3. *If the susceptibility ranking of your plant is within 5 EFPY of ONS3, addressees are requested to provide the following information:*

   a. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

Response:

This item is not applicable to Braidwood Station Units 1 or 2.

b. *your basis for concluding that the inspections identified in 3.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

   (1) *If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

   Response:

   This item is not applicable to Braidwood Station Units 1 or 2.

   (2) *If your future inspection plans include only visual inspections, discuss the corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.*

   Response:

   This item is not applicable to Braidwood Station Units 1 or 2.

4. *If the susceptibility ranking of your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information;*

   a. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

   Response:

   This item is not applicable to Braidwood Station Units 1 or 2.

   b. *your basis for concluding that the inspections identified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

      (1) *If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

      Response:

      This item is not applicable to Braidwood Station Units 1 or 2.

      (2) *The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.*

Response:

This item is not applicable to Braidwood Station Units 1 or 2.

5. *Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage:*

   a. *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack identified;*

   b. *if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.*

Response:

Exelon will provide the information requested by item 5 of NRC Bulletin 2001-01 or indicate that no leakage was identified within 30 days after plant restart following the next refueling outages at Braidwood Units 1 and 2, which are currently scheduled for September 2001 and April 2002 respectively.

While Braidwood Units 1 and 2 do not plan on doing any specific inspections of the RPV head nozzles during the next refueling outage, both units will perform a general visual inspection of the head in accordance with its Generic Letter 88-05 commitments.

References:

1. PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48), TP-1006284, EPRI, Palo Alto, CA.

2. Letter from W. E. Morgan (Commonwealth Edison Company) to A. Bert Davis (NRC), "Zion Station Units 1 and 2, Byron Station Units 1 and 2, Braidwood Station Units 1 and 2, Response to NRC Generic Letter 88-05", dated May 31, 1988.

ATTACHMENT 3

Response to NRC Bulletin 2001-01
Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

Byron Station, Units 1 and 2

Exelon Generation Company, LLC

Attachment 3

Response to NRC Bulletin 2001-01

Byron Station Units 1 and 2

NRC requested information

1. *All addressees are requested to provide the following information:*

   a. *the plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;*

   Response:

   Byron Units 1 and 2 have been ranked for the potential for primary water stress corrosion cracking (PWSCC) of the reactor pressure vessel (RPV) top head nozzles using the time-at-temperature model and plant-specific input data reported in MRP-48 (i.e., Reference 1). As shown in Table 2-1 of MRP-48, this evaluation indicates that it will take Byron Units 1 and 2 160.6 and 165.9 effective full power years (EFPY), respectively, of additional operation from March 1, 2001, to reach the same time at temperature that Oconee Nuclear Station Unit 3 (ONS3) had at the time that its leaking nozzles were discovered in February 2001.

   Therefore, Byron Units 1 and 2 are in the NRC category of plants which can be considered as having low susceptibility based on a susceptibility ranking of more than 30 EFPY from the ONS3 condition.

   b. *a description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles;*

   Response:

   Byron Units 1 and 2 each have 79 total RPV head penetration nozzles. Figure 1 shows the Byron Unit 1 layout. Byron Unit 2 is a mirror image of the Unit 1 layout. The requested nozzle information is provided in Table 2-3 of MRP-48.

   The Byron configuration has 53 CRDM nozzles, 18 spare CRDM nozzles, 5 in-core thermocouple nozzles, 2 reactor vessel level indication system (RVLIS) nozzles, all equally sized, plus 1 smaller reactor head vent nozzle. The head vent has an outside diameter of 1.315 inches and an inside diameter of 0.815 inches (1 inch schedule 160 piping). The head vent is a J-groove design as shown in Figure A-12 of the MRP-44, Part 2, report. The other 78 nozzles have approximately 4 inch outside diameter housings with an inside diameter of 2.75 inches. All the vessel head penetrations are SB-167 material. The J-groove weld was fabricated with Alloy 182 weld material. The center to center distance between most penetrations is approximately 12 inches. Accounting for the diameters of the penetrations, the minimum outer surface to outer surface distance between the VHP is about 8 inches.

