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Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors

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Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors

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LIST OF ACRONYMS

ALARA	As Low As is Reasonably Achievable
ANO	Arkansas Nuclear One
ASME	American Society of Mechanical Engineers
AOV	Air-Operated Valve
B&WOG	Babcock and Wilcox Owners Group
BAC	Boric Acid Corrosion
BACC	Boric Acid Corrosion Control
BACCP	Boric Acid Corrosion Control Program
BAT	Boric Acid Tank
BATP	Boric Acid Transfer Pump
BIT	Boron Injection Tank
BRS	Boron Recycle System
CAR	Containment Air Recirculation
CCP	Centrifugal Charging Pump
CE	Combustion Engineering
CEDM	Control Element Drive Mechanism
CFCU	Containment Fan Cooler Unit
CRDM	Control Rod Drive Mechanism
CVCS	Chemical and Volume Control System
DPT	Dye Penetrant Testing
EPRI	Electric Power Research Institute
ET	Eddy Current Testing
GL	Generic Letter
HP	Health Physics
HPI	High Pressure Injection
HPI/MU	High Pressure Injection/Make-up
I&C	Instrumentation and Control
ICI	In-Core Instrumentation
ID	Inner Diameter
IGSCC	Intergranular Stress Corrosion Cracking

LIST OF ACRONYMS (cont.)

INPO	Institute of Nuclear Power Operations
ISI	In-Service Inspection
MNSA	Mechanical Nozzle Seal Assembly
MOV	Motor-Operated Valve
MRP	Materials Reliability Program
MSC	Materials Subcomittee
MT	Magnetic Particle Testing
MU	Make-up
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PP	Partial Penetration
PT	Penetrant Testing
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMCS	Reactor Makeup Control System
RMWST	Reactor Makeup Water Storage Tank
RP	Radiation Protection
RPV	Reactor Pressure Vessel
RT	Radiographic Testing
RV	Reactor Vessel
RVH	Reactor Vessel Head
RTE	Resistance Temperature Element
SD	Shutdown Decay
SDC	Shutdown Decay Cooling
SE	Safe End

LIST OF ACRONYMS (cont.)

SER	Safety Evaluation Report
SG	Steam Generator
SI	Safety Injection
SIS	Safety Injection System
SOER	Significant Operating Event Report
SS	Stainless Steel
TGSCC	Transgranular Stress Corrosion Cracking
TS	Tubesheet
UT	Ultrasonic Testing
VCT	Volume Control Tank
WOG	Westinghouse Owners Group

1 INTRODUCTION

The purpose of this document is to provide generic guidance to serve as a basis for the operating Pressurized Water Reactor (PWR) power plants in developing plant-specific Boric Acid Corrosion Control Programs (BACCPs) and procedures. Additionally, this document is to provide a basic understanding of boric acid issues currently being addressed in the industry. The plant-specific BACCP needs to be designed to identify, evaluate, and correct small boric acid leaks in the primary systems that could cause corrosion damage on Reactor Coolant Pressure Boundary (RCPB) components or other auxiliary system components in PWRs. A small boric acid leak in the primary system is defined as a leak that is smaller than the plant technical specification limits. The subject generic guidance is intended to identify potential enhancements to previous utility responses to Nuclear Regulatory Commission (NRC) Generic Letter GL 88-05 (Ref. 1).

This report is written as part of the industry initiative to address the cracking and leakage of Alloy 600 reactor vessel head penetration tubes, also commonly referred to as Control Rod Drive Mechanism (CRDM) tubes (Refs. 2 and 3), due to Primary Water Stress Corrosion Cracking (PWSCC) and the resulting reactor vessel head wastage such as was found at the Davis-Besse nuclear plant (Refs. 4 and 5). In addition to this guideline, the NRC, the Institute of Nuclear Power Operations (INPO) (Ref. 6), and the industry are evaluating what constitutes an adequate program. This report offers essential elements of an effective BACCP for a typical PWR plant that could be utilized by the utilities while reviewing the adequacy of their plant-specific programs.

The BACCP and the inspection procedure should bring various aspects of different related plant programs together to ensure that all boric acid leakage, as well as any consequential or collateral damage, is identified, evaluated, cleaned, and/or dispositioned in a timely manner to maintain component integrity. The data from system pressure testing and visual inspections performed under the rules of American Society of Mechanical Engineers (ASME) Section XI should be accounted for in the identification of boric acid leaks as part of the BACCP procedures. A site-specific BACCP should also ensure that all personnel involved with boric acid activities are adequately trained and knowledgeable.

In addition, other potential leak locations included in the program are: i) PWSCC susceptible Alloy 600 and Alloy 82/182 weld locations in borated systems, ii) other potentially sensitized, stainless steel heat affected zone locations (such as shop and field repair and modification locations), and iii) locations covered under the existing ASME Section XI programs that perform many of the same functions. For example, the 1989 edition of the ASME Code Section XI required that bolted components in systems borated for the purpose of reactivity control have their insulation removed and the bolted connections inspected for degradation. This program is intended to recognize the potential impact of boric acid attack on carbon steel components resulting from leakage.

While not specifically required by the NRC as a part of GL 88-05, many other opportunities exist during plant activities to detect and identify small leaks that occur during the normal plant operation. The information from these plant-specific opportunities can be valuable and should be utilized to benefit a comprehensive BACCP. This guidance covers the issues involving the recent CRDM penetration tube cracking in US PWRs, leakage and wastage issues that occurred at Davis-Besse nuclear plant, and the inspection requirements addressed in recent NRC Bulletins issued in 2001 and 2002. Background information relating to the relevant industry service experience is provided in Section 2.

The guidance provided here is intended to identify attributes to be incorporated as appropriate, by the utilities, when developing or assessing a plant-specific BACCP. This document is not intended to be an inspection procedure for implementation at any PWR plant or to infer that all the attributes identified in this document are recommended to be implemented at each nuclear site. It is the responsibility of each utility to review their BACCP for effectiveness and incorporate those attributes that will improve their program.

The guidance for inspection of the reactor vessel head and CRDM penetration tubes is contained in the Electric Power Research Institute (EPRI) Materials Reliability Project (MRP) document EPRI-MRP-75, (Ref. 7, the draft MRP-75 is under revision to address the NRC comments at the time of this report). Until MRP-75 document is finalized and approved by the NRC, the interim reactor vessel head inspections will be performed as per the requirements of the NRC Order EA-03-009, (Ref. 8) "Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," dated February 11, 2003.

