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January 24, 2003
L-03-004

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: **Beaver Valley Power Station, Unit No. 1 and No. 2**
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Reply to Request for Additional Information Regarding
60-Day Response to Bulletin 2002-01

References:

Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, dated March 18, 2002

BVPS 60-day Response to Bulletin 2002-01, L-02-066 dated May 17, 2002

On November 19, 2002, the NRC issued a Request for Additional Information (RAI) regarding the Beaver Valley Power Station (BVPS) 60-day response to Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." This letter provides the FirstEnergy Nuclear Operating Company (FENOC) response for BVPS Units 1 and 2 to the RAI.

There are no new regulatory commitments contained in this submittal. If there are any questions concerning this matter, please contact Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement at 724-682-5284.

I declare under penalty of perjury that the following is true and correct. Executed on January 24, 2003.

Sincerely,

Mark B. Bezilla

Enclosure

A095

c: Mr. D. S. Collins, NRR Project Manager
Mr. D. M. Kern, NRC Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)
Ms. C. O'Clair, Ohio Emergency Management Agency

Enclosure 1

Reply to Request for Additional Information Regarding Beaver Valley Power Station (BVPS) Units 1 and 2 60-Day Response to NRC Bulletin 2002-01 "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

Note that these responses apply for both BVPS Units 1 and 2 because the inspections and maintenance programs are common for both units. Plant specific information for each unit is provided where necessary.

The Request for Information requested a response to the following nine items:

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 attached tables (Table 1-A for Unit 1 and Table 1-B for Unit 2) describe the current status of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB) at BVPS, as of the date of this submittal. Alloy 600 Primary Reactor Coolant Loop locations and Alloy 82/182 welds and connections have been determined from a review of component drawings and vendor recommendations (ref.14: Westinghouse Letter Report MSE-MNA-368 (94) Revision, 1, dated January 1995).

Volumetric Non-destructive Examinations (NDE) of the Control Rod Drive Mechanisms (CRDM) Alloy 600 penetrations and J-groove welds on the reactor pressure vessel closure heads at both Units is scheduled for refueling outage 1R15 at Unit 1 (March 2003) and 2R10 at Unit 2 (September 2003).

The technical basis for the requested information contained within Tables 1-A and 1-B is derived from the ASME Section XI (1989 Edition for both BVPS Units) Inservice Inspection (ISI) requirements as defined within each unit's 10-year Inservice Inspection Program Plan, as previously docketed. Industry Operating Experience is routinely reviewed for lessons learned, and any actions emanating from these issues are incorporated via the BVPS Corrective Action program.

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:

[Note: Please refer to Tables 1-A and 1-B for specific location insulation considerations.]

The reactor coolant pressure boundary (RCPB) piping at both BVPS Unit 1 and Unit 2 is stainless steel. The reactor pressure vessel (RPV), Pressurizer, and primary portions of the Steam Generators are all clad carbon steel vessels that are susceptible to corrosion from high concentrations of boric acid.

Mirror insulation is removed from the reactor pressure vessel closure head to the degree necessary to perform the required visual inspections for evidence of leakage from the penetrations, or corrosion damage from other leakage sources such as conoseals, canopy seal welds, or head vent mechanical joints.

Mirror insulation is not removed from the RPV bottom head for visual inspection of the bottom head for VT-2 leakage examinations, neither at the beginning of each outage nor during the system pressure test at nominal operating pressure and temperature following each refueling outage.

Mirror insulation on the Pressurizer is removed when scheduled Section XI volumetric or surface examinations are performed. This applies to both vessel welds and dissimilar metal welds, when applicable. Insulation is not removed for VT-2 examinations, although the bottom insulation seams and heater penetration convection covers of the pressurizers at both units are examined for evidence of leakage at the beginning of each outage and during the system pressure test at nominal pressure and temperature following each refueling outage.

Mirror Insulation is removed on the bottom of the primary portions of the Steam Generators during each refueling outage to allow access for the tubing examinations. Additionally, mirror insulation on the Steam Generators is removed when scheduled Section XI volumetric or surface examinations are performed. Insulation is not removed for VT-2 examinations, although the primary manway insulation covers and bottom head insulation at both units are examined for evidence of leakage at the beginning of each outage and during the system pressure test at nominal pressure and temperature following each refueling outage.