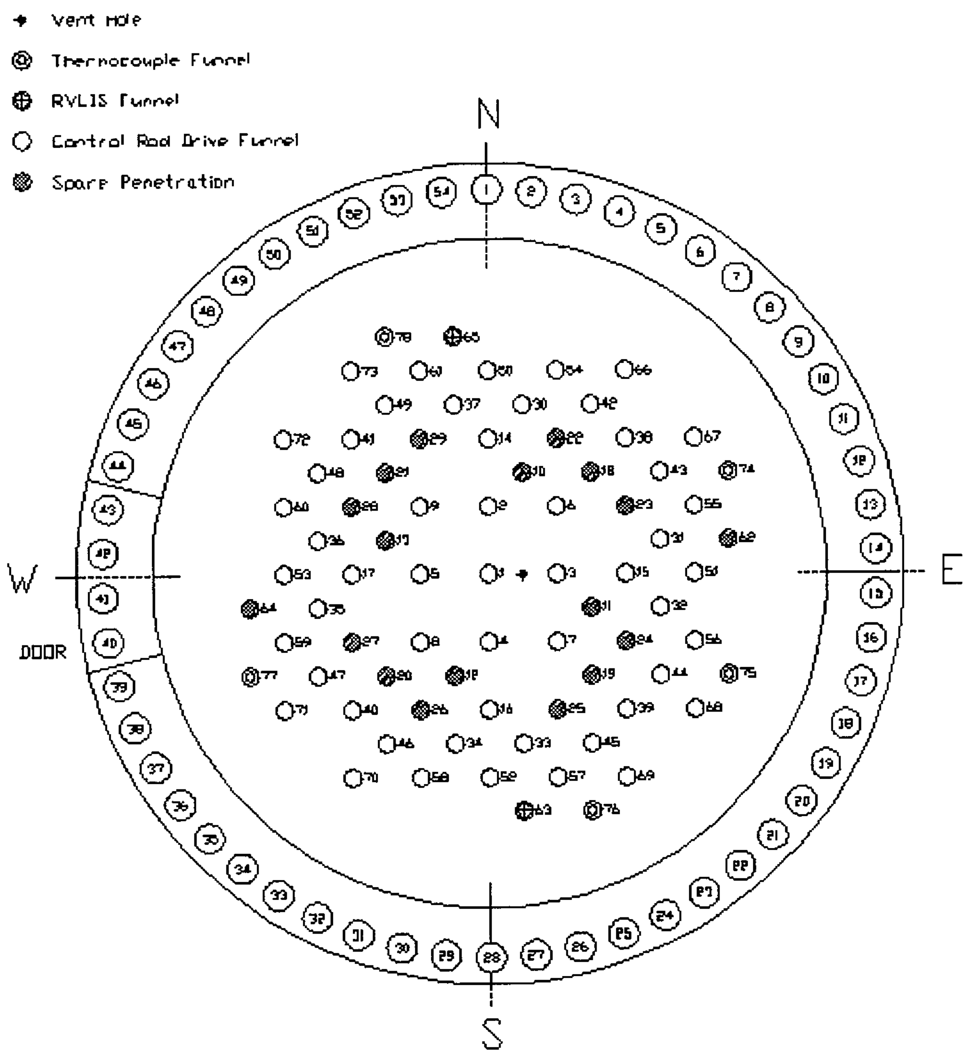

Figure 1

Byron Unit 1 VHP Layout

c. a description of the RPV head insulation type and configuration;

As reported in Table 2-1 of MRP-48, Byron Units 1 and 2 have reflective horizontal RPV head insulation.