This guidance should not be considered as final. Revisions to this document are anticipated based on the results of INPO audits currently being conducted and additional considerations from service experience from stations.

2 BACKGROUND

Since the late 1970s, numerous boric acid leaks have been reported in the primary and other borated systems in Pressurized Water Reactors (PWRs). These events have often resulted in the corrosion and wastage of the Reactor Coolant Pressure Boundary (RCPB) components or subcomponents, or degradation of other safety system components. A summary listing of industry documented leaks from Nuclear Regulatory Commission (NRC) bulletins is provided in Attachment 5.1.

On March 17, 1988, the NRC issued Generic Letter GL 88-05 (Ref. 1) "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." GL 88-05 identified the key elements to ensure that there is an extremely low probability of abnormal leakage, rapidly propagating failure, or a gross rupture of RCPB components from the boric acid leakage.

GL 88-05 stated that boric acid leakage potentially affecting the integrity of the RCPB should be procedurally controlled to ensure continued compliance with the licensing basis. The NRC requested each plant to provide a Boric Acid Corrosion Control Program (BACCP) to include the following:

1. A determination of the principal locations where leaks that are smaller than the allowable technical specification limit can cause degradation of the primary pressure boundary by boric acid corrosion. Particular consideration should be given to identifying those locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces.

2. Procedures for locating small coolant leaks (that is, leakage rates at less than technical specification limits). It is important to establish the potential path of the leaking coolant and the reactor pressure boundary components it is likely to contact. This information is important in determining the interaction between the leaking coolant and RCPB materials.

3. Methods for conducting examinations and performing engineering evaluations to establish the impact on the RCPB when leakage is located. This should include procedures to promptly gather the necessary information for an engineering evaluation before the removal of evidence of leakage, such as boric acid crystal buildup.

4. Corrective actions to prevent recurrences of this type of corrosion. This should include any modifications to be introduced in the present design or operating procedures of the plant that (a) reduce the probability of primary coolant leaks at the locations where they may cause corrosion damage and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings/claddings.

GL 88-05 was focused primarily on mechanical joints such as conoseals, flanges, valve bonnets, valve packing, gasketted joints, and mechanical seals. The program did not address potential leaks from the Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600 and Alloy 82/182 weld locations or leaks from other borated water system components.

Based on a review of some 50 utility responses to the GL 88-05, the NRC issued a report in 1990 that suggested that utility performances had a wide variation, ranging from "excellent" to "unsatisfactory."

More recent service experience over the past decade with Control Rod Drive Mechanism (CRDM) penetrations in reactor vessel heads confirmed PWSCC of Alloy 600 penetration base material and the Alloy 82/182 J-groove welds. Between January 2000 and July 2001, the industry reported several events involving leakage from Reactor Coolant System (RCS) piping, penetrations, and components (Ref. 2). These included leaks from CRDM penetrations, hot-leg nozzles, pressurizer nozzles, and reactor coolant pump and reactor vessel flanges.

As a result of the leaks detected in the CRDM penetrations at Oconee Units 1, 2, and 3 and Arkansas Nuclear One (ANO) Unit 1, the NRC issued Bulletin 2001-01 (Ref. 3) in August 2001, requiring susceptibility ranking-based inspection of Alloy 600 CRDM nozzles to ensure the structural integrity of the penetrations. As part of the industry response to the head penetration cracking issue, the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) developed a comprehensive inspection plan for the vessel head penetrations, which is described in MRP-75 (Ref. 7). The plan is currently being revised and will be submitted to the NRC. Until the MRP-75 document is finalized, the interim reactor vessel head inspections will be performed per the requirements of the NRC Order EA-03-009 (Ref. 8).

In March 2002, while conducting inspections in response to Bulletin 2001-01, Davis-Besse discovered evidence of substantial wastage of the low alloy steel vessel head due to boric acid leakage on the head resulting in a significant cavity around nozzle No. 3 (Ref. 4).

In response to the findings at Davis-Besse, the NRC issued Bulletin 2002-01 (Ref. 3). This bulletin requires licensees to submit a 60-day response to include *"The basis for concluding that your boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in GL88-05 and this bulletin. If documented basis does not exist, provide your plans, if any, for a review of your program."*

A sample review of the current GL 88-05 inspection programs and licensee 60-day responses to Bulletin 2001-01 suggested that significant variations exist in the inspection procedures, ownership and responsibility, personnel qualification and training, as well as in the detection and evaluation methods for boric acid corrosion.

Based on the above considerations, the Westinghouse Owners Group (WOG) Materials Subcommittee (MSC) appointed a task team to develop generic guidance to serve as a basis for the development of station-specific inspection procedures by the licensees.

In August 2002, the NRC released a bulletin (Ref. 5) on the missed early warning indicators and key findings of the Davis-Besse incident. The NRC document identified that significant evidence of boric acid deposits on the CRDM shroud fans, containment air circulation fan coils, cooling coils, and radiation monitor airborne filters were missed. The following are some of the deficiencies identified: inadequate attention to Alloy 600 leaks, management oversight and lack of ownership, inadequate documentation and initial assessment of boric acid deposits prior to clean up, and deficiencies in the personnel qualification.

As part of an effort to promote excellent BACCPs at nuclear power stations, the Institute of Nuclear Power Operations (INPO) issued a draft document "Guidance for Performing INPO Review Visits – PWR Primary System Integrity" (Ref. 6). The document provided a comprehensive listing of areas where

INPO review will be conducted to evaluate the licensee's performance in monitoring and maintaining the integrity of reactor pressure vessel head, the RCS, and other borated system pressure boundary components.

This WCAP report is intended to provide comprehensive guidance with good practice attribute to address any current deficiencies and provide a means of formulating uniform BACCP inspection programs for the PWR stations.

3 SCOPE

This generic guidance document is intended to describe Boric Acid Corrosion Control Program (BACCP) attributes and to serve as a technical basis in improving and enhancing existing plant-specific boric acid programs currently being used by utilities. While many programs are effective, sharing a common expectation based on recent industry experience can help avoid future corrosion damage to important components and support structures. This guidance document provides a structured approach for the inspection and mitigation of boric acid leakage and corrosion wastage in the American Society of Mechanical Engineer (ASME) Class 1, 2, and 3 systems and components which, when integrated into existing plant programs, can improve program effectiveness.

