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January 21, 2003

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

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

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

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

References:
(1) Letter from J. A. Benjamin (Exelon Generation Company, LLC) to NRC, dated May 17, 2002

(2) Letter from Mahesh Chawla (NRC) to John L. Skolds (Exelon Generation Company, LLC), dated November 20, 2002

(3) Letter from Timothy G. Colburn (NRC) to John L. Skolds (AmerGen Energy Company, LLC), dated November 21, 2002

On March 18, 2002, the NRC issued Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The Bulletin included a request for the basis for concluding licensees' boric acid inspection programs meet the requirements of Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR plants." Reference 1 provided the requested information for Braidwood and Byron Stations and Three Mile Island Station, Unit 1. The NRC concluded that additional information is necessary to complete their review and requested additional information, relative to the specificity of the boric acid corrosion control programs, be provided within sixty days of receipt of the letters (i.e., References 2 and 3). The requested additional information is attached and as suggested in Reference 1, each of the attributes identified for an improved boric acid corrosion control program were considered in the preparation of the attached responses.

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Attachments 1 and 2 to this letter provide the Exelon Generation Company, LLC response for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. Attachment 3 provides the AmerGen Energy Company, LLC response for Three Mile Island Station, Unit 1. These responses are due to the NRC January 21, 2003.

The following table identifies commitments made in this document. Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.

COMMITMENT	COMMITTED DATE OR "OUTAGE"
Braidwood Station will comply with the recent Material Reliability Project (MRP) recommendations as stated in the letter from Leslie Hartz (Chair, MRP) to the Electric Power Research Institute (EPRI) PWR Materials Management Program (PMMP) Steering committee, dated December 2, 2002	Not Applicable
Byron Station will comply with the recent MRP recommendations as stated in the letter from Leslie Hartz (Chair, MRP) to the EPRI PMMP Steering committee, dated December 2, 2002	Not Applicable
Three Mile Island Station, Unit 1 is planning to install a camera above the insulation to visually inspect the bare metal around the incore instrumentation nozzle penetrations.	During the refueling outage currently scheduled for October 2003 (T1R15).

If you have any questions or desire additional information regarding this letter, please contact Don Cecchett at (630) 657-2826.

Respectfully,

Jeffrey A. Benjamin
Vice President,
Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Respectfully,

Jeffrey A. Benjamin
Vice President,
Licensing and Regulatory Affairs
AmerGen Energy Company, LLC

Attachments: Attachment 1, Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

Attachment 2, Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Attachment 3, Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Three Mile Island Station, Unit 1

ATTACHMENT 1

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

Exelon Generation Company, LLC

Attachment 1

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

On November 20, 2002, the NRC issued a request for additional information (RAI) for NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The below information was requested within 60 days receipt of the RAI.

1. Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel (RPV) bottom head).

Response

The following is a listing of the requested information regarding the Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and locations in the Braidwood Station, Unit 1 and Unit 2 reactor coolant pressure boundary (RCPB).

Some of the Alloy 600 and Alloy 82/182 materials are not listed in this table because they are internal to components and, assuming their failure, do not have a potential to degrade the RCPB with boric acid leakage. These items include, in the steam generators, the steam generator tubing, the tube sheet cladding, the primary head divider plate and attachment welds, the primary nozzle closure rings and closure ring welds. For the reactor pressure vessels (RPVs), the core support guide lugs and welds, six per RPV, are not included in the list. The RPV flange o-ring leakage monitoring tube is not included on the list because it is not exposed to the temperature and pressure conditions that facilitate primary water stress corrosion cracking (PWSCC).

Also not listed are the Alloy 690 materials and Alloy 52/152 weld materials in the Braidwood Station, Unit 1 replacement steam generators. Installed in the Fall of 1998, the Unit 1 steam generators do not contain any Alloy 600 or Alloy 82/182 materials.

The American Society of Mechanical Engineers ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," serves as the technical basis for the inspections, techniques, scope and extent of coverage, inspection frequency, personnel qualifications, and extent of insulation removal for those components listed in the following table. Braidwood Station, currently in the 2nd Inservice Inspection Interval, is committed to the 1989 Edition, no addenda, of ASME Section XI. In addition, the visual examinations of the RCPB and associated systems, structures, and components are supplemented by the requirements of Braidwood Station's commitment to Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in pressurized water reactor (PWR) Plants."

The ASME Section XI and GL 88-05 requirements for visual examinations conducted at Braidwood Station to locate evidence of leakage from pressure retaining components are contained in procedure ER-AA-335-015, "VT-2 Visual Examination."

In addressing the extent of coverage, ER-AA-335-015 states:

> *"4.4.4. **PERFORM** VT-2 examination as close to the component as possible, even though the examination is essentially hands-off.*
>
> *1. **USE** existing ladder(s), scaffolding, etc. to reduce the examination distance or to maximize examined surface areas if permitted by plant conditions (such as safety or health physics). **INSTALLATION** of ladder(s) or scaffolding for the sole purpose of conducting the VT-2 examinations is **not** required.*
>
> *2. If permitted by plant conditions (such as safety or health physics), **then CONSIDER** the use of remote optical aids to reduce the examination distance or to maximize examined surface areas."*

This procedure does not specify examination details for specific components in the RCPB and associated systems, but rather provides general requirements applicable to all examined areas.

ER-AA-335-015 requires that accessible external surfaces of pressure retaining components be examined for evidence of leakage including evidence of boric acid accumulations from borated systems. If the components are inaccessible for the direct VT-2 examination, the procedure requires that the surrounding area, including floor area or equipment surfaces located underneath the components, be examined.

The procedure does not require that insulated components be de-insulated to perform a VT-2 examination. The procedure states that for insulated components, accessible and exposed insulation surfaces, including each insulation joint, be examined. For essentially vertical surfaces, the insulation at the lowest elevation where leakage may be detectable must be examined. Also, the surrounding area, including floor areas or equipment surfaces located underneath the component, is examined for evidence of leakage. The procedure requires that examiners give particular attention to discoloration of residue on surfaces in order to detect evidence of boric acid accumulations from borated reactor coolant leakage.

While the removal of insulation is not necessarily required for the performance of a VT-2 exam, if evidence of leakage or boric acid residue is detected, the procedure requires that the leakage source be located which may require insulation removal.

In summary, the components listed in the following table are examined using ASME Section XI and ER-AA-335-015 requirements. These requirements address the technical basis for the scope, inspection technique, inspection frequency, personnel qualifications, and, in general, extent of coverage.

Attachment 1

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

Braidwood Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
1	1	RPV nozzle to safe-end welds (4 hot leg safe-ends, 4 cold leg safe-ends, 8 total)	Visual Inspection for Leakage (VT-2)	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe insulation joints and along the annulus of the piping penetration.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is a stainless steel reflective (mirror) design. There are 3, 120 ° segments buckled around the pipe-nozzle OD.	No corrective actions required to date.
			Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Exams performed by Certified Level II or III ultrasonic Examiners.	100% of the risk informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Spring of 1997.	Insulation is not removed for this exam. The exam is performed from the pipe/nozzle inner diameter with an automated inspection tool.	

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

Braidwood Station RCPB Alloy Material Listing

Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
2	1	RPV head penetrations 79 total - 53 Control Rod Drive Mechanisms (CRDMs) - 2 Reactor Vessel Level Indication System (RVLIS) - 5 thermocouple - 18 spare VHPs - 1 head vent	Previous VT-2 exams performed with the RPV head insulation in-place. Bare metal visual (BMV) inspection using specific inspection guidelines have been developed and will be implemented.	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	VT-2 examinations have been performed on the accessible areas of the head. These exams are conducted on the RPV head with the shroud assembly access doors opened and the RPV head insulation in-place.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time. A BMV exam is scheduled for the Spring of 2003. The extent and frequency of additional examinations to be in accordance with MRP.	Insulation is removed as necessary to facilitate the BMV exam. Peripheral vertical and horizontal panels are designed for removal. The RPV head insulation is a series of 3 thick mirror insulation panels. 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 corrective actions required to date.

Attachment 1

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

Braidwood Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
3	1	RPV lower head in-core instrumentation penetrations (58 total)	Previous VT-2 exams performed with the RPV lower head insulation in-place.	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	The examination has been performed with RPV lower head insulation in-place looking for leakage at insulation joints.	The examination is performed each refuel outage or during a forced outage of sufficient duration. The examination is performed at operating pressure and temperature, with a 4 hour hold time, beneath the RPV.	For previous exams, the insulation has not been removed. The insulation at the bottom of the RPV is a flat, horizontal deck of stainless steel mirror panels. The deck stands off from the bottom of the lower head providing a clearance of 8". The center panels are fixed around the 58 in-core guide tubes and are not designed for removal. The peripheral panels are removable and allow access to the lower RPV head surface.	No corrective actions required to date.

Attachment 1

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

Braidwood Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
4	1	Pressurizer surge nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end weld.	Exams performed by certified Level II or III ultrasonic Examiners.	100% of the risk informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI Interval. Last examined in the Spring of 1997.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer lower head insulation arrangement. There are 4 segments enclosing the surge nozzle. The panels are stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe insulation joints, along the surge line piping and on the containment floor beneath the pressurizer surge nozzle.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer lower head insulation arrangement. There are 4 segments enclosing the surge nozzle. The panels are stainless steel, mirror panels with buckled snaps for removal.	

Braidwood Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
5	1	Pressurizer spray nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end weld.	Exams performed by certified Level II or III ultrasonic Examiners.	100% of the risk informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Fall of 1998.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 2 semi-circular, flat panels enclosing the spray nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exam performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the nozzle to pipe insulation joint.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 2 semi-circular, flat panels enclosing the spray nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Braidwood Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
6	1	Pressurizer relief valve nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end weld.	Exams performed by certified Level II or III ultrasonic Examiners.	100% of the risk informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI Interval. Last examined in the Fall of 1998.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing the relief nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exam performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joint.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing the relief nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Attachment 1

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

Braidwood Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
7	1	Pressurizer safety valve nozzle to safe-end weld (3 welds)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Exams performed by certified Level II or III ultrasonic Examiners.	100% of the risk informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI Interval. Two of the welds were examined in the Spring of 1994, and the third weld was examined in the Fall of 1995.	Insulation has been, and will be. removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing each of the safety valve nozzles. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joints.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing each of the safety valve nozzles. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Braidwood Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
8	2	RPV nozzle to safe-end welds (4 hot leg safe-ends, 4 cold leg safe-ends, 8 total)	Visual Inspection for leakage (VT-2)	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe insulation joints and along the annulus of the piping penetration.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is a stainless steel reflective (mirror) design. There are 3, 120° segments buckled around the pipe nozzle OD.	No corrective actions required to date.
			Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Fall of 1997.	Insulation is not removed for this exam. The exam is performed from the pipe/nozzle inner diameter with an automated inspection tool.	

Attachment 1

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

Braidwood Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
9	2	RPV head penetrations (VHP) 79 total - 53 CRDMs - 2 RVLIS - 5 Thermocouple - 18 spare VHPs - 1 head vent	A Bare Metal Visual (BMV) using VT-2 techniques was performed in Spring 2002. Specific inspection guidelines were developed and implemented. Previous VT-2 exams performed with the RPV head insulation in-place.	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	BMV of the RPV head surface and a 360 degree view of all VHPs with slight obstruction of the head vent penetration.	The BMV exam was performed in the Spring of 2002. The extent and frequency of additional examinations to be in accordance with MRP.	Insulation is removed as necessary to facilitate the BMV exam. Peripheral vertical and horizontal panels are designed for removal. The RPV head insulation is a series of 3" thick mirror insulation panels. 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 corrective actions required to date. In addition, for the Spring 2002 examination, there were no corrective actions taken as there was no evidence of leakage on the RPV head surface and no RPV head degradation was identified.

Braidwood Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
10	2	RPV lower head in-core instrumentation penetrations (58 total)	Previous VT-2 exams performed with the RPV lower head insulation in-place.	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	The examination has been performed with RPV lower head insulation in-place looking for leakage at insulation joints.	The examination is performed each refuel outage or during a forced outage of sufficient duration. The examination is performed at operating pressure and temperature, with a 4 hour hold time, beneath the RPV.	For previous exams, the insulation has not been removed. The insulation at the bottom of the RPV is a flat, horizontal deck of stainless steel mirror panels. The deck stands off from the bottom of the lower head providing a clearance of 8". The center panels are fixed around the 58 in-core guide tubes and are not designed for removal. The peripheral panels are removable and allow access to the lower RPV head surface.	No corrective actions required to date.