Byron Units 1 and 2 RPV heads have 3 inch mirror insulation installed with overlapping joints in an interwoven pattern. The insulation is installed in a flat field across the top of the RPV head and is stepped down as it approaches the outer perimeter of the RPV head. No inspection ports or other access exists at this time to view the bare metal of the RPV head.

The insulation is designed for removal, however, with the configuration and close quarters on the RPV head, the removal would require significant dose.

The minimum clearance between the vessel surface and the bottom of the insulation is 0.625 inches at the top of the dome (see Figure 2).

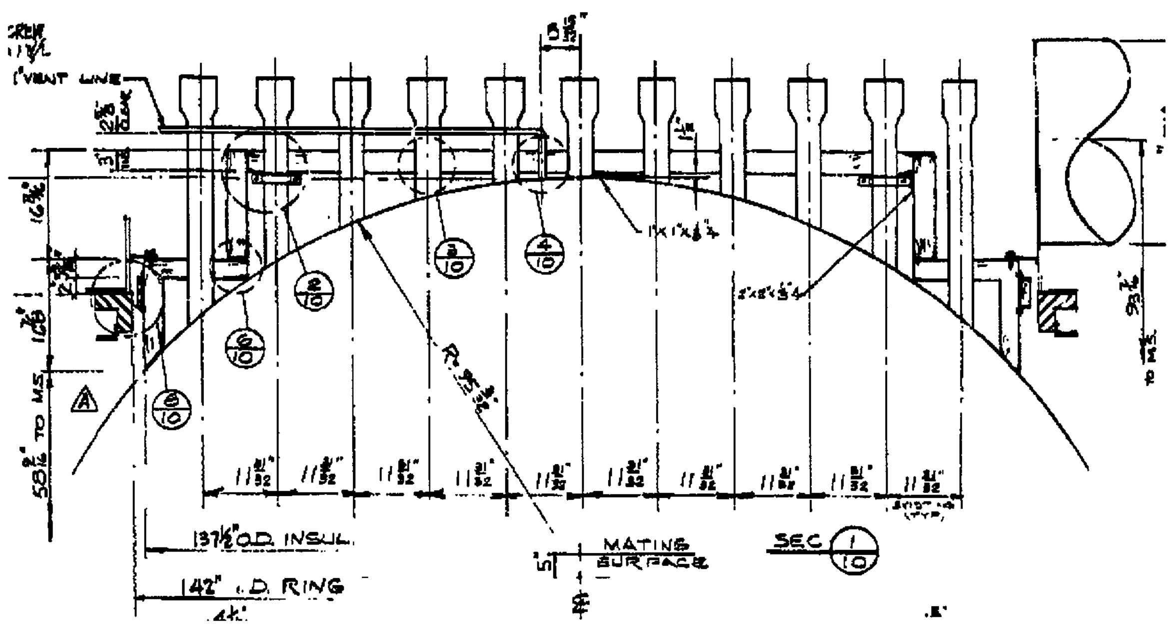

Figure 2
Reactor Vessel Dome Insulation Layout

d. a description of the VHP nozzle head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;

Response:

Byron performs a visual examination of the CRDM housings and VHP housing areas above the vessel head insulation each refueling outage. This examination is performed at both shutdown and startup. The examination performed at shutdown is in response to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants", as documented in Reference 2. The examination performed prior to startup is required by ASME Section XI, Table IWB-2500-1.

Both of these examinations are performed by certified ASME Section XI VT-2 examiners and are intended to identify any evidence of leakage including boric acid deposits. The examinations are performed by direct VT-2 method through the access doors in the cooling shroud assembly. The current mirror insulation configuration (insulation is not normally removed) does not allow for a bare metal examination of the head. The insulation is designed for removal, however, with the configuration and close quarters on the RPV head, the removal would require significant dose.