A boric acid leak in a Class 3 or non-safety system may not challenge the Reactor Coolant Pressure Boundary (RCPB). However, the effect of the boric acid leakage on adjacent systems and close proximity components should be evaluated on a plant-specific basis. Accordingly, auxiliary system components containing borated water and component supports are also included in the current guidance. As was noted previously, the purpose of this guidance is to improve the effectiveness the overall boric acid program and not to be limited to the Generic Letter GL 88-05 scope.

Consideration of inspection locations included industry-documented leaks, Primary Water Stress Corrosion Cracking (PWSCC) susceptible Alloy 600 and Alloy 82/182 locations, other susceptible sensitized stainless steel heat affected zones, Transgranular Stress Corrosion Cracking (TGSCC) locations, use of Operating Experience (OE) Report, and mechanical connections such as bolted joints, valve packings, gaskets, and seal welds.

The primary focus is directed to components inside containment, but the process to inspect and assess boric acid leakage effects is common to components both inside and outside the containment. Additionally, the principles and guidance provided can be applied to any medium that is corrosive to the surrounding pressure boundary components.

Included in the guidance are key elements such as: basis for identifying inspection locations, methods of inspection and data collection, damage assessment and corrective actions, program ownership and management oversight, personnel training, and continuous improvement by self assessment. Coordination of data from related parallel programs and utilization of critical early-warning indicators to detect the occurrence and location of a leak are also considered.

Attributes relating to management oversight, accountability, and program ownership are included in the document.

As a clarification, it is noted that the word "should" used throughout this report is intended to be a "recommendation" to the utilities. Utilities are expected to improve and enhance existing plant-specific BACCPs using the guidance provided in this document and other relevant information as it becomes available.

4 KEY ELEMENTS

The key elements that constitute a structured approach to Boric Acid Corrosion Control Program (BACCP) are the following:

1. Identification of Inspection Locations
2. Obstructions to Visual Inspections
3. Inspection Procedures
4. Inspection Methods
5. Other Inspections and Parallel Programs
6. Evaluation and Assessments
7. Data Collection and Documentation
8. Corrective Actions
9. Program Ownership and Responsibility
10. Personnel Training
11. Continuous Improvement and Self-Assessment

The following subsections describe each of the key elements.

4.1 IDENTIFICATION OF INSPECTION LOCATIONS

In this section, various considerations are discussed to identify the inspection locations. These considerations are based on the materials, system design aspects, and the industry operating experience. The following considerations, along with the referenced attachments (and any other plant-specific considerations) should be used as criteria in selecting the inspection locations. A schematic representation of the technical approach is illustrated later in Flow Chart 6-1.

1. The inspection program shall consider component locations that have a potential to leak borated water across the pressure boundary resulting in Boric Acid Corrosion (BAC) and wastage of low alloy steel or carbon steel in the proximity. The key factors for the occurrence of BAC include aqueous condition, boric acid concentration, exposure to oxygen, and metal surface temperature. Oxygen plays a crucial role in that no BAC of carbon or low alloy steel is expected unless the leak is exposed to oxygen from the outside environment. Austenitic stainless steels, martensitic stainless steels, precipitation hardening martensitic stainless steels, and nickel base alloys are resistant to BAC.

2. Service experience shows that mechanical seals and connections such as bolted joints, gasket and flanged connections, valve packing, and seal welds should be considered potential leak locations. A listing of documented leaks that resulted in BAC in the industry is included in Attachment 5.1. Information on documented leaks in the industry should be given particular attention while identifying inspection locations.

3. Some nickel-based alloys such as Alloy 600 and Alloy 82/182 welds exposed to borated water in the primary system are susceptible to Primary Water Stress Corrosion Cracking (PWSCC). Therefore, these locations where cracking can result in a leak path to the external surface should be considered for inspection as potential principal leak locations. Listings of typical Alloy 600 and Alloy 82/182 locations in the primary pressure boundary components of Westinghouse, Combustion Engineering (CE), and Babcock and Wilcox (B&W) designed Pressurized Water Reactor (PWR) units are provided in Attachments 5.2, 5.3, and 5.4, respectively.

4. Alloy 600 reactor vessel head penetration tubes and Alloy 82/182 penetration welds in the vessel head should be considered for inspection as outlined in the Materials Reliability Project (MRP) document MRP-75 (Ref. 7).

5. Sensitized stainless steels (weld heat affected zones) are subject to Intergranular Stress Corrosion Cracking (IGSCC) from exposure to service temperature and dissolved oxygen in the system. The resulting IGSCC leaks can readily come in contact with low alloy or carbon steel components in the close proximity leading to BAC. Accordingly, weld locations where extensive field modifications were conducted can be susceptible to IGSCC. These locations should be included in the inspection program as applicable, based on the industry service experience.

6. Some of the martensitic stainless steel materials, such as Type 410 stainless steel, may be adversely affected by contact with boric acid and oxygen. Components made out of such materials should be considered for inclusion into the BACCP scope.

7. The concentrated boric acid systems such as the Chemical and Volume Control System (CVCS), Safety Injection (SI) System, Residual Heat Removal System (RHR), and Boron Recycle System contain carbon steel components (such as fasteners) that have potential for exposure to ambient (oxygen) conditions. These components are therefore susceptible to BAC. Typical examples of potential leak locations with BAC wastage significance of carbon steel components in selected systems of Westinghouse and CE designed PWR plants are provided for guidance in Attachments 5.5 and 5.6, respectively. These should be considered only as examples for limited systems. The inspections should be inclusive of all systems containing borated water. A typical listing of systems containing boric acid is provided in Attachment 5.7.

8. Indication of possible boric acid leakage, such as surface streaks on bare metal, boric acid residue at insulation seams, or bulges in insulation shall be investigated to determine if a boric acid leak does exist.

9. Carbon steel or low alloy steel components in the proximity of a potential leak location are vulnerable to BAC, both on external surfaces and along the leak path. On this basis, all potential leak locations should be prioritized for inspection based on their potential for carbon steel/low alloy steel corrosion and its safety significance.