Braidwood Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
11	2	Pressurizer surge nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Fall of 1997.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer lower head insulation arrangement. There are 4 segments enclosing the surge nozzle. The panels are stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe insulation joints, along the surge line piping and on the containment floor beneath pressurizer surge nozzle.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer lower head insulation arrangement. There are 4 segments enclosing the surge nozzle. The panels are stainless steel, mirror panels with buckled snaps for removal.	

Braidwood Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
12	2	Pressurizer spray nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Spring of 1999.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 2 semi-circular, flat panels enclosing the spray nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joint.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 2 semi-circular, flat panels enclosing the spray nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Braidwood Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
13	2	Pressurizer relief valve nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI Interval. Last examined in the Spring of 1999.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing the relief nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joint.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing the relief nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Braidwood Station, Units 1 and 2

Braidwood Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
14	2	Pressurizer safety valve nozzle to safe-end weld (3 welds)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI Interval. Two of the welds were examined in the Fall of 1994, and the third was examined in the Spring of 1996.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing each of the safety nozzles. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joints.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing each of the safety nozzles. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Braidwood Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
15	2	Steam generator primary head drain lines (4 total)	A visual using direct VT-2 techniques was performed in the Spring of 2002 on 2 of the 4 steam generator drain lines and surrounding head surface. Previous VT-2 exams performed with the steam generator lower head insulation in-place.	Exams performed by Certified Level II VT-2 Examiners.	The drain line, weld, and lower head surface around the drain line were examined for steam generators "A" and "C".	The direct visual examination is performed each time the steam generator primary channel head is de-insulated, (typically for eddy current inspection).	The insulation is removed to facilitate the examination. The insulation around the steam generator lower head is a series of removable stainless steel mirror panels. The bottom panel is a horizontal disc that is set off of the steam generator head and encloses the head, the drain line, and the drain line isolation valve.	No corrective actions required to date.

2. Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

Response

Except for bolted connections on borated ASME Section XI Class 1, 2 and 3 systems discussed below, Braidwood Station procedures do not require that insulated RCPB components be de-insulated to perform VT-2 examinations. The basis for this has been that leakage from RCPB components should be detectable at insulated joints or surrounding areas given that the systems have been at normal pressure for, in most cases, a full operating cycle. While the removal of insulation is not necessarily required for the performance of a VT-2 exam, if evidence of leakage or boric acid residue is detected, the procedure requires that the leakage source be located which may require insulation removal.

In general, Braidwood Station ASME Section XI Class 1, 2, and 3 components in the containment containing borated water, if insulated, have removable stainless steel reflective style insulation. Outside the containment, blanket insulation is used. Most insulation is removable, but there are exceptions. The insulation panels around the incore instrumentation guide tubes at the bottom of the RPV lower head are not intended to be removed. A set of peripheral horizontal panels are buckled in place and can be removed for access to the incore instrumentation guide tubes at the bottom of the RPV lower head. The horizontal panels around the VHPs are not removable; however, the side vertical panels are removable. The specific style of insulation for other locations in the RCPB is listed in the table in the response to Question 1.

Bolted Connections

Insulated bolted connections in the RCPB (i.e., ASME Section XI, Class 1) and in borated ASME Section XI Class 2 and Class 3 support systems have insulation removed in order to perform ASME section XI VT-2 examinations. The scope includes bolted connections that are installed in systems that are borated for the purpose of controlling reactivity. The ASME Section XI requirements on the extent of insulation removal and the plant conditions under which the insulation is removed have been modified by Braidwood Station Inservice Inspection Relief Requests I2R-12 and I2R-30. Both of these alternatives to ASME Section XI requirements have been authorized for use at Braidwood Station by the NRC.

Relief Request I2R-12 allows for insulation removal and the performance of the VT-2 exam on bolted connections in borated systems to be performed with the system depressurized. The approved alternative requires that a system be pressurized for a minimum of four hours at normal operating pressure prior to the VT-2 examination. Additionally, for ASME Section XI Class 2 and 3 borated systems, VT-2 examinations are performed on approximately 36 month frequencies, which coincides with plant refueling outages, not allowing the period between inspections on individual components to exceed 45 months. This frequency for individual components is more restrictive than the "Periodic Frequency" allowed by ASME Section XI for Class 2 and 3 systems described in tables IWC-2500 or IWD-2500.

Relief Request I2R-30 allows the removal of insulation from certain ASME Section XI, Class 1 valves for VT-2 examination to be performed on an extended frequency. The insulation is removed from the bolted connections and a VT-2 examination is conducted, with the system depressurized, on a once per 10 year interval frequency. These valves are also VT-2 examined with the insulation installed after a minimum four hour hold time at normal operating pressure at the end of each refueling outage and in Mode 3 during shutdown for each refueling outage.

Dissimilar Metal Welds

Insulated dissimilar metal welded piping connections that contain Alloy 82/182 are typically not de-insulated for VT-2 examinations. A list of these nozzle to safe-end welds and the type of insulation is provided in the table in the response to Question 1. Examinations of these areas are performed in response to GL 88-05 and are conducted in Mode 3 going into an outage, typically, after a cycle of operation. Therefore, an adequate time is allowed for leakage to propagate through the insulation joints and be observed by direct VT-2 examination.

The eight RPV nozzle to safe-end welds are considered inaccessible to perform direct VT-2 examination. The welds are located in an area between the concrete RPV shield wall and the concrete primary shield wall. This area enclosing the RPV nozzles and connected piping is referred to as the "sand box" area. The sand box area is only accessible from above, from the refueling cavity floor, by removing normally sealed steel plates. The normal technique for viewing potential leakage in this area is to look along the reactor coolant piping as it passes though the annulus of the bio-shield wall towards the RPV. The piping is insulated, so the examiner looks for evidence of boric acid at the insulation joints inside the annulus. The RPV nozzle to safe-end weld insulation is not routinely removed since the sand box area is considered a high dose, confined space.

3. Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

Response

For the RCPB components, Braidwood Station personnel perform walkdowns during refueling outages, forced outages and, depending on circumstances, during power operations. Typical at-power containment walkdowns of accessible areas for leakage would be initiated at the request of the Operations Department if unidentified leakage was trending up or at the request of Radiation Protection personnel if there was an unexpected increase in containment atmosphere gas or particulate levels. For at-power walkdowns, the area of interest may be limited for as low as reasonably achievable (ALARA) considerations. VT-2 certified personnel perform the walkdowns during refueling outages and, in most cases, during forced outages and at power operations.

As described in the response to Question 2, the areas considered inaccessible are the "sand boxes" which enclose the eight RPV nozzles. Leakage from these areas is identified by visual examination along the horizontal surface of the insulated piping as it passes through the concrete annulus.

Braidwood Station uses a containment floor drain sump and a reactor cavity sump to collect, measure and record unidentified leakage in accessible and inaccessible areas in the containment. Both of these sumps are instrumented to identify leakages of 1.0 gpm within one hour and are recorded and alarmed in the main control room. The containment floor drain and reactor cavity sump inputs are checked each shift. A reactor coolant system (RCS) mass balance is performed when unidentified leakage is suspected and at the prescribed Technical Specification intervals. This provides early indication to the operator of potential unidentified leakage. Also, the reactor makeup control system is used to maintain proper reactor coolant inventory, volume control tank (VCT) level is continuously recorded and quantities of boric acid and makeup water injected are totaled and flow rates recorded in the control room.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Radioactivity detection systems are used for monitoring both particulate and gaseous activities and can be used to identify RCS leakage. The detection of RCS leakage using radiation monitors depends on the concentration of radioactivity in the RCS and detector background count rate.

Air temperature and pressure monitoring methods may also be used to infer unidentified leakage to the containment. Although containment temperature and pressure fluctuate slightly during unit operation, a rise above the normally indicated range of values may indicate RCS leakage into the containment.

4. Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections), to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,

> ***a. If observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or***
>
> ***b. If observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.***

<u>Response</u>

Braidwood Station personnel use engineering procedure BwVP 200-11, "Evaluation of ASME Section XI Class 1, 2, and 3 Bolted Connections." This procedure describes the requirements and instructions for the evaluation of ASME Section XI Class 1, 2, and 3 bolted connections when external leakage is detected. The procedure provides instructions for the corrective actions that must be implemented in the event the structural integrity of any ASME Section XI Class 1, 2, or 3 bolted connection is determined to be suspect.

When evidence of leakage is identified from a bolted connection, the procedure requires an evaluation to be performed. This evaluation considers the location of the leak, leak rate, extent of deposit accumulation, extent of wastage, corrosiveness of process fluid, materials, length of time bolting has been in service and the effect on other structures or components. A condition report is initiated for non-conforming conditions and a recommendation for repair, replacement, or monitoring is provided. Active leakage from the RCS is repaired prior to completing a refuel outage.

The bases for the evaluation/determination criteria of BwVP 200-11 are derived from the provisions of Braidwood Station Inservice Inspection Relief Request I2R-13. This request was approved by the NRC.

Braidwood Station personnel use the corrective action process, specifically a condition report, to determine the acceptability for continued operation when a non-conforming condition is identified. Within the condition report, operability is considered and may require a formal evaluation using procedure LS-AA-105, "Operability Determinations." If the operability assessment determines that the non-conforming condition is acceptable, inspection/monitoring actions may be established and tracked using LS-AA-105-1001, "Supporting Operability Documentation." For a non-conforming condition that is determined to be unacceptable, repair or replacement is required.

5. Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

<u>Response</u>

As explained in the table in the response to Question 1, items 3 and 10, the insulation at the bottom of the RPV lower head stands off of the head surface by a minimum of eight inches at the very bottom of the lower head curvature. Visual examinations of the Braidwood Station, Unit 1 and Unit 2 RPV lower head, scheduled for the Spring of 2003 and Fall of 2003, respectfully, will require removing insulation panels to view the metal surface. Any leakage from lower head penetrations would be visible.

Reactor coolant and boric acid deposits from a lower penetration leak would collect on the inner surface of the flat, horizontal insulation panels. The incore penetration instrument tubes that connect to the nozzles are stainless steel and have an increased resistance to boric acid corrosion/wastage. If boric acid were to leak through the seams or openings of the insulation, there are no pressure retaining components beneath the RPV that could be affected. This leakage from the RPV bottom head nozzles would collect in the reactor cavity sump described in the response to Question 3. The reactor cavity sump input is recorded each shift and abnormal readings are required to be reported to the Shift Manager. This notification would result in an evaluation and a condition report would be initiated, as necessary.

6. Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response

The leakage detection capabilities at Braidwood Station were discussed in the response to Questions 3 and 5 above. The visual examination and evaluation procedures at Braidwood Station require that components in the area of a leak be examined and evaluated. In the case of Braidwood Station, the vast majority of components in the RCPB are stainless steel.

7. Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

Response

Vessel Head Penetrations (VHP)

Braidwood Station personnel have based the examination requirements for the Unit 1 and Unit 2 VHPs on the Electric Power Research Institute (EPRI) Material Reliability Project (MRP) susceptibility model for PWSCC. This ranking (i.e., MRP-48) identifies Braidwood Station, Unit 1 and Unit 2 as "low susceptibility" plants rated 64th and 65th, respectively, out of the 69 PWR units in the study. As of January 1, 2003, the Braidwood Station, Units have an effective degradation year (EDY) value of 1.6 putting the Units in the lowest of the three susceptibility categories established by the NRC in Bulletin 2002-02, "Reactor Pressure Vessel Head and RPV Head Penetration Nozzle Inspection Programs."

Because of this lower susceptibility, Braidwood Station personnel perform qualified, effective bare metal visual (BMV) inspections of the RPV head surface and VHPs. The BMV inspection for Braidwood Station, Unit 2 was completed in the Spring of 2002 with no evidence of VHP leakage found. The BMV inspection for Braidwood Station, Unit 1 is scheduled for the Spring of 2003.