RCPB piping insulation is removed for access to perform UT (volumetric) and PT (surface) examinations on the Alloy 82/182 welds as scheduled by the applicable BVPS 10-year ISI Examination Plan for each unit.

Except for the Reactor Pressure Vessel CRDM head penetrations, insulation is not routinely removed for the purpose of performing VT-2 leakage examinations.
Other attributes of the BVPS leakage inspection requirements include:

- The low point of any insulated borated piping system is carefully examined for evidence of leakage at insulation seams, since leakage from vertical portions of the piping would accumulate at this point.
- Insulated pressure boundary components are examined at the horizontal or vertical insulation seams for bulging, or at the lowest point to detect pressure boundary leakage in accordance with ASME Section XI requirements. Areas adjacent to and below components are examined for evidence of leakage.
- Evidence of leakage is identified and evaluated, and site procedure(s) require the source of leakage to be determined when observed from insulated and inaccessible components. Additional detail is provided in the response to RAI Item #9.

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:

Boric acid walkdown inspections are performed during refueling outages to identify evidence of leakage following a cycle of power operation. The inspections identify the need for corrective maintenance at the onset of an outage. Additionally, ASME Code required system pressure (leakage) examinations at nominal Reactor Coolant System (RCS) operating temperature and pressure are performed at the end of refueling outages prior to power operation.

The boric acid walkdown inspections conducted by VT-2 qualified personnel are performed in containment at the beginning of each refueling outage on all floor elevations and accessible locations to specifically identify, photograph and report leakage or deposits of boric acid. As initially inaccessible components become accessible during refueling outage activities, additional examinations are performed by VT-2 qualified personnel in conjunction with ISI NDE and visual examinations, walkdowns, and bolted connection examinations.

During power operation, there are areas inside the Reactor Containment Building (RCB) that are inaccessible due to physical and radiation constraints. However, during refueling outages, areas inside containment receive a walkdown per Site NDE procedure VT-510, "Visual Examinations for Boric Acid." This walkdown uses the industry and code standard for insulated components, noted in ASME Section XI, Paragraph IWA-5242. This paragraph states that insulation shall be removed for pressure retaining bolted connections, and for other components, visual examinations can be performed without insulation removal by examining the accessible and exposed surfaces and joints of the insulation.

At the end of each refueling outage, VT-2 qualified personnel perform a system leakage examination on the RCS pressure boundary and other pressurized containment systems when the RCS reaches nominal operating temperature and pressure. This examination activity also serves as the post-maintenance system leakage examination on items that were repaired, replaced, or disassembled during the refueling outage.

The only known RCS pressure boundary component exterior surface that we have determined to be inaccessible is the vertical, insulated portion of the reactor vessel between the hot and cold leg nozzles and the RPV bottom head (the vertical portion of the vessel surrounded by the neutron shield tank). This portion of the RPV was insulated with mirror insulation prior to being lowered onto the vessel nozzle supports. There are no Alloy 600 or Alloy 82/182 materials within this segment of the reactor vessel. Any pressure boundary leakage affecting this inaccessible vertical portion of the reactor vessel would be evident as staining or boric acid accumulations under the reactor vessel or at the bottom head insulation seams. The longitudinal and circumferential welds in this vessel region are volumetrically inspected on a ten-year interval from the vessel inside surface as part of the RPV 10-year ISI inspection plan.

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 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:

FENOC procedure NOP-ER-2001, "Boric Acid Corrosion Control Program", provides specific direction for the evaluations that would be conducted upon discovery of leakage at bolted connections. The procedure requires consideration of the bolting material, field verification of the bolting material, and monitoring frequency in the event a component cannot be repaired. Beaver Valley Engineering Standard ES-M-031, "Leakage and Corrosion Evaluations" provides the technical basis for such engineering evaluations based on the materials, fluid chemistry, leak rate, operating conditions, corrosion rates and other guidance derived from EPRI Publication TR-104748, "Boric Acid Corrosion Guidebook".

Boric acid leakage process flow charts were developed for inclusion in 1/2-ADM-2112, "Boric Acid Corrosion Control," and are attached for your information (see Figures 4A and 4B). These flow charts clearly identify the process and the actions to be taken.