A schematic illustration of the summary of inspection locations is provided later in Flow Chart 6-1. The flow chart illustrates the technical approach in choosing the inspection locations under the scope of the current Guidance as compared to the Generic Letter GL 88-05 inspection locations. The additional locations included Alloy 600/82/182 locations with wastage significance, Auxiliary System BAC susceptible locations, locations chosen from industry experience (Nuclear Regulatory Commission (NRC) documented leaks), and other plant-specific locations chosen on the basis of trending, service experience, or other cycle-specific reports.

4.2 OBSTRUCTION TO VISUAL INSPECTIONS

The following describes considerations to be included in deciding when insulation should be removed to perform the visual inspections. The criteria for insulation removal should be set by the plant-specific BACCP procedures:

1. Criteria for inspecting all potential boric acid corrosion locations that are inaccessible, including locations covered with insulation, should be developed. Alternate methods should be specified for locations obstructed from visibility. These may include camera surveillance, monitoring adjacent location, etc. For example, camera surveillance can be effective for monitoring locations where visibility is obstructed by physical constraints, and monitoring an adjacent location may be effective for locations where accessibility is restricted.

2. Criteria for removing insulation for inspection should be established. These may include any evidence of leakage such as streaks originating from under the insulation, a bulge in the insulation, past plant-specific leak history, industry experience, or safety significance of the specific location.

3. Consideration should be given to the need to remove insulation at Alloy 600 and Alloy 82/182 locations. This consideration should include the potential need to complete this effort at some regular interval so that bare metal at the susceptible locations may be inspected. Industry Operating Experience (OE) records should be reviewed in selecting the susceptible locations ultimately included in this inspection plan. The inspection interval should be based on industry operating experience and plant-specific susceptibility rankings.

4. A discussion of allowance for system pressure tests and VT-2 visual examination without removal of insulation is provided in the American Society of Mechanical Engineers (ASME) Code Cases N-533-1 and N-616 for reference.

4.3 INSPECTION PROCEDURES

Plant-specific boric acid inspection procedures should be developed by each Utility based on the generic guidance given in the current WCAP report. The implementation procedure should meet the management expectations in controlling the boric acid corrosion leaks and subsequent corrosion. The inspection procedures include the following:

1. A top-tier document outlining the management expectations, standards, and commitments on Boric Acid Corrosion Control (BACC) should be issued by each utility. The implementation procedure should meet the expectations of the top-tier document.

2. All identified boric acid leaks should be documented on a standard BACCP or other site-issued report tracking form. An example of such a document is provided in Attachment 5.8. Plant procedures should include guidance on requirements for cleaning boric acid crystals and definition of cleaned surface. The criteria for as-found and as-left cleanliness of the bare metal should be established on a plant-specific basis.

3. Written guidance should be provided to the qualified personnel conducting the initial inspection of the BAC leaks to ensure all necessary information is properly collected and documented. Plant procedures should contain instructions for evaluation of the effects of boric acid leakage on carbon steel components. Procedural changes that could affect the site's NRC commitments on the BACCP should be evaluated for safety significance and regulatory basis impact.

4. Procedures should ensure that boric acid leaks are reported and properly inspected and evaluated. Inspections should be performed prior to cleaning any deposits present to ensure that leaks are properly characterized, as described. Procedures should ensure that, once inspected, evaluations are performed and documented and that leaks are entered into the corrective action process.

5. All deposits should be properly documented. The documentation may include, but not be limited to:

   - Photographs or digital images of the location
   - Size and physical appearance of the deposit
   - Assessment to establish if boric acid leakage is active
   - Color and chemical composition assessment of the deposit
   - Any radiochemistry assessment of the age of the deposit, if performed, should be properly documented

6. Clear instructions should be provided to assess the collective significance of various related inspections and monitoring devices, such as changes in containment ventilation equipment, fans, atmospheric radiation monitors, and changes in containment temperature and humidity monitors. Inspections considered significant should be completed on a routine basis and trended with alert and action levels documented.

7. The procedural guidance for inspection of the reactor vessel head, Control Rod Drive Mechanism (CRDM) penetration tubes, and 'J-groove' welds are described in the Electric Power Research Institute (EPRI) report EPRI-MRP-75 (Ref. 7).

8. The corrective action procedures should give guidelines for periodic evaluation of a boric acid leak in a system if the system is returned to service without fixing the leak. Guidelines provided in the EPRI Boric Acid Corrosion Guidebook (Ref. 9) should be integrated into site standards when performing the engineering evaluations.

4.4 INSPECTION METHODS

Inspection methods can vary from plant to plant. A combination of general area walkdowns and specific checklists can be used to perform the inspections. Regardless of the approach used, the inspections should encompass some common elements so that an effective boric acid inspection is performed to meet the expectations of the management and regulators. Some of the key attributes are described below:

1. Boric acid corrosion inspections inside containment are required to be performed during each refueling outage as per the requirements of NRC GL 88-05. Additional opportunities to perform boric acid corrosion walkdowns inside containment should be considered during unscheduled shutdowns, if such a walkdown has not been performed within a certain time frame, (that is, since the last refueling outage). Plant procedures should provide specific frequencies or conditions for performing boric acid walkdowns during shutdowns under permissible radiological and environmental conditions.

2. The containment BACC walkdowns should be conducted as soon as practical after taking the core subcritical. Some small leaks are easily detected while the system is still under pressure. (Some plants, such as sub-atmospheric containment design, may have restrictions precluding Mode 3 walkdown but initial walkdowns should be encouraged as soon as practical following the units' shutdown.)

3. General area walkdown inspections should be conducted both inside and outside containment, looking for any signs of leakage. The inspections shall provide sufficient assurance that boric acid leakage will be detected and potential leakage locations and leak paths in all areas are adequately inspected. The inspectors should be knowledgeable of the locations of susceptible nickel alloys and carbon steel fasteners and components that could be adversely affected by boric acid leakage.

4. Specific checklists can be used successfully to provide assurance that those potential boric acid leakage locations will be inspected. Adequate procedural requirements and training should be provided to ensure that boric acid leakage on floors, walls, and other locations not specified in the checklists are identified and their associated leak source located. Source components or carbon steel targets that are difficult to locate should be included in the instructions to ensure alternate coverage. The inspection should include potential targets of leak path (such as, supports or electrical cabinets). A balanced combination of general area walkdowns and focused checklists can provide the assurance of effective BACCP coverage throughout the plant.