Steam Generator Drain Lines

As listed in the table in the response to Question 1, the steam generator primary head drain lines for Braidwood Station, Unit 2 are made of Alloy 600 and Alloy 82/182 materials. The bottom surface of the steam generator primary head, the 3/8" drain lines and drain line isolation valves are enclosed in the lower head insulation package. Based on a susceptibility model developed by Westinghouse, Braidwood Station is visually examining the four Unit 2 drain lines whenever the steam generator lower head insulation package is removed to support the eddy current testing of the steam generator tubing.

In the last Unit 2 refueling outage in the Spring of 2002, the "A" steam generator was eddy current tested and therefore the lower head surface and drain line were visually examined. There were no recordable indications, no signs of boric acid deposits or any degradation of the carbon steel lower head surface.

In addition, in the GL 88-05 walkdown performed in Mode 3 during a plant shutdown in the Spring of 2002, boric acid deposits were identified along the seam of the bottom horizontal insulation panel of the "C" steam generator lower head. As required by examination procedure and the corrective action process, the insulation was removed to investigate the source and extent of the leakage. The lower head surface and drain line of the "C" steam generator were visually examined. There were no signs of boric acid deposits on the steam generator lower head or any degradation of the carbon steel surface. The leakage and boric acid deposits were from a leaking drain line isolation valve dripping onto the horizontal insulation panel directly under the drain line. The deposits were cleaned and the valve was repaired.

Based on the recommendations of the susceptibility model, the drain line and the steam generator lower head surface for all four Braidwood Station, Unit 2 steam generators are scheduled for visual examination in the Fall of 2003.

8. Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

Response

Vessel Closure Head Penetrations

As stated above in the response to Question 7, Braidwood Station personnel are following the examination recommendations of the industry effort on the Alloy 600/82/182 issue. Westinghouse, the reactor vendor, and Babcock & Wilcox, the reactor manufacturer, are part of this industry effort. Based on the lower susceptibility of the Braidwood Station, Unit 1 and Unit 2 VHP nozzles, the recommended visual examinations are being performed. Braidwood Station personnel have followed all MRP recommendations and will comply with the recent MRP recommendations as stated in the letter from Leslie Hartz (Chair, MRP) to the EPRI PWR Materials Management Program (PMMP) Steering committee, dated December 2, 2002.

Reactor Vessel Nozzle Safe-Ends, Reactor Lower Head Nozzles, and Pressurizer Nozzle Safe-Ends

The current program for the examination of these components is listed in the table in the response to Question 1. Exelon Nuclear corporate and Braidwood Station personnel are currently working with Westinghouse on developing a comprehensive inspection, repair and/or mitigation program for all Alloy 600/82/182 components in the RCPB.

9. Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.

Response

Braidwood Station Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.13.a states that there shall be no pressure boundary leakage. Pressure bypass boundary leakage is defined as leakage, except steam generator tube leakage, through a non-isolable fault in a reactor coolant system component body, pipe wall, or RPV wall. If pressure boundary leakage is detected, the action statements for this LCO require that the affected unit be in Mode 3 in six hours and be in Mode 5 in 36 hours. The resolution of leakage indications in the corrective action program requires evaluation of the impact on this TS.

Compliance with the zero non-isolable leakage criteria is met by performing GL 88-05 examinations, conducting inspections and repairs in accordance with ASME Section XI, and 10 CFR 50.55a, "Codes and standards." In addition, the unidentified leakage limit of one gpm defined in TS LCO 3.4.13.b is established as a quantity that can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a short time, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

10 CFR 50.55a requires that inservice inspection and testing be performed in accordance with the requirements of ASME Section XI. ASME Section XI contains applicable rules for examination, evaluation and repair of code class components, including the RCPB.

For this, the 2nd Inservice Inspection Interval, Braidwood Station personnel have implemented the 1989 edition, with no addenda, of ASME Section XI. Paragraph IWA-5250 (b), "Corrective Measures," of this edition states:

> "If boric acid residues are detected on components, the leakage source and the areas of general corrosion shall be located. Components with local areas of general corrosion that reduce wall thickness by more than 10% shall be evaluated to determine whether

the component may be acceptable for continued service, or whether repair or replacement is required."

To incorporate these requirements, Braidwood Station personnel use Exelon Nuclear Procedure ER-AA-335-015, "VT-2 Visual Examination." Paragraph 4.6.1.4 of this procedure states:

> *"If boric acid residues are detected on components,* ***then LOCATE*** *the leakage source and the areas of general corrosion.* ***EVALUATE*** *components with local areas of general corrosion that reduce the wall thickness by more than 10% to determine whether the component may be acceptable for continued service, or whether repair or replacement is required."*

The Exelon Nuclear procedure LS-AA-125, "Corrective Action Program (CAP) Procedure" defines the requirements for condition identification, condition review, investigation, and closeout.

ATTACHMENT 2

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Exelon Generation Company, LLC

Attachment 2

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

On November 20, 2002, the NRC issued a request for additional information (RAI) for NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The below information was required within 60 days of receipt of the RAI.

1. Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel (RPV) bottom head).

Response

The following is a listing of the requested information regarding the Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and locations in the Byron Station, Unit 1 and Unit 2 reactor coolant pressure boundary (RCPB).

Some of the Alloy 600 and Alloy 82/182 materials are not listed in this table because they are internal to components and, assuming their failure, do not have a potential to degrade the RCPB with boric acid leakage. These items include, in the steam generators, the steam generator tubing, the tube sheet cladding, the primary head divider plate and attachment welds, the primary nozzle closure rings and closure ring welds. For the reactor pressure vessels (RPVs), the core support guide lugs and welds, six per RPV, are not included in the list. The RPV flange o-ring leakage monitoring tube is not included on the list because it is not exposed to the temperature and pressure conditions that facilitate primary water stress corrosion cracking (PWSCC).

Also not listed are the Alloy 690 materials and Alloy 52/152 weld materials in the Byron Station, Unit 1 replacement steam generators. Installed in the Winter of 1997, the Unit 1 steam generators do not contain any Alloy 600 or Alloy 82/182 materials.

The American Society of Mechanical Engineers (ASME) Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," serves as the technical basis for the inspections, techniques, scope and extent of coverage, inspection frequency, personnel qualifications, and extent of insulation removal for those components listed in the following table. Byron Station, currently in the 2nd Inservice Inspection Interval, is committed to the 1989 Edition, no addenda, of ASME Section XI. In addition, the visual examinations of the RCPB and associated systems, structures, and components are supplemented by the requirements of Byron Station's commitment to Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

The ASME Section XI and GL 88-05 requirements for visual examinations conducted at Byron Station to locate evidence of leakage from pressure retaining components are contained in procedure ER-AA-335-015, "VT-2 Visual Examination."

In addressing the extent of coverage, ER-AA-335-015 states:

> *"4.4.4. **PERFORM** VT-2 examination as close to the component as possible, even though the examination is essentially hands-off.*
>
> *1. **USE** existing ladder(s), scaffolding, etc. to reduce the examination distance or to maximize examined surface areas if permitted by plant conditions (such as safety or health physics). **INSTALLATION** of ladder(s) or scaffolding for the sole purpose of conducting the VT-2 examinations is **not** required.*
>
> *2. **If** permitted by plant conditions (such as safety or health physics), **then CONSIDER** the use of remote optical aids to reduce the examination distance or to maximize examined surface areas."*

This procedure does not specify examination details for specific components in the RCPB and associated systems, but rather provides general requirements applicable to all examined areas.

ER-AA-335-015 requires that accessible external surfaces of pressure retaining components be examined for evidence of leakage including evidence of boric acid accumulations from borated systems. If the components are inaccessible for the direct VT-2 examination, the procedure requires that the surrounding area including floor area or equipment surfaces located underneath the components be examined.

The procedure does not require that insulated components be de-insulated to perform a VT-2 examination. The procedure states that for insulated components, accessible and exposed insulation surfaces, including each insulation joint, be examined. For essentially vertical surfaces, the insulation at the lowest elevation where leakage may be detectable must be examined. Also, the surrounding area including floor areas or equipment surfaces located underneath the component is examined for evidence of leakage. The procedure requires that examiners give particular attention to discoloration of residue on surfaces in order to detect evidence of boric acid accumulations from borated reactor coolant leakage.

While the removal of insulation is not necessarily required for the performance of a VT-2 exam, if evidence of leakage or boric acid residue is detected, the procedure requires that the leakage source be located which may require insulation removal.

In summary, the components listed in the following table are examined using ASME Section XI and ER-AA-335-015 requirements. These requirements address the technical basis for the scope, inspection technique, inspection frequency, personnel qualifications, and, in general, extent of coverage.

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Byron Station RCPB Alloy Material Listing

Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
1	1	RPV nozzle to safe-end welds (4 hot leg safe-ends, 4 cold leg safe-ends, 8 total)	Visual inspection for leakage (VT-2)	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe insulation joints and along the annulus of the piping penetration.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is a stainless steel reflective (mirror) design. There are 3, 120 ° segments buckled around the pipe-nozzle OD.	No corrective actions required to date.
			Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Spring of 1996.	Insulation is not removed for this exam. The exam is performed from the pipe/nozzle inner diameter with an automated inspection tool.	

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Byron Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
2	1	RPV head penetrations 79 total - 53 CRDMs - 2 RVLIS - 5 Thermocouple - 18 spare VHPs - 1 head vent	Previous VT-2 exams performed with the RPV head insulation in-place. Bare Metal Visual (BMV) using specific inspection guidelines have been developed and will be implemented.	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	VT-2 examinations have been performed on the accessible areas of the head. These exams are conducted on the RPV head with the shroud assembly access doors opened and the RPV head insulation in-place.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time. A BMV exam is scheduled for the Fall of 2003. The extent and frequency of additional examinations to be in accordance with MRP. A partial BMV was completed during the Spring of 2002 on Unit 1 to verify that no degradation to the head had occurred from a previous leak. Approximately 80% of the RPV head was examined with no degradation found.	Insulation is removed as necessary to facilitate the BMV exam. Peripheral vertical and horizontal panels are designed for removal. The RPV head insulation is a series of 3" thick mirror insulation panels. 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 corrective actions required to date.

Byron Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
3	1	RPV lower head in-core instrumentation penetrations (58 total)	Previous VT-2 exams performed with the RPV lower head insulation panels removed as necessary to visually exam the bottom of the vessel and in-core nozzles.	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	The examination has been performed with RPV lower head insulation removed during Mode 5. The examination has been performed with RPV lower head insulation in-place looking for leakage at insulation joints.	The examination is performed each refuel outage or during a forced outage of sufficient duration. The examinations are performed twice, once during Mode 5 and then again at operating pressure and temperature, with a 4 hour hold time, beneath the RPV.	Insulation is removed to facilitate Mode 5 BMV examinations. The insulation at the bottom of the RPV is a flat, horizontal deck of stainless steel mirror panels. The deck stands off from the bottom of the lower head providing a clearance of 8". The center panels are fixed around the 58 in-core guide tubes and are not designed for removal. The peripheral panels are removable and allow access to the lower RPV head surface.	No corrective actions required to date.

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Byron Station RCPB Alloy Material Listing

Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
4	1	Pressurizer surge nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Spring of 1997. Not currently included in risk informed inspection program scope.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer lower head insulation arrangement. There are 4 segments enclosing the surge nozzle. The panels are stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe insulation joints, along the surge line piping and on the containment floor beneath pressurizer surge nozzle.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer lower head insulation arrangement. There are 4 segments enclosing the surge nozzle. The panels are stainless steel, mirror panels with buckled snaps for removal.	

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Byron Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
5	1	Pressurizer spray nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Fall of 1988.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 2 semi-circular, flat panels enclosing the spray nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the nozzle to pipe insulation joint.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 2 semi-circular , flat panels enclosing the spray nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Attachment 2

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Byron Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
6	1	Pressurizer relief valve nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Fall of 1997.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing the relief nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joint.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing the relief nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Byron Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
7	1	Pressurizer safety valve nozzle to safe-end weld (3 welds)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. The three welds were each examined in the Fall of 1988, Spring of 1990 and Spring of 1996 for the 1st Interval, and in the Fall of 1997 for the 2nd Interval.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing each of the safety nozzles. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joints.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing each of the safety nozzles. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Byron Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
8	2	RPV nozzle to safe-end welds (4 hot leg safe-ends, 4 cold leg safe-ends, 8 total).	Visual inspection for leakage (VT-2).	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe insulation joints and along the annulus of the piping penetration.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is a stainless steel reflective (mirror) design. There are 3, 120 ° segments buckled around the pipe-nozzle OD.	No corrective actions required to date.
			Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Spring of 1998.	Insulation is not removed for this exam. The exam is performed from the pipe/nozzle inner diameter with an automated inspection tool.	