As noted in section 5.a of the 60-day response to Bulletin 2002-01 (Ref.10), BVPS has formal leakage criteria for operational acceptance of both Identified and Unidentified Leakage. The following excerpt is provided for reference:

The BVPS Unit 1 and Unit 2 Technical Specifications provide a formal leakage acceptance criteria for Reactor Coolant System (RCS) Identified Leakage and RCS Unidentified Leakage. The RCS leakage rate is calculated by the performance of Operations Surveillance Test (OST) 1OST-6.2 "*Reactor Coolant System Water Inventory Balance*" at Unit 1 and 2OST-6.2 "*Reactor Coolant System Water Inventory Balance*" at Unit 2. The following acceptance criteria have been established for these procedures:

- Identified Leakage greater than 10 gpm or Unidentified Leakage greater than 1.0 gpm (refer to Technical Specification 3.4.6.2).
- Unidentified Leakage between 0.5 gpm and 1.0 gpm; however, if leakage has not been increasing since the last performance of the procedure, then re-perform test every 72 hours as scheduled.

- Unidentified Leakage between 0.5 gpm and 1.0 gpm and leakage has been increasing since the last performance of the procedure, then consult Station Management for a possible Containment entry for leak identification.
- Unidentified Leakage less than 0.5 gpm, then re-perform test every 72 hours, as scheduled.

Based on recent industry information, enhancements have been incorporated in the program to include additional monitoring considerations in the event of increasing RCS unidentified leak rates. Surveillance tests 1OST-6.2 and 2OST-6.2 have been revised, such that the OST is re-performed if the RCS Unidentified Leak Rate increases by greater than or equal to 0.2 gpm when compared to the last performance of the OST, or is equal to 0.5 gpm. If the re-performance also yields the same RCS Unidentified Leak Rate or greater, then the following actions are specified:

- Review Radiation Monitors for upward trends.
- Review other Control Room Indications to identify leakage.
- Walkdown systems as required to identify the source of the leakage (this would include the RCS).
- Inform Station Management of the increase in unidentified leakage.

<u>Response to 4.a:</u>

If the observed leakage is determined to be acceptable for continued operation, the actions and monitoring for trending and evaluating are described in our procedure 1/2-ADM-2112. A summary of the process is provided in the attached flowcharts (see Figures 4A and 4B).

<u>Response to 4.b:</u>

If the observed leakage is determined to be unacceptable for continued operation, the leak source would be removed from service. If the leakage can not be isolated or removed from service, corrective actions would be undertaken to repair the condition.

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.***

Response:

Visual examination of the area beneath the Reactor Vessel is performed as part of the Boric Acid Walkdown program during refueling outages. A visual examination of the exterior of the insulation covering the bottom of the Reactor Vessel is performed to identify any evidence of boric acid leakage. Additionally, the area around each Bottom Mounted Instrumentation (BMI) penetration through the insulation is examined for evidence of leakage.

In a recent Maintenance Outage at Beaver Valley Unit 1 during November 2002, an opportunity to inspect the area between the insulation and the bottom of the Reactor Vessel was available. As a result, an inspection using a 6mm video probe was performed under the insulation, to determine if any accumulations of boric acid crystals or evidence of boric acid residue were present at or near the BMIs. Although each BMI was not inspected, the inspection at the bottom apex of the vessel and insulation package revealed no evidence of boric acid leakage from a penetration. (Ref. 13)

If evidence of boric acid leakage was to be observed during the inspection of this area, our Boric Acid Program (NOP-ER-2001) would require the source of the leakage to be determined and repaired. Additionally, our procedure for the evaluation of boric acid leakage (ES-M-031) would require the identification of any carbon steel or other susceptible materials that were in the leakage flowpath, to allow for the assessment of the material condition of all materials in the flowpath.

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:

As stated in the BVPS 15 and 60-Day Response to Bulletin 2002-01 (Ref. 7 and 10), due to the sealed nature of the BVPS containment buildings (no external ventilation), the reactor building radiation monitoring system is extremely sensitive to low levels of leakage from the RCS pressure boundary. The data from this system is monitored, recorded and trended by Health Physics personnel and anomalies reported to the Control Room staff and Health Physics Supervision. Upon detection of a reactor coolant pressure boundary leak, a containment entry would be initiated to visually locate the source of the leak. Since the detection of a RCS pressure boundary leak by the reactor building radiation monitoring system would be at its earliest stages, impact to components (targets) within the leakage path would be minimal and addressed by the Boric Acid Corrosion Control (BACC) Program.