Also, each ASME Section XI inspection period, a qualified remote VT-3 visual examination is performed on the underside of the reactor vessel head looking at the interior surfaces including the cladding and the areas around the VHPs. These examinations are required per ASME Section XI, Category B-N-1. There have not been any bare metal examinations performed under the reactor vessel head insulation.

All individuals performing the VT-2 examinations are certified to corporate procedure SPPM 2-1-0, which meets the requirements of the ASME Section XI 1989 and 1992 Editions. In accordance with the current acceptance criteria for boric acid deposits, any amount detectable is considered a recordable indication and must be dispositioned.

In the last four years only one recordable indication was noted during the RPV head examination. This indication was identified during Unit 2 refueling outage B2R08 (10/99). The CRDM at core location F-14 had a pinhole leak in the middle canopy seal pressure boundary weld. This leak required the replacement of the CRDM at the F-14 location.

There have been no recordable indications identified in the Generic Letter 88-05 examinations conducted on the reactor vessel head other than the CRDM middle canopy seal pressure boundary weld at core location F-14. Section XI Category B-N-1 examinations performed under the reactor vessel head have resulted in no recordable indications.

Byron VHP Nozzle/RPV Head Inspections

Unit	Exam Date	Exam Qualification / Scope	Acceptance Criteria	Results
1	10/97	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	No Recordable Indications (NRI)
1	11/97	VT-3 of the interior surfaces of Reactor Vessel Head. Note 3.	No cracks, linear indications. No erosion, corrosion, wear, or damage to CRD guide funnels.	NRI
1	3/98	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	3/99	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	4/99	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	9/00	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
1	10/00	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	4/98	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	4/98	VT-3 of the interior surfaces of Reactor Vessel Head. Note 3.	No cracks, linear indications. No erosion, corrosion, wear, or damage to CRD guide funnels.	NRI
2	5/98	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	10/99	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	Pinhole leak CRDM Middle Canopy Weld. Note 1.
2	11/99	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	4/01	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI
2	4/01	VT-3 of the interior surfaces of Reactor Vessel Head. Note 3.	No cracks, linear indications. No erosion, corrosion, wear, or damage to CRD guide funnels.	NRI
2	4/01	VT-2 of the accessible areas on top of the head – CRDM housings. Note 2.	No evidence of leakage or boric acid residue. No degradation of material due to corrosion.	NRI

Notes:

(1) Examination of leak determined that most probable cause was Transgranular Stress Corrosion Cracking (TGSSC). CRDM Material of construction is SA-336-F8.

(2) Exam performed per commitments in response to Generic Letter 88-05 and ASME Section XI, IWB 2500-1, Category B-P, Item B15.10 during shutdown and startup.

(3) Exam performed using underhead robotic inspection camera per the requirements of ASME Category B-N-1, Item B13.10.

e. *a description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.*

Response:

Mounted directly on the reactor vessel head, the Integrated Reactor Vessel Head Assembly (see Figure 3) combines the head lifting rig, seismic platform, lift columns, reactor vessel missile shield, control rod drive mechanism (CRDM), forced air cooling system, and electrical and instrumentation cable routing into one system.

Vessel Missile Shield

The reactor vessel missile shield is a 4 inch thick flat circular steel plate mounted above the head cooling assembly. The bottom of the missile shield is located 338 inches up from the mating surface of the reactor vessel head. The shield prevents any missiles that may be ejected from atop the head from penetrating other reactor coolant system pressure boundaries and/or the containment structure. The missile shield also provides seismic support for the CRDMs. Extensions on the CRDM rod travel housings protrude through holes in the missile shield plate, limiting the lateral displacement of the housings during a seismic occurrence. The missile shield is in turn attached to 3 seismic tie rods, which are attached to clevises on the cavity walls. In addition, the missile shield serves as a spreader for the head lift rig, transferring the reactor vessel head load to the lift rig during the lift of the head. Supported by the three lift rods that extend down to the head, the missile shield may be leveled as well as removed to provide access to the CRDMs. The lift rods, missile shield, and seismic tie rods comprise the CRDM seismic support system. They are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF.