5. Information from other routine outside containment inspections by various personnel (operators, system managers/system engineers, radiation protection personnel, etc.) should be properly coordinated to integrate information from all sources into an effective BACCP.

6. The reactor vessel head inspections should follow the guidance identified in the MRP-75 document (Ref. 7). No additional guidance for reactor vessel head, CRDM tubes, and reactor vessel head 'J-groove' weld inspection is provided in this document.

7. The scope of inspection should include a multifaceted approach to detect and verify the source of leaks. These inspections should include all ASME inspection locations regardless of the purpose or inspection technique being used for the ASME examination. Also, on-line detection and trend tracking techniques should be included as appropriate.

8. The BACCP should consider on-line detection and trend-tracking techniques, such as, airborne particulate radioactivity monitors, airborne gaseous radioactivity monitors, humidity monitors, temperature monitors, Reactor Coolant System (RCS) water inventory balancing, and containment air cooler thermal performance to identify potential leakage during power operation. Weekly, daily, or even shift-based review of leak rates and trending should be conducted based on identified RCS leak locations, unidentified RCS leak rate, and average containment sump pumping frequency.

9. Detection of significant increases in any of the various indications noted above should be investigated. Significant changes in the on-line detection signals should be correlated with the RCS inventory balance to identify leak rates below the technical specification limit and their possible location.

10. The BACCP should include early warning indicators, such as, evidence of boric acid deposits on CRDM shroud fans, containment air recirculation (CAR) fan coils, containment fan cooler units (CFCUs), and airborne filter deposits. Triggers for action items should be provided to require followup evaluations and inspections. When various indicators are suggesting borated water leakage, consideration should be given to increase the inspections to determine the root cause.

11. VT-2 visual inspections, as well as other inspections, may be utilized in identifying boric acid deposits due to a leak. Dye Penetrant Testing (PT) and/or Magnetic Particle Testing (MT) may be used to determine the location and length of cracks at the suspected leak locations.

12. A boric acid leak should be considered as a challenge to carbon steel components in its proximity. Removal of boric acid deposits should not be treated as a decontamination activity without an engineering evaluation. Boric acid deposits on components should not be removed until the as-found condition is documented or the source of the leak is identified and evaluated. Removal of the deposit may be necessary in some cases to identify the source of the leak.

13. As described in GL 88-05, inspectors should ensure that the source of a boric acid leak is identified. If insulation is required to be removed to locate a leakage source, this should be done in such a way to preserve any evidence of leak prior to completion of the inspection procedure documentation. Critical consideration should be given to tracing of surface streaks on bare metal, indications of boric acid at insulation seams, or bulges in the insulation in identifying the origin of a leak. The concentrated boric acid systems, such as the CVCS, may contain carbon steel components (such as fasteners) and have potential to be exposed to oxygen.

14. Inspectors should be cognizant of the various forms of boric acid and the meaning of the various deposit types, from white powdery deposits, brownish boric acid crystals, or reddish rusty deposits. For example, brown, red, and pink colored deposits may be indicative of active carbon steel corrosion. Some leaks, particularly PWSCC in Alloy 600 and Alloy 182/82, may only be

apparent at service conditions. These locations may be active leaks, but have the appearance of a dry leak during system walkdowns.

15. The inspection data collection and data recording methods should be adequate to provide quality images of the observed condition. Consideration should be given to utilize the state-of-the-art technology for data recording, such as, portable devices, hand-held PC, and integrated digital recording and imaging system, including voice recordings to document as-found conditions.

16. All site personnel must be intolerant of leakage, and therefore should be initiating corrective action documents to get leaks into the plant corrective action system.

4.5 OTHER INSPECTIONS AND PARALLEL PROGRAMS

There are many opportunities to inspect Reactor Coolant Pressure Boundary (RCPB) components, other systems and components, and other plant areas that could be affected by boric acid leakage and corrosion. The BACCP should take the benefit of other parallel programs in identifying the potential leak locations. Interfacing with other programs will help in identifying abnormal plant conditions or indications that may not be readily explained. The BACCP should include evaluation, tracking, and correction of any indications of leakage detected during these plant activities. The inspection results should integrate and correlate data collected from parallel activities that impact the BACCP. A few examples of opportunities to detect boric acid leaks and corrosion are the following:

1. Inside containment examinations (examples):

    a. Refueling outage
    b. Initial containment entries
    c. Boric acid walkdown
    d. Disassembly for refueling
    e. Maintenance and In-Service Inspection (ISI) activities
    f. System leakage pressure tests (ISI) (this includes Class 1, 2, and 3 systems)
    g. Operational surveillance tests
    h. Scheduled bolted connection (ISI) examinations
    i. Containment startup walkdown for system leak test
    j. System pressure test and hydro tests
    k. Routine containment entries

2. Outside containment examinations (examples):

    a. Normal shift tours by operators
    b. Operational surveillance tests
    c. Routine Health Physics (HP) surveys
    d. System engineering walkdowns
    e. Scheduled system leakage pressure examinations
    f. Component disassembly and maintenance activities

3. The inspection results from various other programs described above should be evaluated for their impact on the BACCP. When various indicators suggest boric acid leakage, management should be made aware of the conditions and consideration should be given to additional inspections to determine the root cause.

4.6 EVALUATIONS AND ASSESSMENTS

It is preferable that the boric acid leak (and deposit) evaluations and assessments are performed in a systematic method. This key element discusses the details of the contents of the evaluations, type of information required for an effective evaluation, and criteria for dispositioning the inspection findings.

1. An initial evaluation of the boric acid leak/deposit should be performed prior to removal of boric acid deposits. This is necessary to ensure that vital information on the leak, such as, time history of origin, composition, and leak pathway can be clearly established prior to the removal of evidence. A flow chart representation of evaluation criteria of the boric acid deposit is illustrated later in Flow Chart 6-2.

   Documentation should include the following information, as a minimum:

   a. For an identified boric acid leak, the location of the leak should always be determined.

   b. A determination should be made whether the leak has affected or degraded other components in the proximity of the leak or in the leak path.

   c. A determination should be made if the leakage is wet or dry. The plant conditions during the walkdown (such as, pressurized or not, hot or cold) should be taken into consideration during this determination.

   d. A determination as to the quantity and color of boric acid leakage should be made.

   e. All leakage onto carbon steel components or subcomponents should be entered into the site corrective action program for evaluation, maintenance, or replacement.

   f. Test results from system pressure tests may be utilized in quantifying leakage and assessing the significance of a leak location.