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:

The BVPS existing BACC program and ASME Section XI pressure testing program do not make use of or take into account consequence models. Our programs are based on susceptibility as defined in ASME Section XI and industry operating experience.

BVPS is in the process of implementing a Risk-Informed Inservice Inspection (RI-ISI) program using topical report WCAP-14572, Revision 1-NP-A, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report" for the ASME Class 1 and Class 2 piping systems. The implementation process of the WCAP requires the evaluation of all locations within the scope of the program for potential active failure mechanisms. In the development of the Beaver Valley Unit 1 and Unit 2 RI-ISI programs, the Alloy 600/82/182 piping locations were all considered to be susceptible to Primary Water Stress Corrosion Cracking (PWSCC). The susceptibility to known failure mechanisms was a basis used in the RI-ISI Program for determining the examination locations within high safety significant segments.

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:

Westinghouse, the reactor vendor for BVPS, has provided no recommendations to date regarding visual inspections of Alloy 600/82/182 materials. This was confirmed by a Westinghouse review of its databases and applicable communications made with owners of Westinghouse reactors. This was documented in Section 4.1 of Westinghouse Owners Group Letter, WOG-02-223, dated December 13, 2002. Since no recommendations were made, no actions or plans could be undertaken.

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:

FENOC Nuclear Operating Procedure (NOP) NOP-ER-2001, "Boric Acid Corrosion Control Program" governs BVPS Boric Acid Corrosion Control Program. This procedure was developed using Operating Experience from Davis-Besse to ensure that boric acid corrosion does not degrade the Reactor Coolant Pressure Boundary. This procedure requires evidence of boric acid leakage to be evaluated using the FENOC Corrective Action Process (NOP-LP-2001).

Additional BVPS site-specific requirements are contained in procedure 1/2-ADM-2112, "Boric Acid Corrosion Control" including process flow charts (see Figures 4A and 4B). An additional implementing NDE procedure, NDE-VT-510, "Visual Examination for Boric Acid", provides more detailed instructions, maps, and specific component lists of potential leak locations (Bolted connections, Alloy 600 Materials or Alloy 82/182 welds).

BVPS procedure NDE-VT-502, "Leakage Examination Requirements" specifies instructions for performing visual examinations of pressure retaining components for evidence of leakage. This procedure requires the removal of insulation from all Class 1 and 2 bolted connections in systems borated for the purpose of controlling reactivity. This procedure provides the following conditions that would be deemed unsatisfactory:

- Components with local areas of general corrosion that reduce the wall thickness by more than 10%
- Any leakage whose source is from a pressure retaining weld
- Any leakage whose source is through the base material
- Leakage from insulated or inaccessible components that will require location of the leakage source
- Discoloration or accumulated residues on surfaces of components, insulation or floor areas that may be evidence of borated water leakage

Conditions that are unsatisfactory (i.e., fail to meet NDE procedure acceptance standards) require generation of a Condition Report to place the unsatisfactory condition into the BVPS site corrective action program. Subsequent actions and reviews mandated by the corrective action program provide numerous reviews of both the cause analysis and any corrective actions generated to remediate the observed unsatisfactory condition(s).

Additionally, there are also specific instructions within both NOP-ER-2001 and 1/2-ADM-2112 with respect to the evaluation of boric acid leakage, leak path evaluation, wastage, monitoring, replacement and cleaning/restoration of components affected by boric acid leakage. The procedural requirements currently in place are sufficient to ensure that leakage of borated fluids will be identified through periodic inspection activities performed by qualified individuals.

Both NOP-ER-2001 and 1/2-ADM-2112 were developed in 2002 to address the issues identified through FENOC program evaluations and industry response to NRC Bulletin 2002-01, "Reactor Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity" and INPO SER 2-02, "Undetected Leak in Control Rod Drive Mechanism Nozzle and Degradation of Reactor Pressure Vessel Head".