Byron Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
9	2	RPV head penetrations (VHP) 79 total - 53 CRDMs - 2 RVLIS - 5 Thermocouple - 18 spare VHPs - 1 head vent	A Bare Metal Visual (BMV) using VT-2 techniques was performed in Fall 2002. Specific inspection guidelines were developed and implemented. Previous VT-2 exams performed with the RPV head insulation in-place.	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	BMV of the RPV head surface and a 360 degree view of all VHPs with slight obstruction of the head vent penetration.	The BMV exam was performed in the Fall of 2002. The extent and frequency of additional examinations to be in accordance with MRP.	Insulation is removed as necessary to facilitate the BMV exam. Peripheral vertical and horizontal panels are designed for removal. The RPV head insulation is a series of 3" thick mirror insulation panels. 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 corrective actions required to date. In addition, for the Fall 2002 examination, there were no corrective actions taken as there was no evidence of leakage on the RPV head surface and no RPV head degradation was identified.

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Byron Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
10	2	RPV lower head in-core instrumentation penetrations (58 total).	Previous VT-2 exams performed with the RPV lower head insulation panels removed as necessary to visually exam the bottom of the vessel and in-core nozzles.	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	The examination has been performed with RPV lower head insulation removed during Mode 5. The examination has been performed with RPV lower head insulation in-place looking for leakage at insulation joints.	The examination is performed each refuel outage or during a forced outage of sufficient duration. The examinations are performed twice, once during Mode 5 and then again at operating pressure and temperature, with a 4 hour hold time, beneath the RPV.	Insulation is removed to facilitate Mode 5 BMV examinations. The insulation at the bottom of the RPV is a flat, horizontal deck of stainless steel mirror panels. The deck stands off from the bottom of the lower head providing a clearance of 8". The center panels are fixed around the 58 in-core guide tubes and are not designed for removal. The peripheral panels are removable and allow access to the lower RPV head surface.	No corrective actions required to date.

Byron Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
11	2	Pressurizer surge nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Fall of 1998.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer lower head insulation arrangement. There are 4 segments enclosing the surge nozzle. The panels are stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe insulation joints, along the surge line piping and on the containment floor beneath pressurizer surge nozzle.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer lower head insulation arrangement. There are 4 segments enclosing the surge nozzle. The panels are stainless steel, mirror panels with buckled snaps for removal.	

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Byron Station, Units 1 and 2

Byron Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
12	2	Pressurizer spray nozzle to safe-end weld (1 weld).	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. Last examined in the Spring of 1990.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 2 semi-circular, flat panels enclosing the spray nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joint.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 2 semi-circular, flat panels enclosing the spray nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Byron Station RCPB Alloy Material Listing								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
13	2	Pressurizer relief valve nozzle to safe-end weld (1 weld).	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI Interval. Last examined in the Spring of 1995 for the 1st Interval and in the Spring of 2001 for the 2nd Interval.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing the relief nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joint.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing the relief nozzle. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Byron Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
14	2	Pressurizer safety valve nozzle to safe-end weld (3 welds).	Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II or III Examiners.	100% of the Risk Informed ISI examination volume, Figure 4-9 in EPRI TR-112657 Rev. B-A, "Revised Risk Informed Inservice Inspection Evaluation Procedure."	Once per ISI interval. The 3 welds were each examined in the Spring of 1990 and the Fall of 1993 for the 1st interval and in the Spring of 2001 for the 2nd interval.	Insulation has been, and will be, removed for the volumetric examination. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing each of the safety nozzles. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II or III VT-2 Examiners. Results reviewed by Certified Level III Examiners.	Exam is performed by looking for evidence of leakage around the pipe to nozzle insulation joints.	The VT-2 exams are performed twice each refuel outage: first, in Mode 3 going into the outage after cycle operation at a pressure and temperature at or slightly less than normal operating conditions, and then again in Mode 3 coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement. There are 4 segmented, flat panels enclosing each of the safety nozzles. The panels are 4" thick stainless steel, mirror panels with buckled snaps for removal.	

Byron Station RCPB Alloy Material Listing								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
15	2	Steam generator primary head drain lines (4 total).	A visual using direct VT-2 techniques was performed on the four steam generator drain lines and surrounding head surface. The "C" steam generator visual inspection was performed during the June 2002 forced outage and the "A", "B" and "D" steam generator visual inspections were performed during the Fall 2002 refuel outage. Previous VT-2 exams performed with the steam generator lower head insulation in-place.	Exams performed by Certified Level II VT-2 Examiners.	The drain line, weld, and lower head surface around the drain line were examined for steam generators "A", "B", "C" and "D".	The direct visual examination is performed each time the steam generator primary channel head is de-insulated, (typically for eddy current inspection).	The insulation is removed to facilitate the examination. The insulation around the steam generator lower head is a series of removable stainless steel mirror panels. The bottom panel is a horizontal disc that is set off of the steam generator head and encloses the head, the drain line, and the drain line isolation valve.	No corrective actions required to date.

2. Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

Response

Except for bolted connections on borated ASME Section XI Class 1, 2 and 3 systems discussed below, Byron Station procedures do not require that insulated RCPB components be de-insulated to perform VT-2 examinations. The basis for this has been that leakage from RCPB components should be detectable at insulated joints or surrounding areas given that the systems have been at normal pressure for, in most cases, a full operating cycle. While the removal of insulation is not necessarily required for the performance of a VT-2 exam, if evidence of leakage or boric acid residue is detected, the procedure requires that the leakage source be located which may require insulation removal.

In general, Byron Station ASME Section XI Class 1, 2, and 3 components in the containment containing borated water, if insulated, has removable stainless steel reflective style insulation. Outside the containment, blanket insulation is used. Most insulation is removable, but there are exceptions. The insulation panels around the incore instrumentation guide tubes at the bottom of the RPV lower head are not intended to be removed. A set of peripheral horizontal panels are buckled in place and can be removed for access to the incore instrumentation guide tubes at the bottom of the RPV lower head. The horizontal panels around the VHPs are not removable; however, the side vertical panels are removable. The specific style of insulation for other locations in the RCPB is listed in the table in the response to Question 1.

Bolted Connections

Insulated bolted connections in the RCPB (i.e., ASME Section XI, Class 1) and in borated ASME Section XI Class 2 and Class 3 support systems have insulation removed in order to perform ASME Section XI VT-2 examinations. The scope includes bolted connections that are installed in systems that are borated for the purpose of controlling reactivity. The ASME Section XI requirements on the extent of insulation removal and the plant conditions under which the insulation is removed have been modified by Byron Station Inservice Inspection Relief Requests I2R-11, Rev. 2 and I2R-34. Both of these alternatives to ASME Section XI requirements have been authorized for use at Byron Station by the NRC.

Relief Request I2R-11 allows for insulation removal and the performance of the VT-2 exam on bolted connections in borated systems to be performed with the system depressurized. The approved alternative requires that a system be pressurized for a minimum of four hours at normal operating pressure prior to the VT-2 examination. Additionally, for ASME Section XI Class 2 and 3 borated systems, VT-2 examinations are performed on approximately 36 month frequencies, which coincides with plant refueling outages, not allowing the period between inspections on individual components to exceed 45 months. This frequency for individual components is more restrictive than the "Periodic Frequency" allowed by ASME Section XI for Class 2 and 3 systems described in tables IWC-2500 or IWD-2500.

Relief Request I2R-34 allows the removal of insulation from certain ASME Section XI Class 1 valves for VT-2 examination to be performed on an extended frequency. The insulation is removed from the bolted connections and a VT-2 examination is conducted, with the system depressurized, on a once per 10 year interval frequency. These valves are also VT-2 examined with the insulation installed after a minimum four hour hold time at normal operating pressure at the end of each refueling outage and in Mode 3 during shutdown for each refueling outage.

Dissimilar Metal Welds

Insulated dissimilar metal welded piping connections that contain Alloy 82/182 are typically not de-insulated for VT-2 examinations. A list of these nozzle to safe-end welds and the type of insulation is provided in the table in the response to Question 1. Examinations of these areas are performed in response to GL 88-05 and are conducted in Mode 3 going into the outage, typically, after a cycle of operation. Therefore, an adequate time is allowed for leakage to propagate through the insulation joints and be observed by direct VT-2 examination.

The eight RPV nozzle to safe-end welds are considered inaccessible to perform direct VT-2 examination. The welds are located in an area between the concrete RPV shield wall and the concrete primary shield wall. This area enclosing the RPV nozzles and connected piping is referred to as the "sand box" area. The sand box area is only accessible from above, from the refueling cavity floor, by removing normally sealed steel plates. The normal technique for viewing potential leakage in this area is to look along the reactor coolant piping as it passes though the annulus of the bio-shield wall towards the RPV. The piping is insulated, so the examiner looks for evidence of boric acid at the insulation joints inside the annulus. The RPV nozzle to safe-end weld insulation is not routinely removed since the sand box area is considered a high dose, confined space.

3. Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

Response

For the RCPB components, Byron Station personnel perform walkdowns during refueling outages, forced outages and, depending on circumstances, during power operations. Typical at-power containment walkdowns of accessible areas for leakage would be initiated at the request of the Operations Department if unidentified leakage was trending up or at the request of Radiation Protection personnel if there was an unexpected increase in containment atmosphere gas or particulate levels. For at-power walkdowns, the area of interest may be limited for as low as reasonably achievable (ALARA) considerations. VT-2 certified personnel perform the walkdowns during refueling outages and, in most cases, during forced outages and at power operations.

As described in the response to Question 2, the areas considered inaccessible are the "sand boxes" which enclose the eight RPV nozzles. Leakage from these areas is identified by visual examination along the horizontal surface of the insulated piping as it passes through the concrete annulus.

Byron Station uses a containment floor drain sump and a reactor cavity sump to collect, measure and record unidentified leakage in accessible and inaccessible areas in the containment. Both of these sumps are instrumented to identify leakages of 1.0 gpm within one hour and are recorded and alarmed in the main control room. The containment floor drain and reactor cavity sump inputs are checked each shift. A reactor coolant system (RCS) mass balance is performed when unidentified leakage is suspected and at the prescribed Technical Specification intervals. This provides early indication to the operator of potential unidentified leakage. Also, the reactor makeup control system is used to maintain proper reactor coolant inventory, volume control tank (VCT) level is continuously recorded and quantities of boric acid and makeup water injected are totaled and flow rates recorded in the control room.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Radioactivity detection systems are used for monitoring both particulate and gaseous activities and can be used to identify RCS leakage. The detection of RCS leakage using radiation monitors depends on the concentration of radioactivity in the RCS and detector background count rate.

Air temperature and pressure monitoring methods may also be used to infer unidentified leakage to the containment. Although containment temperature and pressure fluctuate slightly during unit operation, a rise above the normally indicated range of values may indicate RCS leakage into the containment.

4. Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections), to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,

> ***a. If observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or***
>
> ***b. If observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.***

Response

Byron Station personnel use engineering procedures BVP 200-10, "ISI/IST Recordable Indication Investigations," BVP 200-10T1, "Recordable Indication Record," and BVP 200-10T3, "ASME Section XI Bolted Connection Evaluation." These procedures describe the requirements and instructions for the evaluation of ASME Section XI Class 1, 2, and 3 bolted connections when external leakage is detected. These procedures provide instructions for the corrective actions that must be implemented in the event the structural integrity of any ASME Section XI Class 1, 2, or 3 bolted connection is determined to be suspect.

When evidence of leakage is identified from a bolted connection, the procedure requires an evaluation to be performed. This evaluation considers the location of the leak, leak rate, extent of deposit accumulation, extent of wastage, corrosiveness of process fluid, materials, length of time bolting has been in service and the effect on other structures or components. A condition report is initiated for non-conforming conditions and a recommendation for repair, replacement, or monitoring is provided. Active leakage from the RCS is repaired prior to completing a refuel outage.