CRDM Housings

The full-length Control Rod Drive Mechanism (CRDM) consists of five basic assemblies:

1. Drive Rod Assembly
2. Latch Assembly
3. Pressure Vessel Assembly
4. Seismic Sleeve Assembly
5. Operating Coil Stack Assembly

The pressure vessel assembly and the seismic sleeve assembly were designed to meet the design requirements of the 1974 Edition, Summer 1974 Addenda, ASME Code Section III. The pressure vessel assembly is classified as safety Class 1. The seismic sleeve is designed as a safety Class 1 component support and provides seismic support for the CRDM pressure boundary.

The pressure vessel assembly encloses the drive rod and latch assemblies to provide a pressure barrier for the primary reactor coolant water in the CRDM. The latch housing portion of the pressure vessel assembly has an internal thread at the bottom which mates with the external thread on the reactor head adapter. This connection between the CRDM pressure vessel assembly and the reactor head adapter is located approximately 230 inches down from the missile shield. The vessel head penetration housing to CRDM head adapter weld is located approximately 241.5 inches down from the missile shield.

The seismic sleeve assembly is supported by the top cap portion of the drive rod assembly via a close tolerance sliding fit. A reduced diameter at the top of the sleeve protrudes through a hole in the missile shield to provide the sleeve-to-missile shield interface. In this way CRDM inertia loads that may result from a seismic event are transferred from the CRDM to the sleeve which reacts the loading onto the seismically supported missile shield. The seismic sleeve to CRDM top cap interface is located approximately 22 inches down from the missile shield.

Cabling

The power and instrumentation cables are located approximately 13 inches down from the missile shield. The cables extend from the CRDM connections, across the messenger tray assembly, around the shroud, and to the connector plate on the cooling shroud. These consist of power cables for the CRDMs and cooling fans, an auxiliary power cable for the hoist assemblies, and instrumentation cables for the Digital Rod Position Indicators.

Cooling Shroud

The cooling shroud structure provides support for the CRDM cooling system fans and the stud tensioner hoists. Cooling air is directed through openings in the shroud, down along the mechanisms, back up the shroud through the CRDM cooling fans, and then upward into the containment atmosphere. Four fans are provided on the shroud to deliver the required flow. The shroud structure is bolted to a support ring on the reactor vessel head and is also attached to the three lift columns.