Table 1-A
BVPS UNIT 1

RC Pressure Boundary Alloy 600 Base Material or Alloy 82/182 Weld Material Locations						
Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency (minimum)	Degree of Insulation Removal/Insulation Type	Corrective Action
Reactor Pressure Vessel Closure Head CRDM, Instrumentation and Head Vent Penetrations	Visual, VT-2	Visual, VT-2 (ASME XI, 1989 Edition)	100% of periphery for each nozzle penetration.	18 months (fuel cycle) – initial examination in September 2001.	Tiered, mirror insulation is removed to the extent needed to obtain visual examination coverage using remote visual inspection equipment.	Evidence of boric acid leakage from penetrations results in additional volumetric NDE to determine the source of leakage.
Reactor Pressure Vessel Bottom Head Incore Instrumentation Penetrations	Visual, VT-2	Visual, VT-2 (ASME XI, 1989 Edition)	VT-2 inspection for evidence of leakage from the insulation seam(s) at the bottom of the reactor vessel.	18 months (fuel cycle) as part of the ASME Section XI pressure testing program and the BACC Program.	Permanent mirror insulation has not been removed for routine VT-2 examinations. In 1M02 (November 2002), several penetration convection flashings were removed to allow a small video probe to be inserted under the insulation.	Evidence of boric acid leakage from insulation seam results in the requirement to determine the source of leakage.
Reactor Pressure Vessel Closure Head CRDM Housing and vent piping welds	Visual, VT-2 PT of 10% of peripheral CRDM housings	Visual, VT-2, Surface (PT) (ASME XI, 1989 Edition)	VT-2 inspection for evidence of leakage; PT -360 degrees of selected welds	VT-2 -18 months (fuel cycle) as part of the ASME Section XI pressure testing program. PT – every 10 years	Not insulated – accessible through shroud ventilation openings for both VT-2 and PT examinations.	VT -Evidence of leakage from pressure boundary (weld) is unacceptable. PT – indications exceeding IWB-3523 acceptance standard are unacceptable.
Steam Generator heat transfer tubing	Bobbin Coil Probe Plus Point Probe	Eddy Current Level II, IIA, III Qualified Data Analyst (ASME XI, 1989 Edition)	Bobbin Coil - 100% full length Plus Point - Area of Interest	18 months (fuel cycle)	N/A	Repair limit for degradation that can be sized with qualified NDE techniques is 40% through wall. All other crack-like degradation is repaired upon detection except for degradation bounded by Generic Letter 95-05.
Steam Generator heat transfer tubing plugs	All Alloy 600 plug material has been replaced by Alloy 690 material.					

Table 1-B
BVPS UNIT 2

RC Pressure Boundary Alloy 600 Base Material or Alloy 82/182 Weld Material Locations						
Component	Inspection Techniques	Personnel Qualifications	Extent of Coverage	Frequency	Degree of Insulation Removal/Insulation Type	Corrective Action
Reactor Pressure Vessel Closure Head CRDM, Instrumentation and Head Vent Penetrations	Visual, VT-2	Visual, VT-2 (ASME XI, 1989 Edition)	100% of periphery for each nozzle penetration.	18 months (fuel cycle) – initial examination in February 2002.	Tiered, mirror insulation is removed to the extent needed to obtain visual examination coverage using remote visual inspection equipment.	Evidence of boric acid leakage from penetrations results in additional volumetric NDE to determine the source of leakage.
Reactor Pressure Vessel Bottom Head Incore Instrumentation Penetrations	Visual, VT-2	Visual, VT-2 (ASME XI, 1989 Edition)	VT-2 inspection for evidence of leakage from the insulation seam(s) at the bottom of the reactor vessel.	18 months (fuel cycle) as part of the ASME Section XI pressure testing program and the BACC Program.	Permanent mirror insulation has not been removed for routine VT-2 examinations.	Evidence of boric acid leakage from insulation seam results in the requirement to determine the source of leakage.
Reactor Pressure Vessel Closure Head CRDM Housing and vent piping welds	Visual, VT-2 PT (Surface) of 10% of peripheral CRDM housings	Visual, VT-2, PT (ASME XI, 1989 Edition)	VT-2 inspection for evidence of leakage; PT -360 degrees of selected welds	VT-2 -18 months (fuel cycle) as part of the ASME Section XI pressure testing program. PT – every 10 years	Not insulated – accessible through shroud ventilation openings for both VT-2 and PT examinations.	VT -Evidence of leakage from pressure boundary (weld) is unacceptable. PT – indications exceeding IWB-3523 acceptance standard are unacceptable.
Reactor Pressure Vessel Hot Leg & Cold Leg Nozzle vessel-to-safe-end welds	Visual, VT-2 UT (Volumetric), PT (Surface)	Visual, VT-2, UT – ASME XI-1995 Edition Appendix VIII) PT – ASME XI, 1989 Edition	VT-2 inspection for evidence of leakage; UT, PT – Per Table IWB-2500 (Category B-F)	Per Table IWB-2500-1 (Category B-F)	Mirror insulation not removed for VT-2 examinations. Mirror insulation removed for UT and PT examinations.	VT-Evidence of leakage from pressure boundary (weld) is unacceptable UT, PT – indications exceeding IWB-3514 are unacceptable.