The bases for the evaluation/determination criteria of BVP 200-10T3 are derived from the provisions of Byron Station Inservice Inspection Relief Request I2R-12. This request was approved by the NRC.

Byron Station uses the corrective action process, specifically a condition report, to determine the acceptability for continued operation when a non-conforming condition is identified. Within the Condition Report, operability is considered and may require a formal evaluation using procedure LS-AA-105, "Operability Determinations." If the operability assessment determines that the non-conforming condition is acceptable, inspection/monitoring actions may be established and tracked using LS-AA-105-1001, "Supporting Operability Documentation." For a non-conforming condition that is determined to be unacceptable, repair or replacement is required.

5. Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

<u>Response</u>

As explained in the table in the response to Question 1, items 3 and 10, the insulation at the bottom of the RPV lower head stands off of the head surface by a minimum of eight inches at the very bottom of the lower head curvature. Visual examinations of the Byron Station, Unit 1 and Unit 2 RPV lower head, scheduled for the Fall of 2003 and Fall of 2003, respectfully, will require removing insulation panels to view the metal surface. Any leakage from lower head penetrations would be visible. This examination is normally performed twice each refueling outage as described in the table in the response to Question 1.

Reactor coolant and boric acid deposits from a lower penetration leak would collect on the inner surface of the flat, horizontal insulation panels. The incore penetration instrument tubes that connect to the nozzles are stainless steel and have an increased resistance to boric acid corrosion/wastage. If boric acid were to leak through the seams or opening of the insulation, there are no pressure retaining components beneath the RPV that could be affected. This leakage from the RPV bottom head nozzles would collect in the reactor cavity sump described in the response to Question 3. The reactor cavity sump input is recorded each shift and abnormal readings are required to be reported to the Shift Manager. This notification would result in an evaluation and a condition report would be initiated, as necessary.

6. Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response

The leakage detection capabilities at Byron Station were discussed in the response to Questions 3 and 5 above. The visual examination and evaluation procedures at Byron Station require that components in the area of a leak be examined and evaluated. In the case of Byron Station, the vast majority of components in the RCPB are stainless steel.

7. Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

Response

Vessel Head Penetrations (VHPs)

Byron Station personnel have based the examination requirements for the Unit 1 and Unit 2 VHPs on the EPRI Material Reliability Project (MRP) susceptibility model for PWSCC. This ranking (i.e., MRP-48) identifies Byron Station, Unit 1 and Unit 2 as "low susceptibility" plants rated 66th and 67th, respectively, out of the 69 PWR units in the study. As of September 2002, Byron Station, Unit 1 had an effective degradation year (EDY) value of 1.8 and as of the next refueling outage for Unit 2 in September of 2003, an EDY of 2.0, putting the Units in the lowest of the three susceptibility categories established by the NRC in Bulletin 2002-02, "Reactor Pressure Vessel Head and RPV Head Penetration Nozzle Inspection Programs."

Because of this lower susceptibility, Byron Station personnel perform qualified, effective bare metal visual (BMV) inspections of the RPV head surface and VHPs. The BMV inspection for Byron Station, Unit 2 was completed in the Fall of 2002 with no evidence of VHP leakage found. The BMV inspection for Byron Station, Unit 1 is scheduled for the Fall of 2003. A partial BMV was completed during the Spring 2002 on Unit 1 to verify that no degradation to the head had occurred from a previous leak. Approximately 80% of the RPV head was examined with no degradation found.

Steam Generator Drain Lines

As listed in the table in the response to Question1, the steam generator primary head drain lines for Byron Station, Unit 2 are made of Alloy 600 and Alloy 82/182 materials. The bottom surface of the steam generator primary head, the 3/8" drain lines and drain line isolation valves are enclosed in the lower head insulation package. Based on a susceptibility model developed by Westinghouse, Byron Station is visually examining the four Unit 2 drain lines whenever the steam generator lower head insulation package is removed to support the eddy current testing of the steam generator tubing.

All four steam generator drain lines were visually examined as described in the table in the response to Question 1. There were no recordable indications, no signs of boric acid deposits or any degradation of the carbon steel lower head surface.

8. Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

Response

Vessel Head Penetrations

As stated above in the response to Question 7, Byron Station personnel are following the examination recommendations of the industry effort on the Alloy 600/82/182 issue. Westinghouse, the reactor vendor, and Babcock & Wilcox, the reactor manufacturer, are part of this industry effort. Based on the lower susceptibility of the Byron Station, Unit 1 and Unit 2 VHP nozzles, the recommended visual examinations are being performed. Byron Station personnel have followed all MRP recommendations and will comply with the recent MRP recommendations as stated in the letter from Leslie Hartz (Chair, MRP) to the EPRI PWR Materials Management Program (PMMP) Steering committee, dated December 2, 2002.

RPV Nozzle Safe-Ends, Reactor Lower Head Nozzles, and Pressurizer Nozzle Safe-Ends

The current program for the examination of these components is listed in the table in the response to Question 1. Exelon Nuclear corporate and Byron Station personnel are currently working with Westinghouse on developing a comprehensive inspection, repair and/or mitigation program for all Alloy 600/82/182 components in the RCPB.

9. Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.

Response

Byron Station Technical Specification (TS) Limiting Condition for Operation (LCO) 3.4.13.a states that there shall be no pressure boundary leakage. Pressure boundary leakage is defined as leakage, except steam generator tube leakage, through a non-insoluble fault in a reactor coolant system component body, pipe wall, or RPV wall. If pressure boundary leakage is detected, the action statements for this LCO require that the affected unit be in Mode 3 in six hours and be in Mode 5 in 36 hours. The resolution of leakage indications in the corrective action program requires evaluation of the impact on this TS.

Compliance with the zero non-isolable leakage criteria is met by performing GL 88-05 examinations, conducting inspections and repairs in accordance with ASME Section XI, and 10 CFR 50.55a, "Codes and standards." In addition, the unidentified leakage limit of one gpm defined in TS LCO 3.4.13.b is established as a quantity that can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a short time, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

10 CFR 50.55a, requires that inservice inspection and testing be performed in accordance with the requirements of ASME Section XI. ASME Section XI contains applicable rules for examination, evaluation and repair of code class components, including the RCPB.

For this, the 2nd Inservice Inspection Interval, Byron Station personnel have implemented the 1989 edition, with no addenda, of ASME Section XI. Paragraph IWA-5250 (b), "Corrective Measures," of this edition states:

> "If boric acid residues are detected on components, the leakage source and the areas of general corrosion shall be located. Components with local areas of general corrosion that reduce wall thickness by more than 10% shall be evaluated to determine whether the component may be acceptable for continued service, or whether repair or replacement is required."

To incorporate these requirements, Byron Station personnel use Exelon Nuclear Procedure ER-AA-335-015, "VT-2 Visual Examination." Paragraph 4.6.1.4 of this procedure states:

> *"If boric acid residues are detected on components, then* ***LOCATE*** *the leakage source and the areas of general corrosion.* ***EVALUATE*** *components with local areas of general corrosion that reduce the wall thickness by more than 10% to determine whether the component may be acceptable for continued service, or whether repair or replacement is required."*

The Exelon Nuclear procedure LS-AA-125, "Corrective Action Program (CAP) Procedure" defines the requirements for condition identification, condition review, investigation, and closeout.

ATTACHMENT 3

Response to Request for Additional Information Regarding NRC Bulletin 2002-01, Three Mile Island Station, Unit 1

AmerGen Energy Company, LLC (AmerGen)

On November 21, 2002, the NRC issued a Request for Additional Information (RAI) for NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The below information was required within 60 days of the date of the RAI for TMI Unit 1 that was categorized as a Bin 2 plant in the RAI.

NRC Question

1. Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel (RPV) bottom head).

Response

Introduction / Background

Regulatory Requirements:

10 CFR 50.55a, "Codes and standards," identifies the codes and standards requirements for operating nuclear power plants. Section (f) identifies the in-service testing requirement and section (g) identifies the in-service inspection requirements. For facilities whose construction permit was issued prior to July 1, 1971 (the construction permit for Three Mile Island Unit 1 was issued on May 18, 1968), components are required to meet the requirements of paragraphs (g)(4) and (g)(5) to the extent practical. Paragraph (g)(4) states, "...components (including supports) which are classified as American Society of Mechanical Engineers (ASME Code) Class 1, Class 2 and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) ....to the extent practical within the limitations of design, geometry and materials of construction...."

The ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," serves as the technical basis for the inspections, techniques, scope and extent of coverage, inspection frequency, personnel qualifications, and extent of insulation removal for those components listed in the following table. TMI Unit 1, currently in the 3rd Inservice Inspection Interval, is committed to the 1995 Edition, through the 1996 Addenda, of ASME XI. In addition, the visual examinations of the RCPB and associated systems, structures, and components are supplemented by the requirements of the TMI Unit 1 commitment to Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

Development of the Boric Acid Corrosion Control (BACC) Program at TMI Unit 1 (since 1982):

On June 2, 1982, the NRC issued IE Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants." The purpose of this bulletin was twofold; first, to notify licensees and construction permit holders about incidents of severe degradation of threaded fasteners (bolts and studs) in closures in the reactor coolant pressure boundary (RCPB) and secondly, to require appropriate actions. The scope of the resulting action items included threaded fasteners (studs or bolts) in steam generator and pressurizer manway closures, valve bonnets and pump flange connections installed on lines having a nominal diameter of six inches or greater and control rod drive flange and pressurizer heater connections that do not have seal welds to provide leak-tight integrity.

The response identified bolted closures of the RCPB that had experienced leakage with the resultant inspections made and corrective measures taken to eliminate leakage. It also identified RCPB closures and connections where fastener lubricants were used and the lubricant's composition. The only instance of the use of injection sealant was on the decay heat drop line and was reported, as required. A review of maintenance procedures and training was performed to assess threaded fastener practices. Maintenance procedures were determined to be adequate to meet the requirements specified in IE Bulletin 82-02. In addition, training on bolting procedures was added to the mechanical maintenance training program. Finally, a report of specific connections examined since the issuance of NRC Information Notice No. 82-02 was provided to the NRC in letters dated August 3, 1982 and December 5, 1985. No evidence of RCPB stud or bolt failures was noted.

In 1989, to address concerns identified in NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel RCPB Components in PWR Plants," TMI Unit 1 developed Technical Data Report (TDR) No. 946, "TMI-1 Evaluation of Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components". The existing TMI Unit 1 boric acid corrosion inspection requirements are based on the recommendations of this report. This report identified specific RCPB locations most vulnerable to concentrated boric acid corrosion and the carbon steel targets of this leakage. These targets include;

- reactor vessel,
- pressurizer,
- steam generator upper and lower heads,
- hot and cold leg piping,
- reactor coolant pump casing to piping bi-metallic weld,
- studs on all manway / handholds or nozzles or valve packing assemblies,
- core flood and high pressure injection nozzle safe ends,
- pressurizer spray and surge lines, and
- carbon steel sections of instrument piping and carbon steel valve bodies.

TDR No. 946 also provides clarification of the methodology for assessing component damage. Programmatic controls were developed to provide continuing assurance of extremely low probability of significant degradation to the RCPB components due to concentrated boric acid corrosion.

TDR No. 946 divided RCPB components into three groups according to the potential consequences of their leakage onto other RCPB carbon steel components. This was intended to assist in placing appropriate emphasis on leakage inspections and evaluations. Group 1 components are RCPB components with mechanical joints, and the RCPB targets of the leakage. Group 2 components are RCPB components with mechanical joints for which unchecked corrosion of their targets could result in a RCPB leak equivalent to a leak from a line size greater than one inch (within the plant's normal makeup capability). Group 2 components are a subset of Group 1 components. Group 3 components are RCPB components with mechanical joints for which unchecked corrosion of their targets could result in a RCPB failure that exceeds the plant's normal makeup capability without first alerting the operator by exceeding the unidentified leak rate limit. Group 3 components are a subset of Group 2 components.

The reactor pressure vessel incore monitoring instrument (IMI) nozzles are not addressed as a component in TDR No. 946. A detailed discussion of the IMI nozzles is provided in response to Question No. 5.