2. The following techniques may be helpful in determining the timing, source, and specific conditions about a leak and resulting corrosion:

   a. Radiochemistry (isotope half-life) analysis to estimate the time history of the leak (such as, from a recent leak or from a previous outage)

   b. Concentration analysis to assess the corrosion rate

   c. Depth and distribution of the corrosion attack assessment by visual and surface replication technique, as appropriate

   d. Crack characteristic determination if leakage is from a throughwall crack

   e. Location, orientation, and geometry of the attack/crack by applicable volumetric Non-Destructive Examination (NDE) techniques when warranted (See Item 10, Section 3.4)

f. For structural integrity evaluations, while developing NDE techniques, Eddy Current Testing (ET), Ultrasonic Testing (UT), and/or Radiographic Testing (RT) can be utilized. Metallographic surface replication can be useful to determine the precise location of cracks (such as, at a weld) and to establish the cracking morphology.

3. Significance assessment of the BAC-affected component should be conducted to establish criteria for disposition as follows. The BAC assessment should employ wastage rates provided in the EPRI Boric Acid Corrosion Guidebook (Ref. 9), unless other wastage rates can be determined to be more appropriate and documented as applicable:

   a. Immediate corrective action needed

   b. No action

   c. Need to follow (monitor)

   d. Establishment of monitoring frequency

   e. Perform an engineering evaluation of the observed condition for continued operations, including a schedule for corrective action or repair

4. If a complete evaluation of the leak cannot be performed due to accessibility restraints or As-Low-As-is-Reasonably-Achievable (ALARA) considerations, the program should require appropriate management involvement to resolve the restraint.

5. The program should require evaluation of the boric acid leakage by knowledgeable Engineering Department personnel if the leakage cannot be stopped, if base material is degraded (evidence of wastage is detected), or if it is returned to service with an active leak. Concurrence from the Operations Department should be obtained for a system returned to service with an active leak. Engineering should be provided with detailed guidance for performing such an evaluation. The evaluation guidance should be based on the EPRI Boric Acid Corrosion Guidebook (Ref. 9) and industry operating experience.

6. The repair and corrective actions should comply with site repair procedures and ASME Code requirements as applicable.

7. Minor corrosion that is determined to be acceptable as is (either on a temporary or permanent basis) should have an evaluation documented by a knowledgeable person for future reference.

8. Any corrosion that may affect structural integrity or component function should have a documented evaluation prepared by a trained individual. This could include visual and other surface techniques and volumetric NDE techniques as required. Engineering personnel should be involved in the determination of structural integrity.

9. Component evaluations should include consideration of leakage rates, temperatures during plant operation, and the potential corrosion.

10. Any boric acid corrosion evaluation should employ wastage rates provided in the EPRI Boric Acid Corrosion Guidebook (Ref. 9).

11. Once completed, all walkdown results should be reviewed by the BACCP owner.

12. Plant procedures should provide criteria for dispositioning inspection findings. The list provided below is one example of criteria for dispositioning leakage and prioritizing corrective actions:

    a. Any boric acid deposit emanating from a pressure boundary component other than a mechanical joint should be thoroughly investigated, including other NDE techniques (such as, VT-1, liquid penetrant, or eddy current). Any flaw in the base material should be entered into the plant's corrective action program, with the priority and schedule requirements commensurate with technical specifications and other safety system requirements.

    b. Leakage detected from or on an ASME Class 1, 2, 3 systems component and non-safety carbon steel subcomponent or structure should be considered for immediate repair or maintenance as required. A specific schedule should be identified for disposition. This should be entered into the plant corrective action program.

    c. Minor leakage from a bolted mechanical joint not leaking onto carbon steel ASME sub-components could be considered for repair or maintenance subject to a specific constraint or time frame. This should be entered into the plant's corrective action program. An evaluation should be performed to support the proposed time frame for corrective action.

    d. Boric acid buildup on or from a mechanical joint should be considered for cleaning and monitoring as required. This should be entered into the plant's corrective action program. An evaluation should be performed to support the proposed time frame for corrective action. The following examples are offered for inclusion in a site BACCP:

        – Minor boric acid film from a mechanical joint - Accept as is with no cleaning required.

        – Minor boric acid deposit from a mechanical joint - Cleaning and tracking for one subsequent outage only is required. This may be entered into the plant's corrective action program or tracked in another positive manner. Sufficient traceability should be provided to allow trending of the frequency of cleaning. Repeated boric acid cleanup of a component may require additional corrective actions.

        – Assessment of borated water leaks at seal joints should be consistent with the station's program on leakage from components outside containment.

4.7 DATA COLLECTION AND DOCUMENTATION

This key element lists the attributes required to document the data collection when a boric acid leak and deposit is identified:

1. Written guidance should be provided for the inspection of boric acid leaks and associated areas of boric acid corrosion to ensure all necessary information is properly collected and documented in a consistent format. The evaluation and assessment of leaks should be documented. An example of such a form is provided in Attachment 5.8.

2. At the time of detection of a BAC leak/deposit, a complete description including photographs or digital image recordings should be made to document as-found conditions. Voice recordings may also be useful in describing the as-found condition.

3. The inspection data collection and data recording methods should consider the use of state-of-the-art technology for data recording, such as, portable devices, hand-held PC, integrated digital recording and imaging system, including voice recordings to document as-found conditions.

4. A log or database of all boric acid leak locations (active and inactive) should be maintained for evaluation and monitoring proposes.

5. The walkdown inspections should consider recording methods that would allow quick and easy downloading of the data into a computer database.

6. The data collection methods should include recording the cycle-specific or refueling-outage-specific data such that cycle-to-cycle changes and trending can be identified. The methods should lend to issuance of timely reports and data retrieval opportunities.

7. BAC inspection records should be treated as quality records.

8. The disposition activity for an identified leak should not be completed until boric acid cleanup is sufficient to ensure that base metal condition is adequately assessed.

9. The criteria for cleanliness of the bare metal should be established on a plant-specific basis.

4.8 CORRECTIVE ACTIONS

This section describes the corrective actions to be considered, if needed, to prevent reoccurrence of a boric acid leak.