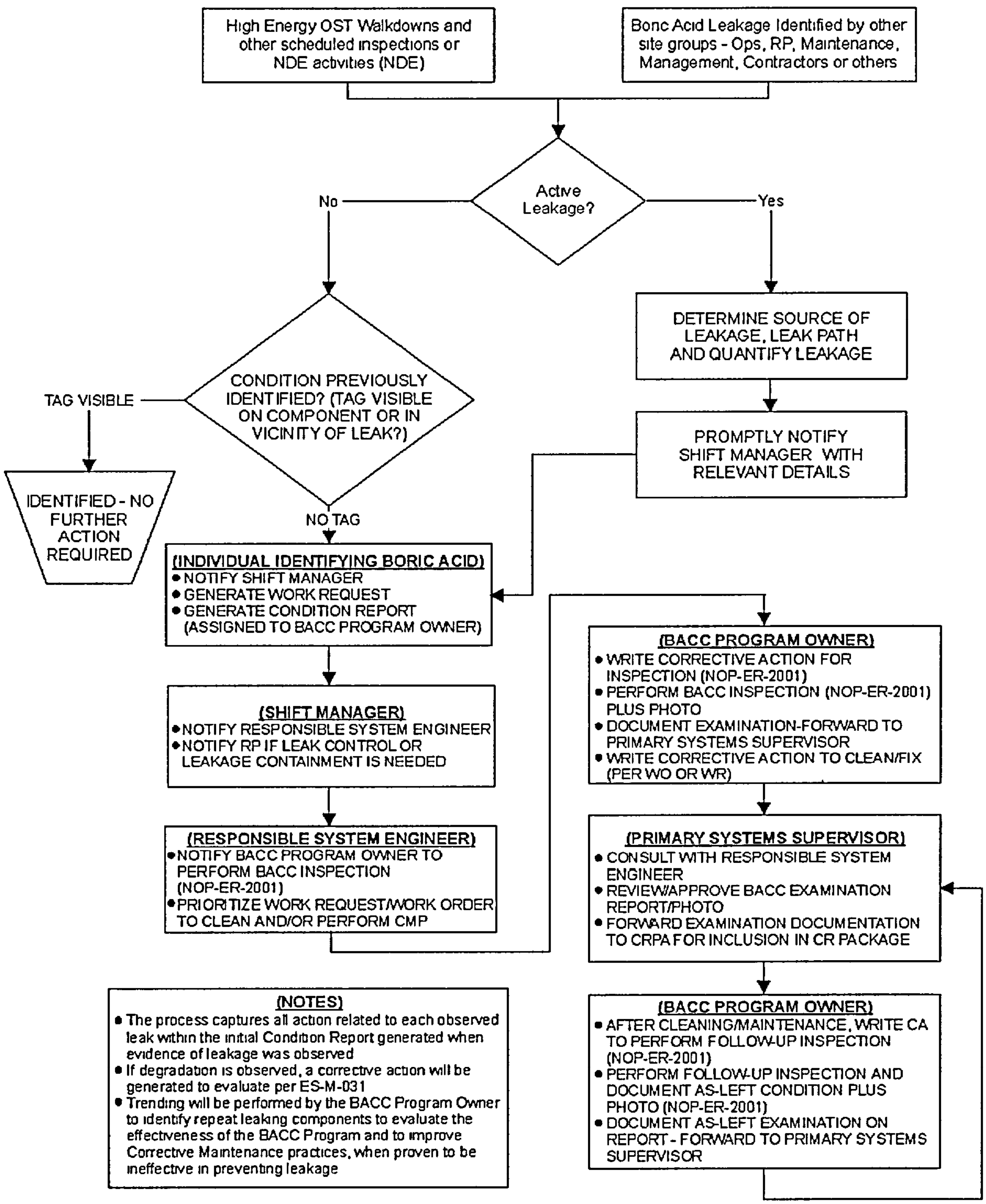
Table 1-B (continued)
BVPS UNIT 2

RC Pressure Boundary Alloy 600 Base Material or Alloy 82/182 Weld Material Locations						
Pressurizer Vessel Surge Nozzle-to-safe-end weld, spray nozzle-to-safe-end weld, Code Safety and PORV nozzle-to-safe-end welds	Visual, VT-2 UT (Volumetric), PT (Surface)	Visual, VT-2, UT – ASME XI-1995 Edition Appendix VIII) PT – ASME XI, 1989 Edition	VT-2 inspection for evidence of leakage; UT, PT – Per Table IWB-2500 (Category B-F)	Per Table IWB-2500-1 (Category B-F)	Mirror insulation not removed for VT-2 examinations. Mirror insulation removed for UT and PT examinations when scheduled.	VT-Evidence of leakage from pressure boundary (weld) is unacceptable UT, PT – indications exceeding IWB-3514 are unacceptable.
Steam Generator heat transfer tubing	Bobbin Coil Probe Plus Point Probe	Eddy Current Level II, IIA, III Qualified Data Analyst (ASME XI, 1989 Edition)	Bobbin Coil - 100% full length Plus Point - Area of Interest	18 months (fuel cycle)	N/A	Repair limit for degradation that can be sized with qualified NDE techniques is 40% through wall. All other crack-like degradation is repaired upon detection except for degradation bounded by Generic Letter 95-05.
Steam Generator heat transfer tubing plugs	Visual	ANSI 45.2.6	Tack weld on 18 Westinghouse Plug-in-Plugs (PIPs)	18 months (fuel cycle)	N/A	Any crack-like indication in the tack weld would require removal of the PIP and plug. Replace with Alloy 690 material.
	Except for the 18 locations with PIPs, all plug material is Alloy 690.					

Enclosure 1 (continued)

Figure 4A

PROCESS FLOW CHART FOR EXAMINATIONS OUTSIDE RCB (Attachment A of 1/2-ADM-2112)