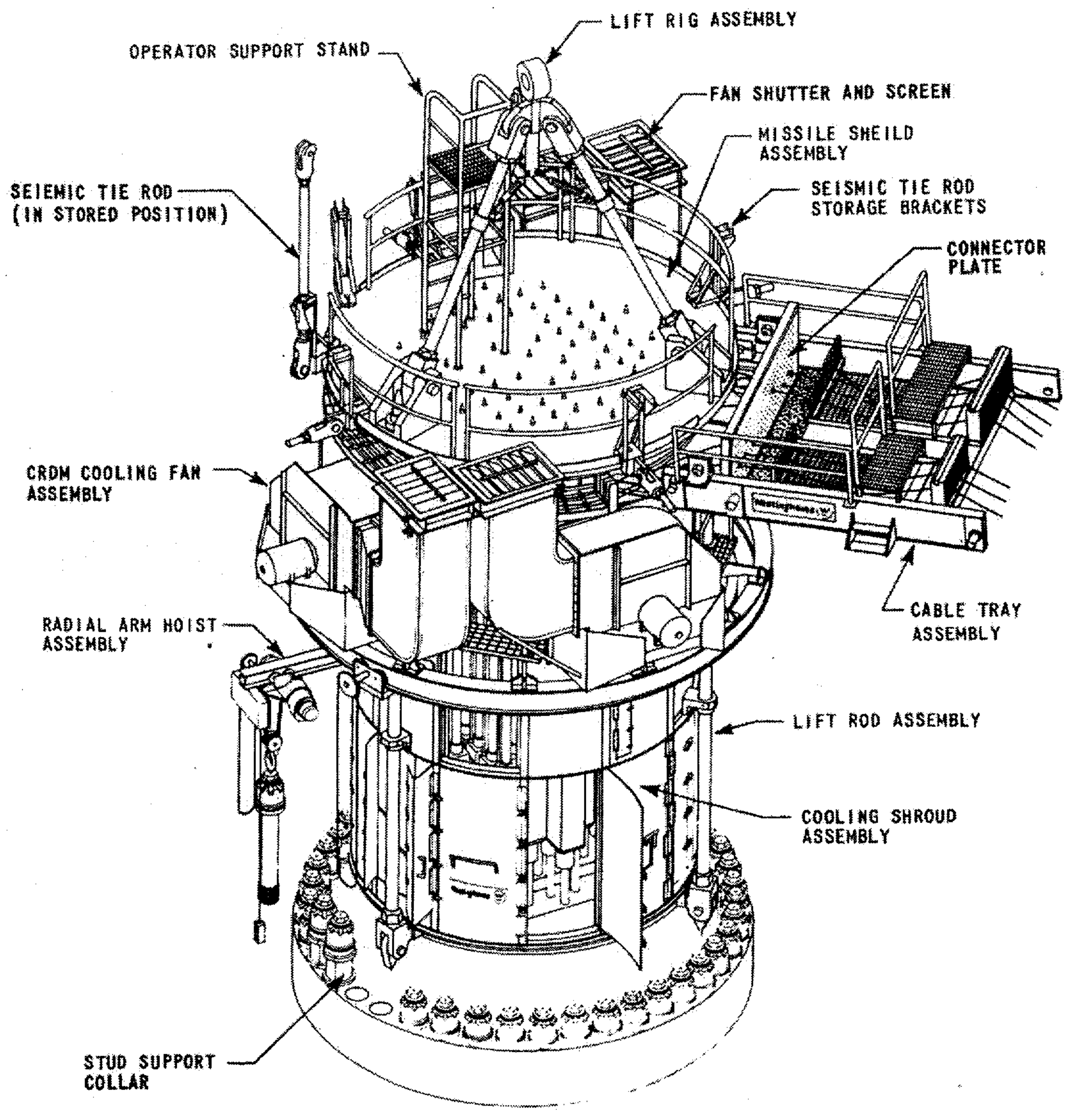

Figure 3
Integrated Reactor Vessel Head Assembly

2. *If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide the following information:*

   a. *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

   Response:

   This item is not applicable to Byron Station Units 1 or 2.

   b. *a description of the additional or supplemental inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken in response to identified cracking to satisfy applicable regulatory requirements;*

   Response:

   This item is not applicable to Byron Station Units 1 or 2.

   c. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

   Response:

   This item is not applicable to Byron Station Units 1 or 2.

   d. *your basis for concluding that the inspections identified in 2.c will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

      (1) *If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

      Response:

      This item is not applicable to Byron Station Units 1 or 2.

      (2) *If your future inspection plans do not include volumetric examination of all VHP nozzles, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will be satisfied.*

      Response:

      This item is not applicable to Byron Station Units 1 or 2.

3. *If the susceptibility ranking of your plant is within 5 EFPY of ONS3, addressees are requested to provide the following information:*

   a. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

   Response:

   This item is not applicable to Byron Station Units 1 or 2.

   b. *your basis for concluding that the inspections identified in 3.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

      (1) *If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

      Response:

      This item is not applicable to Byron Station Units 1 or 2.

      (2) *If your future inspection plans include only visual inspections, discuss the corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.*

      Response:

      This item is not applicable to Byron Station Units 1 or 2.