1. If a leak in the RCS is identified in an ASME Class 1, 2, or 3 system, then the leak should be evaluated and corrective actions to repair/replace or monitor should be implemented. The schedule for corrective actions should be commensurate with the safety significance and equipment functional requirements as well as the requirements of ASME Code.

2. Each station should have a decision-making process and criteria identified to trigger actions in the event of development of a leak (for example, at a bolted connection during startup). The plant should have a formal bare metal inspection or leakage acceptance criteria developed. The corrective action taken should be based on the complete characterization of the boric acid corrosion (deposit and degradation) before and after cleanup.

3. Components detected with a boric acid deposit should be cleaned and removed upon satisfactory documentation of the "as-found" condition to assist with the identification of any new leakage in the future.

4. The corrective action considerations should include leakage reduction guidance as provided in Section 7 of the EPRI Boric Acid Corrosion Guidebook (Ref. 9). For components and bolted joints that experience repeated boric acid leaks, consideration should be given for replacement of the component based on its safety significance and ranking.

5. Any repair or design modification should follow ASME Code requirements as appropriate. Corrective actions should include consideration of alternate materials for replacement to prevent recurrence of the boric acid leakage. Corrective actions to prevent reoccurrence should be prompt and timely.

4.9 PROGRAM OWNERSHIP AND RESPONSIBILITY

The BACCP should have a designated program owner with defined responsibilities. Some of the roles and responsibilities of the program owner are listed below:

1. A BACCP owner should be established with defined responsibilities to ensure appropriate ownership.

2. Management oversight should be provided.

3. The roles and responsibilities of supervisors, HP/Radiation Protection (RP) personnel, inspectors, engineers, and managers who are directly or indirectly involved in the boric acid inspection, assessment, and remediation should be clearly defined in the program documents.

4. A flow chart representation of responsibilities and information flow will be helpful to all personnel involved in the BACCP. The flow chart should clearly identify the responsibilities for decision making.

5. BACCP procedures should be consistent with site-specific commitments made to the NRC.

6. Periodic self-assessments should be conducted.

7. An easily accessible list or database of active leakage should be maintained. This may be a specific database, a periodically updated list, or a retrievable category of corrective action document.

8. The data collection methods should include recording the cycle-specific or refueling-outage-specific data such that cycle-to-cycle changes and trending can be identified. The methods should lend to issuance of timely reports and data retrieval opportunities.

9. Industry operating experience or feedback should be incorporated into the program.

10. A program health report or other communication mechanism should be implemented at a reasonable frequency. This will ensure management awareness of the program performance, as well as plant condition, with respect to the BACCP.

4.10 PERSONNEL TRAINING

Personnel involved in boric acid inspections, evaluations, and corrective actions should be trained and knowledgeable. This key element lists the personnel training requirements:

1. The personnel performing the inspections should be formally trained and certified under a qualified BAC-specific program. The training program should be updated periodically to reflect industry experience and lessons learned and should include specific training on boric acid corrosion issues. The training should include program requirements, inspection methods, code requirements, acceptance criteria, important design aspects, and industry operating experience. The training should also include mechanistic aspects of boric acid corrosion of low alloy and carbon steels and the contributing factors.

2. Training requirements should be established for the engineering personnel who perform boric acid leakage evaluations and safety significance assessments.

3. Any engineering (or other) personnel involved in the inspections or evaluations should be certified per code requirements or trained under a separate documented training method. Engineers should be provided with detailed guidance for performing evaluation. The evaluation guidance should be based on EPRI Boric Acid Corrosion Guidebook (Ref. 9) and industry OEs.

4. Personnel involved in the housekeeping and decontamination should be trained in the program. Specifically, they should be trained to report all leakage to the appropriate organization, typically engineering, prior to cleanup if it is from a borated system.

5. All personnel involved in the BACCP, including managers (and acting managers), should be trained in the program process and plant-specific procedure requirements. A questioning attitude should be emphasized in the training, especially for managers and supervisors.

4.11 CONTINUOUS IMPROVEMENT AND SELF ASSESSMENT

The plant-specific BACCP should be continuously upgraded to include the latest industry experience and any new technology in detecting boric acid leaks. Efforts to continuously improve the inspection and evaluation performance would help to meet high standards rather than meeting minimum standards. This key element lists some of the activities to improve the plant-specific BACCP.

1. The BACCP should be continuously improved by conducting program effectiveness reviews including:

   a. Periodic program self assessments, recommended once every 2 years following significant programatic enhancements. For mature programs, the self assessments can range from 3 to 5 years.

   b. Independent program audits and benchmarking against industry peer plants should be conducted periodically.

   c. Issuing cycle-specific reports to management.

   d. Incorporating/updating program with experience from industry events such as review of Institute of Nuclear Power Operations (INPO) documents, Significant Operating Event Reports (SOERs), OE, Safety Evaluation Reports (SERs), vendor technical bulletins, owner's group documents, and the NRC communications.

   e. Issuance of periodic program health reports based on specific program effectiveness performance indicators as addressed in the INPO document (Ref. 6).

   f. Keeping senior management informed of the status of the program as appropriate to ensure continued management support.