All Group 1, Group 2 and Group 3 components, as well as others identified in TDR No. 946, are identified for ASME Section XI visual inspections (VT-2) prior to startup following each refueling outage. Each item has an individual sign-off to document the inspection. If evidence of leakage is noted during these inspections, programmatic requirements are in place to ensure that an engineering evaluation of the safety consequences of identified leakage is performed, and the condition corrected via corrective maintenance or component replacement. Additional non-ASME Section XI walkdowns are performed during plant cool down conditions to identify any evidence of leakage.

TDR No. 946, dated January 10, 1989, acknowledges the importance of the boric acid corrosion control issue and concludes with:

> "Borated water leakage onto the RCPB can result in severe corrosion damage. The cooling effect of a leak on a hot RCPB surface can be sufficient to keep the surface moist, allowing for development of highly corrosive boric acid solutions. Laboratory corrosion data and documented cases of severe RCPB corrosion demonstrate the problem. Boric acid corrosion rates of up to 1 to 2 inches per year are expected for carbon steels at RCS operating temperatures.
>
> Minimizing the chance of severe RCPB degradation from boric acid corrosion is crucial to safe plant operation. At TMI Unit 1, a program of leakage inspections coupled with engineering evaluations and maintenance is the primary means of minimizing the probability of corrosion damage."

Subsequently, in response to NRC Information Notice 90-10: "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600" (February 23, 1990), TMI Unit 1 commissioned a study of the use of nickel-base alloys. B&W Nuclear Technologies completed their report, "Nickel-Base Alloy Usage at TMI-1" (Report No. 51-1179885) in June 1990. This proprietary report provides a summary listing of nickel-base alloys used in the construction of the TMI Unit 1 reactor coolant system (excluding fuel assemblies and their control components) to be used in assessing possible in-service inspection locations. This report serves as a basis for the component identification in the attached table. BWNT Report No. 51-1179885 also identified that the effect of temperature on the initiation time for Alloy 600 SCC is an important variable, and that Alloy 600 failures have typically occurred at operating temperatures greater than approximately 600 F.

In accordance with the recommendations of BWNT Report No. 51-1179885, TMI Unit 1 inspects all pressurizer nozzle to safe end welds and the reactor vessel internals upper core barrel bolts during each 10-year in-service inspection period. These sites were characterized as most susceptible to Alloy 600 SCC. Other locations that were identified as possibly susceptible to Alloy 600 SCC, based strictly on temperature, were the reactor vessel and head and the reactor coolant piping. This report does not identify the incore monitoring instrumentation nozzles as a priority for Alloy 600 SCC inspections because of the lower operating temperatures.

In 1998, Framatome Technologies completed a study to rank on a relative basis each of the Alloy 600 components for potential failure due to PWSCC. A ranking was performed for each of the B&W Owners Group Plants. The results of this report, "Alloy 600 PWSCC Susceptibility Model, 51-5001951-01, December 16, 1998," are discussed in the response to Question No. 7.

Insulation removal is discussed in detail in response to Question No. 2.

<u>Reactor Pressure Vessel Bottom Head Construction</u>:

The reactor vessel is constructed of low alloy steel. The design pressure is 2500 PSI and the design temperature is 650 degrees F. The sides of the vessel are constructed of SA-533B, Class 1 plate with a stainless steel clad on the inner surface. The minimum wall thickness is 8.4 inches. The lower head is a forging of A-508 material with a minimum thickness of 5 inches. This component is also clad with stainless steel. There are 52 incore instrument nozzles that penetrate the lower vessel head. These are fabricated from ASME SB-166 Alloy 600 (Inconel) barstock. This material was supplied by B&W Tubular Products Division with heat number M6378. The initial size before machining was 2.25 inches outer diameter. This material was either hot rolled or hot finished, annealed and pickled. As stated in BWNT Report No. 51-1179885, these nozzles were most likely installed with a GTAW root pass with Alloy 82 weld metal then completed using the manual metal arc technique with Alloy 182 weld metal.

The reactor vessel was constructed under the following governing specifications: 1) The American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III "Rules for Construction of Nuclear Vessels," 1965, and all applicable Code cases and Addenda for Class A vessels as of June 20, 1967, 2) The American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section IX, Welding Qualifications, 1965, 3) The Babcock and Wilcox Company Equipment Specification CS-3-22, dated May 26, 1970, and 4) The Babcock and Wilcox Company, Quality Control Department Specifications covering Welding, Non-Destructive Testing, Heat Treatment, Cleaning and Testing.

The area below the bottom head insulation is surrounded by concrete and is referred to as the reactor vessel cavity area. This area is located within the primary shield. The area is approximately twelve feet in diameter and approximately nine feet high. Other than the incore instrument piping, there is no equipment in this area. A personal access hatch (approximately four feet high by four feet wide) and three ventilation ports (approximately one foot diameter) are provided through the structural concrete. Access to this area is restricted (due to dose rates) when the reactor is critical or when any incore instrumentation is pulled from the fuel. This area is not normally entered when the reactor coolant system is heated.

Insulation Design:

The thermal insulation for the reactor vessel is reflective type, designed and manufactured by Diamond Power Specialty Corporation and known as SS-MIRROR or MIRROR insulation. The insulation is of 100% stainless steel construction with carbon steel support members. The insulation was provided as pre-fabricated panels with adjacent panels matched to fit (tapped screw holes in overlapping members at the joints). The insulation is designed with clearances between the reactor vessel outer diameter and the inside diameter of the insulation assemblies to allow for thermal expansion of the vessel. The insulation panels are banded together with stainless steel bands.

For the bottom head panel placement, three sections of support steel were used. These were welded together after being put in place in the cavity area. Insulation panels were then bolted through one-inch diameter cutouts and attached with three-inch square washers, lock washers and nuts. This assembly was then raised and attached to the concrete wall using self-drilling anchors.

The following table provides a listing of the requested information of the Alloy 600 pressure boundary material and Alloy 82/182 welds and locations in the TMI Unit 1 Reactor Coolant Pressure Boundary (RCPB).

Some of the Alloy 600 and Alloy 82/182 materials are not listed in this table because they are internal to RCPB components and, assuming their failure, do not have a potential to degrade the RCPB with boric acid leakage. For the reactor vessel, these items include the guide lugs, reactor internal bolting block plate, the internal jack- screw locking cup and bolt locking cup. For the pressurizer, the spray nozzle connection inside the pressurizer is not included. In the once-through steam generators, the items are the steam generator tubing and the tube sheet cladding.

Response to RAI Regarding NRC Bulletin 2002-01, Three Mile Island Station, Unit 1

Three Mile Island Unit-1 Alloy 600 pressure boundary material and Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB)								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
1	1	Reactor vessel nozzle to safe-end welds (2 core flood nozzles)	Visual Inspection for Leakage (VT-2)	Exams performed by Certified Level II VT-2 examiners.	Exam is performed by looking for evidence of leakage around the pipe insulation joints and along the annulus of the piping penetration.	The VT-2 exams are performed each refuel outage: coming out of the outage at normal operating temperature and pressure with a 4 hour hold time. Non-ASME Section XI walkdowns are performed during plant cool down conditions to identify any evidence of leakage.	The insulation is not removed for the VT-2 examinations. The insulation is a stainless steel reflective (mirror) design around the pipe-nozzle OD.	No corrective actions required to date.
			Volumetric (ultrasonic) examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Level II/III Examiners.	100%	Once per 10 years. Last examined in the Fall of 2001. Next scheduled exam for TMI-1 is Fall 2011.	Insulation is not removed for this exam. The exam is performed from the pipe/nozzle inner diameter with an automated inspection tool.	

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Three Mile Island Unit-1 Alloy 600 pressure boundary material and Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB)								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
2	1	Reactor vessel head penetrations (VHP) 71 total - 69 CRDMs - 1 RVLIS - 0 Thermocouple (Plugged) - 1 head vent	Previous VT-2 exams performed with the RPV head insulation in-place. The reactor vessel head is scheduled to be replaced with a new head in October 2003.	Exams performed by Certified Level II VT-2 examiners.	Exam is performed by looking for evidence of leakage around the pipe and insulation joints.	The VT-2 exams are performed each refuel outage: coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The RV head insulation is a series of mirror insulation panels. The insulation is installed in a flat field across the top of the RPV head. Peripheral vertical and horizontal panels are designed for removal. Insulation is not removed for VT-2 examination at normal operating temperature and pressure.	Bare metal, UT and PT examinations and corrective actions taken in the 1R14 refueling outage are described in AmerGen letter to the NRC dated April 1, 2002 (5928-02-20091) in response to NRC Bulletin 2002-01, 15 day response requirement.

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Three Mile Island Unit-1 Alloy 600 pressure boundary material and Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB)								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
3	1	RPV lower head in-core instrumentation penetrations (52 total)	Previous VT-2 exams performed with the RPV lower head insulation in-place. TMI Unit 1 is planning to install a camera above the insulation to visually inspect the bare metal around the incore instrumentation nozzle penetrations during the refueling outage currently scheduled for October 2003 (T1R15).	Exams performed by Certified Level II VT-2 examiners.	Exam is performed by looking for evidence of leakage around the pipe and insulation joints.	The examination is performed once during each refuel outage.	For previous exams, the insulation has not been removed. The insulation at the bottom of the vessel is a flat, horizontal deck of stainless steel mirror panels. The deck stands off from the bottom of the lower head providing a clearance of about 6 inches. The panels are fixed around the 52 in-core guide tubes.	After the 9R refueling outage (1991), cleaning was performed by removing boron residue from six suspect guide tubes and from within the primary shield, and performing a water flush of the area including the vessel insulation, primary shield walls and guide tubes and within the cooling annulus conducted from the seal plate area.
4	1	Pressurizer surge nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) examination of the nozzle to safe-end welds. A surface exam is also required by Section XI Code.	Certified Ultrasonic (UT)/ Penetrant Level II/III Examiners.	100%	Once per 10 years. Last examined in the Fall of 1999. Next scheduled exam for TMI-1 is Fall 2009.	Insulation has been, and will be, removed for the volumetric examination.	No corrective actions required to date.

Response to RAI Regarding NRC Bulletin 2002-01, Three Mile Island Station, Unit 1

Three Mile Island Unit-1 Alloy 600 pressure boundary material and Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB)								
Item	Unit	Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/ Insulation Type	Corrective Action
			VT-2	Exams performed by Certified Level II VT-2 examiners.	Exam is performed by looking for evidence of leakage around the pipe and insulation joints.	The VT-2 exams are performed each refuel outage: coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is a series of form fitting stainless steel, mirror styled panels with stainless steel bands that allow for removal.	
5	1	Pressurizer spray nozzle to safe-end weld (1 weld)	Volumetric (ultrasonic) and Surface examination of the nozzle to safe-end welds.	Certified Ultrasonic (UT) Penetrant Level II/III Examiners.	100%	Once per 10 years. Last examined in the Fall of 2001.	Insulation has been, and will be, removed for the volumetric examination.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II VT-2 examiners.	Exam is performed by looking for evidence of leakage around the pipe and insulation joints.	The VT-2 exams are performed each refuel outage: coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement (NUKON).	
6	1	Pressurizer relief valve nozzle to flange weld (1 weld)	Surface exam , (Less than 4 Nominal Pipe Size (NPS)	Certified Penetrant Level II/III examiners.	100%.	Once per 10 years. Last examined in the Fall of 1997.	Insulation has been, and will be, removed for the surface examination.	No corrective actions required to date.