4. *If the susceptibility ranking of your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information;*

   a. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

   Response:

   This item is not applicable to Byron Station Units 1 or 2.

   b. *your basis for concluding that the inspections identified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

      (1) *If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

Response:

This item is not applicable to Byron Station Units 1 or 2.

(2) *The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.*

Response:

This item is not applicable to Byron Station Units 1 or 2.

5. *Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage:*

   a. *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack identified;*

   b. *if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.*

Response:

Exelon will provide the information requested by item 5 of NRC Bulletin 2001-01 or indicate that no leakage was identified within 30 days after plant restart following the next refueling outages at Byron Units 1 and 2, which are currently scheduled for March 2002 and September 2002 respectively.

While Byron Units 1 and 2 do not plan on doing any specific inspections of the RPV head nozzles during the next refueling outage, both units will perform a general visual inspection of the head in accordance with its Generic Letter 88-05 commitments.

References:

1. PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48), TP-1006284, EPRI, Palo Alto, CA.

2. Letter from W. E. Morgan (Commonwealth Edison Company) to A. Bert Davis (NRC), "Zion Station Units 1 and 2, Byron Station Units 1 and 2, Braidwood Station Units 1 and 2, Response to NRC Generic Letter 88-05", dated May 31, 1988.

ATTACHMENT 4

SUMMARY OF REGULATORY COMMITMENTS

The following table summarizes those regulatory commitments established in this document. Any other actions discussed in the submittal represent intended or planned actions by AmerGen/Exelon. They are described to the NRC for the NRC's information and are not regulatory commitments.

COMMITMENT	**COMMITTED DATE OR "OUTAGE"**
1. Each refueling outage TMI Unit 1 will perform a qualified bare metal visual VT-3 inspection of all VHP nozzles through the access openings around the perimeter of the service structure using site specific procedures and certified ASME Level III inspectors trained in accordance with the EPRI Visual Training Package and specifically trained on VHP nozzle leakage experience from Oconee and ANO.	Fall 2001 (1R14)
2. The TMI Unit 1 reactor head will be cleaned to remove existing deposits and videotaped prior to unit restart following the 1R14 refueling outage.	Fall 2001 (1R14)
3. For any VHP nozzle that is identified and suspected as leaking, a volumetric examination (i.e., best available technology) will be performed for flaw confirmation and characterization. If the characterizations indicate circumferential cracking above the J-groove weld, TMI Unit 1 will perform additional volumetric inspections of other readily available CRDMs (the CRDMs that are removed from the reactor vessel head to facilitate affected nozzle repair). If there are no indications of leakage attributed to CRDM nozzles, no further CRDM nozzle examinations will be conducted.	Fall 2001 (1R14)
4. TMI Unit 1 will repair any leaking VHPs and VHPs that have unacceptable flaws.	Fall 2001 (1R14)
5. TMI Unit 1will provide a report of the results of the visual inspections performed during the Fall 2001 refueling outage, and any corrective actions taken within 30 days following restart of the unit after the next refueling outage.	Fall 2001 (1R14)

COMMITMENT	COMMITTED DATE OR "OUTAGE"
6. Braidwood Unit 1 will provide a report of the results of the visual inspections performed during the next refueling outage, and any corrective actions taken within 30 days following restart of the unit after the next refueling outage.	Fall 2001 (A1R09)
7. Braidwood Unit 2 will provide a report of the results of the visual inspections performed during the next refueling outage, and any corrective actions taken within 30 days following restart of the unit after the next refueling outage.	Spring 2002 (A2R09)
8. Byron Unit 1 will provide a report of the results of the visual inspections performed during the next refueling outage, and any corrective actions taken within 30 days following restart of the unit after the next refueling outage.	Spring 2002 (B1R11)
9. Byron Unit 2 will provide a report of the results of the visual inspections performed during the next refueling outage, and any corrective actions taken within 30 days following restart of the unit after the next refueling outage.	Fall 2002 (B2R10)