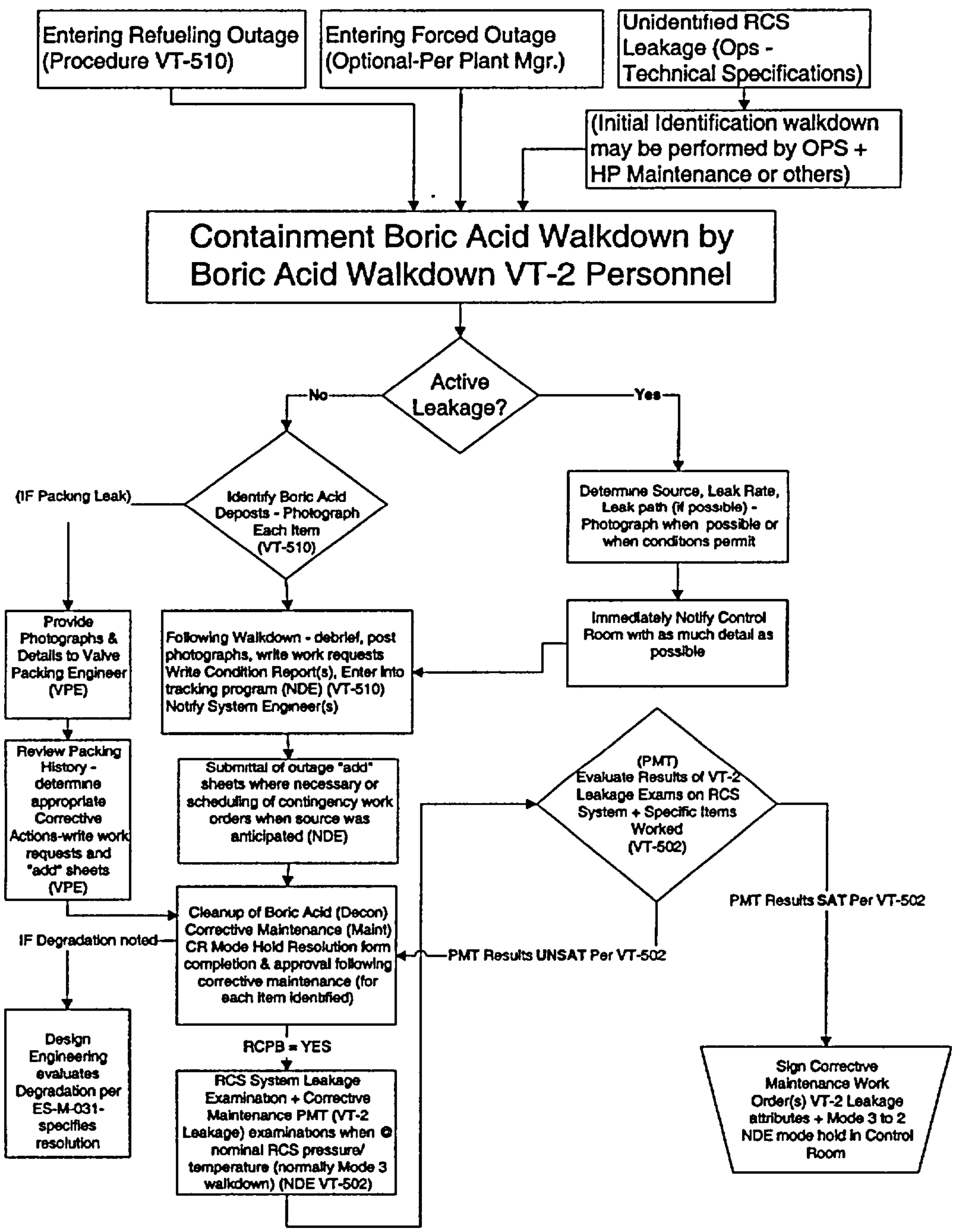


Enclosure 1 (continued)

Figure 4B

PROCESS FLOW CHART FOR EXAMINATIONS INSIDE RCB (Attachment B of 1/2-ADM-2112)

Current Beaver Valley Containment Boric Acid Leakage Examinations & Resolution Process
(Subject to Enhancements as a result of ongoing Corrective Actions)



References:

1) **Bulletin 2001-01**: "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated August 3, 2001.
2) **Bulletin 2001-01 BVPS Reply,** L-01-114 dated August 31, 2001.
3) **Bulletin 2001-01 Reply for 1R14,** L-01-136 dated October 31, 2001.
4) **Bulletin 2001-01 Responses for BVPS,** NRC evaluation dated December 11, 2001.
5) **Bulletin 2001-01 Reply for 2R09,** L-02-021 dated March 28, 2002.

6) **Bulletin 2002-01**: "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2001.
7) **Bulletin 2002-01 BVPS 15-day Reply,** L-02-032 dated April 1, 2002.
8) **Bulletin 2002-01 BVPS Supplemental Reply,** L-02-040 dated April 19, 2002.
9) **Bulletin 2002-01 BVPS Reply to RAI (7 questions),** L-02-054 dated May 10, 2002.
10) **Bulletin 2002-01 BVPS 60-day Reply,** L-02-055 dated May 17, 2002.
11) **Bulletin 2002-01 NRC Review** of BVPS 15-day, Supplemental, and RAI Replies, NRC letter dated November 18, 2002
12) **Bulletin 2002-01 Request for Additional Information** dated November 19, 2002

13) **Bulletin 2001-01 and Bulletin 2002-01, BVPS Updated Information,** L-02-121 dated January 8, 2003

14) **Westinghouse Letter Report** MSE-MNA-368 (94) Revision 1, dated January 1995