Response to RAI Regarding NRC Bulletin 2002-01, Three Mile Island Station, Unit 1

Three Mile Island Unit-1 Alloy 600 pressure boundary material and Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB)								
Item	**Unit**	**Component**	**Inspection Techniques**	**Personnel Qualifications**	**Extent of Coverage**	**Frequency**	**Degree of Insulation Removal/ Insulation Type**	**Corrective Action**
			VT-2	Exams performed by Certified Level II VT-2 examiners.	Exam is performed by looking for evidence of leakage around the pipe and insulation joints.	The VT-2 exams are performed each refuel outage: coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement (NUKON).	
7	1	Pressurizer safety valve nozzle to flange weld (2 welds)	Surface exam, less than NPS 4.	Certified Penetrant Level II/III Examiner.	100%	Once per 10 years. The 2 welds were last examined in the Fall of 1997.	Insulation has been and will be removed for the surface examination.	No corrective actions required to date.
			VT-2	Exams performed by Certified Level II VT-2 examiners.	Exam is performed by looking for evidence of leakage around the pipe and insulation joints.	The VT-2 exams are performed each refuel outage: coming out of the outage at normal operating temperature and pressure with a 4 hour hold time.	The insulation is not removed for the VT-2 examinations. The insulation is part of the pressurizer head insulation arrangement (NUKON).	
8	1	RC drain line nozzle to safe-end and safe-end to nozzle welds (6 welds) RC pumps pipe to elbow and pipe to pipe welds (9 welds)	Volumetric (ultrasonic) and surface exam for > NPS 4 Surface exam for < NPS 4	Certified Ultrasonic (UT) Level II/III Examiners. Certified Penetrant Level II/III Examiners	100%	Once per 10 years per ISI schedule.	Insulation has been and will be removed for the examinations.	No corrective actions required to date.

NOTE : The MIRROR insulation on the pressurizer and once-through-steam-generator upper heads have been replaced with NUKON insulation, manufactured by Owens-Corning Fiberglass Corporation.

Response to RAI Regarding NRC Bulletin 2002-01, Three Mile Island Station, Unit 1

Results of Previous Inspections of the Reactor Vessel Bottom Head Cavity Area:

Due to transfer canal water leakage past the seal plate, an inspection was performed in the cavity area during 1991 (9R refueling outage). In summary, the area was cleaned and no primary system leakage was present. The following provides a summary of the 9R Outage inspection:

During 9R, Plant Engineering performed inspections of remote areas of TMI-1. One such area was within the primary shield underneath the reactor vessel lower head. This inspection was performed when all the incores were removed from the guide tubes awaiting replacement. The initial entry into the area found boron deposition on numerous incore tubes and the interior wall of the primary shield opposite the exhaust of the AH-E-2 fans. The examination during this entry was video taped. It was noted that water was dripping down from the cooling annulus in the vicinity of the incore chase.

After review of the videotape, Plant Engineering noted the amount of boron accumulation and the arrangement of deposition under the vessel. Engineering suspected that heavy localized deposits may have been a result of a leak in one or more incore guide tubes with mixing by the reactor cavity fans. As a result, plans were made to remove boron from six suspect stainless steel guide tubes, perform a visual inspection for a leak followed up by an NDE examination, application of a freeze seal and a 1000 PSI hydro of each tube from the incore seal plate. Each of the six guide tubes was successfully tested with no leakage identified.

Boron removal work was then performed within the primary shield followed by a water flush of the area including the vessel insulation, primary shield walls and guide tubes. After this work was completed, a water flush was also performed within the cooling annulus conducted from the seal plate area. The flush was concentrated in the area where seal plate leakage had been earlier detected. A final entry was made to remove standing water and clean the floor.

The final assessment for boron accumulation was as follows:

- There was 3 to 4 inches of standing water being retained within the primary shield.
- Water leakage was from the seal plate while the transfer canal was flooded.
- The dispersion of boron was due to the air turbulence induced by the AH-E-2 fans entering the cavity near floor level to the left of the primary shield entrance.
- The total dose for the effort was 1.8 man-rem of which 0.6 was attributed to boron removal activities.

Subsequent to the 9R refueling outage transfer canal seal plate leakage, Operations surveillance OPS-S419 was created. This task inspects the reactor vessel cavity area and the area within the reactor head support skirt. A summary of these results, based on a review of the last three refueling outage inspections, follows:

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- In 1997, standing water was found in the cavity area with some minor rusting noted.
- In 1999 (13R), no abnormal conditions were noted and a video tape was made. The videotape was reviewed with no significant abnormal conditions noted. However, a thin layer of water on the floor and some minor boron streaks were noted.
- In 2001 (T1R14), only photos were taken due to dose rate that limited entry into the cavity area. These photos were reviewed with no significant abnormalities noted, except for minor rusting of the air inlet piping and minor boron streaks on shield wall.
- The 1999 and 2001 conditions noted are indicative of some minor transfer canal seal plate leakage.

In conclusion, the reactor cavity area underneath the vessel has been inspected every refueling outage since at least 1991 (9R) with no significant abnormal conditions found. These visual inspections (VT-2) did not specifically examine the bottom of the reactor vessel where the incore instrument nozzles penetrate the vessel because insulation completely covers this area and prevents visual inspection of the actual penetration. These inspections did assess the general condition of this area and did identify indications of boron accumulation (most probably from canal seal plate leakage). Based on previous results, we are confident that any significant indications of leakage from the incore instrument nozzles would have been identified during these visual inspections.

Locations Where RCS Leakage has the Potential to Come Into Contact with the RPV Bottom Head:

There are no components located below the reactor vessel in the cavity area except the incore monitor tubes. These provide the only potential source of leakage located below the bottom head. Leakage onto the bottom head from elevations above can only originate from components whose weld interface is contained within the primary shield enclosure. For Alloy 600/82/182 welds, these components include:

- Reactor vessel upper head control rod housing bodies (69)
- Reactor vessel upper head thermocouple nozzles (2)
- Reactor vessel lower head incore monitoring instrumentation nozzles (52)
- Core flood nozzle to safe-end welds (2)

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NRC Question

2. Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

Response

The basis for determining whether or not insulation is removed for examinations at TMI Unit 1 is consistent with ASME Section XI allowances which reflect practicality, cost, and dose considerations. The degree of insulation removal and insulation design is described in the Table provided in the response to Question No. 1.

Piping and Bolted Connections:

The visual examination (VT-2) of borated systems (for the purpose of controlling reactivity) requires insulation to be removed from pressure retaining bolted connections. For other components, the visual examination may be conducted without the removal of insulation by examining the accessible and exposed surfaces and joints of the insulation. The ASME requirements on the extent of insulation removal and the plant conditions under which the insulation is removed, have been modified by TMI Unit 1 Inservice Inspection Relief Requests RR-00-08 and RR-00-09. Both of these alternatives to the ASME Section XI requirements have been authorized for use at TMI Unit 1 by the NRC.

Relief Request RR-00-08 allows the performance of the leakage tests on bolted connections in Class 1 borated systems to be performed without the removal of insulation. The approved alternative requires that a system be pressurized for a minimum of four hours at normal operating pressure prior to the VT-2 examinations (to allow for leakage propagation from the insulation). Additionally, the insulation is removed from Class 1 bolted connections and a VT-2 visual examination is conducted with the system depressurized. These inspections are performed each refueling outage (nominal 24 month frequency).

Relief Request RR-00-09 allows the insulation around the bolted connections around the electrical pressurizer heater connections to remain installed during the VT-2 examinations of this area. The approved alternative requires that a system be pressurized for a minimum of four hours at normal operating pressure prior to the VT-2 examinations (to allow for leakage propagation from the insulation). These inspections are performed each refueling outage (nominal 24 month frequency) with the plant at hot shutdown.

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For insulated piping and components, a four-hour hold time at nominal operating pressure or hydrostatic test pressure is required as specified in the implementing procedures. For systems borated for the purpose of controlling reactivity, insulation is removed from pressure retaining bolted connections for visual examination VT-2 (unless relief has been authorized). Where normal access methods allow, pressure-retaining nuts, studs, bolts, capscrews and washers are visually inspected for evidence or indication of boric acid corrosion. This includes inspection for wastage, missing metal, cracks, pits, boric acid accumulation, corrosion products, and damaged or missing threads. For other than pressure retaining bolted connections, visual examination (VT-2) is conducted without the removal of insulation by examining the accessible and exposed surfaces and joints of the insulation. In accordance with ASME Section XI, essentially vertical surfaces of insulation are examined at the lowest elevation where leakage may be detectable, and essentially horizontal surfaces of insulation are examined at each insulation joint. When examining insulated components, the examination of the surrounding area (including floor areas or equipment surfaces located underneath the components) for evidence of leakage, or other areas to which such leakage may be channeled, is required. Discoloration or residue on surfaces examined is given particular attention to detect evidence of boric acid accumulations from borated reactor coolant leakage.

During the examination, particular attention is given to the insulated areas of components constructed of ferritic steels to detect evidence of boric acid residues whose sources derive from borated reactor coolant, and which may have accumulated during the service period preceding the examination. Any recordable indications identified during these inspections are evaluated by engineering through the corrective action program. The following conditions are considered recordable:

- Leakage from other than locations where leakage is normally expected and collected
- Inoperative leakage collection system
- Areas of general corrosion
- Corrosion which appears to have reduced the wall thickness by more than 10% of the component, or which results in a surface transition with less than a 3 to 1 taper. (Rust with no visible thickness (no flaking) is acceptable without further evaluation)
- Any areas of structural distress (i.e., bent hangers, etc.)
- Boric acid residues on ferritic steel (verification of leakage source shall also be recorded)

In accordance with TMI Unit 1 VT-2 leakage examination Procedure SP 1300-6, if leakage occurs at a bolted connection, the corrective measures and evaluations of ASME Section XI as modified by NRC approved relief (RR-00-08), are required to be performed. Leakage at a bolted connection also requires at least one bolt to be removed and be VT-3 examined, per SP 1300-6.

Dissimilar Metal Welds:

Insulated dissimilar metal welded piping connections that contain Alloy 82/182 typically do not have insulation removed for VT-2 examinations. Welds NPS 4 and larger receive a volumetric and surface examination per ASME Section XI. A list of these nozzle to safe-end welds and the type of insulation is provided in the Table contained in the response to Question No. 1. Examinations of these areas are performed in response to Generic Letter 88-05 and are conducted at hot shutdown after a four hour hold time to allow for leakage to propagate through the insulation joints and be observed by direct VT-2 examination.

The two reactor vessel nozzle to safe-end welds (core flood nozzles) are the only components considered inaccessible to direct VT-2 examination. The welds are located in an area between the concrete reactor vessel shield wall and the concrete primary shield wall. This area enclosing the reactor vessel nozzles and connected piping is referred to as the "sand plug" area. The sand plug area is only accessible from the refuel canal floor. The normal technique for viewing potential leakage in this area is to visually inspect along the piping as it passes though the D-ring wall. The piping is insulated, so the examiner looks for evidence of boric acid at the insulation joints. The reactor vessel nozzle to safe-end weld insulation is not removed. These welds are examined internally by Ultrasonic Testing (UT) as part of the 10-year Inservice Inspection Program requirements.

Reactor Vessel Penetrations:

TMI Unit 1 reactor vessel penetrations in the lower head are scheduled for visual examination in the Fall 2003 refueling outage. A bare metal visual inspection of the TMI Unit 1 upper vessel head penetrations was performed in the Fall of 2001, prior to startup of the current operating cycle. The upper head is scheduled to be replaced in the Fall of 2003.

In general, TMI Unit 1 Class 1, 2, and 3 RCPB components containing borated water, if insulated, have removable, stainless steel, reflective style insulation. Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," guidance is utilized for review of changes to thermal insulation installed on the primary coolant system piping and components. The panels around the reactor vessel head are removable. The insulation panels around the pressurizer heaters are not easily removed. Relief Request RR-00-09 allows this insulation to remain in place due to personnel dose exposure and the potential damage to the heaters associated with the insulation removal.

TMI Unit 1 VT-2 leakage examination Procedure SP 1300-6 specifies that in the event boric acid residues are detected, the insulation shall be removed from the components to the extent necessary to permit visual examination of the surfaces wetted by reactor coolant leakage, in order to detect evidence of corrosion. This procedure also specifies that for a recordable indication, the specific location of the leak shall be determined by the removal of any insulation, which interferes with the determination of the leakage source. The leakage source, when determined, shall be identified and its specific location defined and documented by measured distances from adjacent welds, components, etc.

NRC Question

3. Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in <u>inaccessible areas</u>. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

Response

For RCPB components, TMI Unit 1 performs inspections to identify any reactor coolant system leakage following a plant shutdown, during any refueling outage and, depending on circumstances, at-power. Typical at-power containment visual inspections of accessible areas for leakage would be initiated at the request of the operations department if unidentified leakage were trending up or if there was an unexpected increase in containment atmosphere gas or particulate levels. For at-power visual inspections, the area of interest would be limited for ALARA considerations. VT-2 certified personnel perform the visual inspections for refueling outages and in most cases for forced and at-power visual inspections. Visual inspections performed during plant shutdowns include the reactor head area, the reactor coolant system and the makeup system, both inside and outside the secondary shields inside the reactor building.

See the response to Question No. 5 for a discussion of the detection methods used for inaccessible areas. TMI Unit 1 does not have any plans to install additional leak detection instrumentation at this time; however, TMI continues to monitor industry activities in this area.

NRC Question

4. Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that was established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,

> ***a. If observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or***

a. If observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or

b. If observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.

Response

TMI Unit 1 utilizes Technical Specification Surveillance Procedure Nos. 1300-6, "VT-2 Leakage Exams, " 1300-6Q, "Leakage Exam for Insulated Bolted Connections," and Administrative Procedure No. ER-AA-335-015, "VT-2 Visual Examination," to control these examinations. These procedures describe the requirements and instructions for the evaluation of any indications found during visual examinations, including those of insulated bolted connections. The procedures provide instructions for the corrective actions that must be implemented in the event the structural integrity of any component or bolted connection is determined to be suspect.

For insulated piping and components, a four-hour hold time at nominal operating pressure or hydrostatic test pressure is required as specified in the implementing procedures. For systems borated for the purpose of controlling reactivity, insulation is removed from pressure retaining bolted connections for visual examination VT-2 unless prior relief has been granted. Where normal access controls allow, pressure retaining nuts, studs, bolts, capscrews and washers are visually inspected for evidence or indication of boric acid corrosion. Inspections are conducted for wastage, missing metal, cracks, pits, boric acid accumulation, corrosion products, and damaged or missing threads. For other than pressure retaining bolted connections, visual examinations are conducted without the removal of insulation by examining the accessible and exposed surfaces and joints of the insulation. Essentially vertical surfaces of insulation are examined at the lowest elevation where leakage may be detectable. Essentially horizontal surfaces of insulation are examined at each insulation joint. When examining insulated components, the examination of the surrounding area (including floor areas or equipment surfaces located underneath the components) for evidence of leakage, or other areas to which such leakage may be channeled, is required. Discoloration or residue on surfaces examined is given particular attention to detect evidence of boric acid accumulations from borated reactor coolant leakage.

During the examination, particular attention is given to the insulated areas of components constructed of ferritic steels to detect evidence of boric acid residues whose sources derive from borated reactor coolant, and which may have accumulated during the service period preceding the examination. Any recordable indications are evaluated. The following conditions are considered recordable:

- Leakage from other than locations where leakage is normally expected and collected
- Inoperative leakage collection system

- Areas of general corrosion
- Corrosion which appears to have reduced the wall thickness by more than 10% of the component, or which results in a surface transition with less than a 3 to 1 taper. (Rust with no visible thickness (no flaking) is acceptable without further evaluation)
- Any areas of structural distress (i.e., bent hangers, etc.)
- Boric acid residues on ferritic steel (verification of leakage source shall also be recorded)

If leakage occurs at a bolted connection, the corrective measures and the evaluations of ASME Section XI (as modified by NRC approved reliefs) are performed. Components with local areas of general corrosion that reduce the wall thickness by more than 10% are required to be evaluated by plant engineering. This evaluation will determine whether the component may be acceptable for continued service or whether repair or replacement is required.

TMI Unit 1 uses the corrective action process, specifically the condition report, to determine the acceptability for continued operation when leakage is identified. Within the condition report, operability is considered and may require a formal evaluation using procedure LS-AA-105, "Operability Determinations." If the operability assessment determines that the leakage is acceptable, inspection/monitoring actions may be established and tracked using LS-AA-105-1001, "Supporting Operability Documentation." For leakage that is determined to be unacceptable repair or replacement is required.

NRC Question

5. Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response

As explained in the response to Question No. 1, Item 2, the insulation at the bottom of the reactor vessel lower head stands off of the head surface by approximately six inches at the very bottom of the lower head curvature. Visual examinations of the TMI Unit 1 reactor vessel lower head, scheduled for the Fall of 2003, may require removing insulation to view the metal surface. Any evidence of leakage from lower head penetrations will be visible.

Reactor coolant water/condensed vapor and boric acid deposits from a lower penetration leak would collect on the inner surface of the flat, horizontal insulation panels and the supporting structural carbon steel. The incore penetration instrument tubes, which connect to the Alloy 600 nozzles, are stainless steel and have an increased resistance to boric acid corrosion/wastage. If boric acid were to leak through the seams or opening of the insulation, there are no pressure retaining components beneath the reactor vessel that could be affected.

Plant Technical Specifications require monitoring leakage (including primary-to-secondary leakage) from the reactor coolant system (RCS) and the makeup and purification system. This differs from Standard Technical Specifications in that the makeup and purification system is included in the scope of monitoring for TMI Unit 1. This monitoring is intended to assure that any reactor coolant leakage does not compromise the safe operation of the facility. When the reactor is critical and above two percent power, two reactor coolant leakage detection systems of different operating principles are required to be in operation for the reactor building in accordance with Technical Specification 3.1.6, with one of the two systems sensitive to radioactivity. These systems include containment radiation monitors, mass balance calculations, reactor building sump level monitoring and reactor building humidity monitors. The primary method used at TMI Unit 1 for quantifying RCS and makeup and purification system leakage is the mass balance calculation.

During power operations, containment radiation monitors, reactor building sump level instruments and reactor building humidity monitors are generally in service continuously. Mass balance calculations are required to be performed daily. These calculations are normally performed over a two-hour period and are calculated by the plant process computer. The accuracy of this calculation allows identification of changes in leakage rates of less than 0.1 gpm.

Plant Technical Specifications also require that if the total reactor coolant leakage rate exceeds ten gpm, the reactor is to be placed into hot shutdown within 24 hours of detection. If unidentified reactor coolant leakage (excluding normal evaporative losses) from the RCS pressure boundary exceeds one gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor is required to be placed into hot shutdown within 24 hours of detection. The unidentified leakage limit of one gpm is established as a quantity that 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.

Unexpected changes in the trend of the measured leak rate are identified and dispositioned in the corrective action program with subsequent investigation and operability evaluation.

If a reactor shutdown is required, the rate of shutdown and the conditions of shutdown are determined by a safety evaluation for each case and justified in writing as soon as practicable. Action to evaluate the safety implications of RCS leakage is initiated within four hours of detection. The nature, as well as the magnitude, of the leak is considered in this evaluation. Maintaining exposure of offsite personnel to radiation within the guidelines of 10 CFR 20, "Standards for Protection Against Radiation," is also required within this evaluation. Other factors that are recommended to be included in this evaluation are;

1. Consideration of leakage effects on operation of associated system instrumentation which is important to safety,
2. Consideration of leakage effects on associated system operation during normal and emergency operations,
3. Consideration of leakage effects on the environmental qualification of surrounding equipment which is important to safety,
4. Consideration of restriction of required personnel access to operate or maintain important to safety equipment under normal and emergency conditions,
5. The potential for identified leak to grow to unsafe magnitudes under continued operation, and
6. The effects of boric acid wastage on carbon steel components, piping, supports, etc. To the extent practical evidence of boric acid leakage is not disturbed until engineering has been notified to evaluate the possibility of carbon steel wastage. RCS leakage should be redirected to minimize carbon steel corrosion.

If reactor shutdown is required, the reactor will not be restarted until the leak is repaired or until the problem is otherwise corrected.

NRC Question

6. Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response

See the response to Question Nos. 3 and 5. The visual examination and evaluation procedures at TMI Unit 1 require that components in the area of a leak be examined and, if necessary, be evaluated.

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NRC Question

7. Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

Response

In 1998, Framatome Technologies completed a study to rank on a relative basis each of the Alloy 600 components for potential failure due to PWSCC. This ranking was performed for each of the B&W Owners Group Plants ("Alloy 600 PWSCC Susceptibility Model, 51-5001951-01, Proprietary, December 16, 1998). However, TMI Unit 1 currently does not make use of any susceptibility models or consequence models to identify or rank the Alloy 600 component locations. The basis for TMI Unit 1 inspections of dissimilar metal welds is ASME Section XI. TMI Unit 1 continues to monitor the status of the susceptibility predictions through the B&W Owners Group.

NRC Question

8. Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

Response

Babcock & Wilcox (B&W), the reactor manufacturer, issued Condition Report No. 6014542 on April 1, 2002. This document raised the concern that, while the incore instrumentation nozzles have been ranked low in susceptibility to PWSCC, additional evaluations have ranked the incore nozzles high in failure consequence. B&W recommended that effective visual inspections of the lower reactor vessel head area (OD) be conducted. They also recommended that development of new NDE techniques and repair processes be substantially accelerated. Framatome ANP also issued a Preliminary Report of Safety Concern (PSC) 4-02, related to the chemical samples taken at Davis-Besse associated with potential IMI nozzle leakage, in a letter to the B&W Owners Group Members, dated October 11, 2002. This letter reiterated the recommendation that based on the current state of qualified non-destructive examination (NDE) techniques and accessibility concerns, a bare metal visual examination of the IMI nozzle/lower reactor vessel head interface area should be performed at the earliest opportunity.

Based on this and other recommendations, TMI Unit 1 is planning to install a camera above the insulation to visually inspect the bare metal around the incore instrumentation nozzle penetrations during the refueling outage currently scheduled for October 2003 (T1R15). TMI Unit 1 is currently evaluating the development and use of new NDE

Based on this and other recommendations, TMI Unit 1 is planning to install a camera above the insulation to visually inspect the bare metal around the incore instrumentation nozzle penetrations during the refueling outage currently scheduled for October 2003 (T1R15). TMI Unit 1 is currently evaluating the development and use of new NDE techniques and repair processes to support the T1R15 outage inspection. No additional specific recommendations have been provided by the reactor vendor.

NRC Question

9. Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.

Response

TMI Unit 1 Technical Specification Limiting Condition for Operation, (LCO), 3.1.6 states that there shall be no pressure boundary leakage, except steam generator tube leakage, through a nonisolable fault in a reactor coolant system component body, pipe wall, or vessel wall. If pressure boundary leakage is detected, the action statements for this LCO requires that the reactor shall be shutdown and a cool-down to cold shutdown initiated within 24 hours of detection. Actions to evaluate the safety implications of reactor coolant system leakage are required to be initiated within four hours of detection.

Compliance with the zero, non-isolable leakage criteria is met by performing Generic Letter 88-05 examinations, conducting inspections and repairs in accordance with ASME Section XI, and 10 CFR 50.55a, "Codes and standards" and by monitoring reactor coolant leakage trends using diverse methods (mass balance calculation, radiation monitors, reactor building sump level accumulation and reactor building humidity monitoring), as required by plant-specific technical specifications. In addition, the unidentified leakage limit of one gpm is established as a quantity that can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a short time, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

10 CFR 50.55a requires that inservice inspection and testing be performed in accordance with 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.

Attachment 3

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For this, the 3RD Inservice Inspection Interval, TMI Unit 1 has implemented the 1995 edition, with 1966 addenda, of ASME Section XI. Paragraph IWA-5250 (b), "Corrective Measures," of this edition states:

> "If boric acid residues are detected on components, the leakage source and the areas of general corrosion shall be located. Components with local areas of general corrosion that reduce wall thickness by more than 10% shall be evaluated to determine whether the component may be acceptable for continued service, or whether repair or replacement is required."

To incorporate these requirements, TMI Unit 1 utilizes Technical Specification Surveillance Procedure Nos. 1300-6, "VT-2 Leakage Exams," 1300-6Q, "Leakage Exam for Insulated Bolted Connections," and Administrative Procedure No. ER-AA-335-015, "VT-2 Visual Examination," to control these examinations and any subsequent corrective actions. Corporate Procedure No. ER-AA-335-015, "VT-2 Visual Examination," Paragraph 4.6.1.4 states:

> *"If boric acid residues are detected on components, **then LOCATE** the leakage source and the areas of general corrosion. **EVALUATE** components with local areas of general corrosion that reduce the wall thickness by more than 10% to determine whether the component may be acceptable for continued service, or whether repair or replacement is required."*

Corporate Procedure No. LS-AA-125, "Corrective Action Program (CAP) Procedure," defines the requirements for condition identification, condition review, investigation, and closeout.