5 ATTACHMENTS

5.1 Summary of Industry-Documented Leaks from NRC Bulletins

5.2 Alloy 600 and Alloy 82/182 Potential Leak Locations in the Primary Components of Westinghouse Units

5.3 Alloy 600 and Alloy 82/182 Potential Leak Locations in the Primary Components of Combustion Engineering Units

5.4 Alloy 600 and Alloy 82/182 Potential Leak Locations in the Primary Components of Babcock and Wilcox (B&W) PWR Plants

5.5 Typical Examples of Potential Leak Locations in the Auxiliary Systems of Westinghouse Units

5.6 Typical Examples of Potential Leak Locations in the Auxiliary Systems of Combustion Engineering Units

5.7 Listing of Systems Containing Boric Acid

5.8 Typical BACC Issue Documentation Form

ATTACHMENT 5.1

SUMMARY OF INDUSTRY-DOCUMENTED LEAKS FROM NRC BULLETINS

Table 5.1-1
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
IN 80-27	Degradation of Reactor Coolant Pump Studs	6/11/80	Corrosion damage to a number of closure studs in two of the four Byron Jackson RCPs at Fort Calhoun (FTC). Cause of the wastage is thought to be corrosive attack by hot boric acid from the primary coolant. The condition of the studs discovered at FTC raises concerns that such severe corrosion, if undetected, could led to stud failures which could result in loss of integrity of the reactor coolant pressure boundary. The lack of effectiveness of current UTs in revealing wastage emphasizes the need for supplemental visual inspections and use of instrumented leak detection systems to preclude unacceptable stud degradation going undetected. Licensees should consider that the potential for undetected wastage of carbon steel bolting by a similar mechanism could exist in other components such as valves.	None required.
IN 82-06	Failure of Steam Generator Primary Side Manway Closure Studs	3/12/82	At Maine Yankee, 6 of 20 manway closure studs failed an another 5 were found by UT to be cracked. Boric acid from a small leak was the cause. Reference was made to similar events at Calvert Cliffs, FTC, Oconee, and ANO-1.	None required.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
BL 82-02	Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants	6/2/82	Recaps the FTC and Maine Yankee bolting problems in IN 80-27 and IN 82-06. Adds that certain lubricants may promote stress corrosion cracking. At the present time, visual examination (e.g., IWA 2210, VT, VT-1) appears to be the only method to detect borated water corrosion wastage or erosion-corrosion damage and may require insulation removal and/or disassembly of the component, in some cases, in order to have direct visual access to the threaded fasteners.	1. Develop and implement procedures for threaded fasteners practices. 2. Threaded fasteners of closure connections, identified in the scope of this bulletin, when opened for component inspection or maintenance shall be removed, cleaned, and inspected per IWA-2210 and IWA 2220 of ASME Code Section XI before being reused. 3a. Identify those bolted closures of the RCPB that have experienced leakage, particularly those locations where leakage occurred during the most recent plant operating cycle. Describe the inspections made and corrective measures taken to eliminate the problem. If the leakage was attributed to gasket failure or its design, so indicate. 3b. Identify those closures and connections, if any, where fastener lubricants and injection sealant materials have been or are being used and report on plant experience with their application particularly any instances of SCC of fasteners. Include types and composition of materials used. 4. A written report to the Regional office within 60 days following the completion of the outage during which Action Item 2 was performed. (4a) A statement that Action Item 1 has been completed. (4b) Identification of the specific connections examined as required by Action Item 2. (4c) The results of examinations performed on the threaded fasteners as required by Action Item 2. If no degradation was observed for a particular connection, a statement to that effect, identification of the connection and, whether the fasteners were examined in place or removed is all that is required. If degradation was observed, the report should provide detailed information. 5. A written report to the Regional office within 60 days of the date of this bulletin. The report is to provide the information requested by Action Item 3.
IN 86-108	Degradation of Reactor Coolant System Pressure Boundary Resulting From Boric Acid Corrosion	12/29/86	Alert recipients of a severe instance of boric acid induced corrosion of ferritic steel components in the reactor coolant system. In October 1986, ANO-1 discovered the wastage of the exterior of the HPI nozzle and some wastage of the RCS cold leg pipe (upon removal of insulation). Leakage of RCS from a leaking HPI valve which was above the nozzle and pipe. The corrosion was approximately 1/4 inch deep. Boric acid corrosion has been found to be most active where the metal surface is cool enough so that it is wetted. If the metal is sufficiently hot, then the surface will stay dry and this loss of electrolyte will slow the corrosion rate. Boric acid corrosion rates in excess of 1 inch depth per year in ferritic steels have been experienced in plants and duplicated in laboratory tests where low quality steam from borated reactor coolant impinged upon a surface and kept it wetted.	None required.

Table 5.1-1. (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
IN 86-108 Sup #1	Degradation of Reactor Coolant System Pressure Boundary Resulting From Boric Acid Corrosion	4/20/87	On 3/13/87, Turkey Pt. 4 discovered more than 500 # of boric acid crystals on the RV head. There also was a large amount of boric acid crystals in the exhaust cooling ducts for the control rod drive mechanisms. After removal of this boric acid and steam cleaning of the RV head, severe corrosion of various components on the RV head was noted. This event has once again demonstrated that boric acid will rapidly corrode ferritic steel components and it also again demonstrated that if a small leakage occurs near hot surfaces and/or surroundings, then the boric acid solution will boil and concentrate, becoming more acidic and thus more corrosive. On 3/13/87, Westinghouse, the NSSS vendor, completed a review of boric acid corrosion rates, as earlier requested by the licensee, and reported that the corrosion rate might be much faster than assumed when the licensee's evaluation was performed. Reference was made to experience in Europe for a PWR in 1970 which experienced high corrosion rates for boric acid induced corrosion. Three RV head bolts, the CRDM cooling shroud were replaced because of corrosion.	None required.
IN 86-108 Sup #2	Degradation of Reactor Coolant System Pressure Boundary Resulting From Boric Acid Corrosion	11/19/87	Two events are presented: Following shutdown of Salem 2 on 8/7/87, inspection teams entered containment building to look for reactor coolant leaks that would account for the increased radioactivity in containment air that was noted before the shutdown. Boric acid crystals were found on a seam in the ventilation cowling surrounding the reactor head area. The licensee then removed some of the cowling and insulation and discovered a mound of boric acid residue at one edge of the reactor vessel head. A pile of rust-colored boric acid crystals 3 feet by 5 feet by 1 foot high had accumulated on the head, and a thin white film of boric acid crystals had coated several areas of the head and extended 1 to 2 feet up the control rod mechanism housings. The source of the leak was the thermocouple instrumentation pinhole leaks. Nine corrosion pits in the vessel head were found. The pits were 1 to 3 inches in diameter and 0.4 to 0.36 inches deep. While attempting to open a shutdown cooling valve at San Onofre 2 on 8/31/87, the packing area came apart (fasteners corroded by boric acid) and eventually dumped 18,000 gallons of reactor coolant in to the containment. Westinghouse reported that boric acid corrosion rates are greater than those that were either previously known or estimated.	None required.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
IN 86-108 Supplement 3	Degradation of Reactor Coolant System Pressure Boundary Resulting From Boric Acid Corrosion	1/5/95	Presents two additional events involving boric acid corrosion: Calvert Cliffs 1 (2/94), and TMI1 (3/7/94). In 2/94, Calvert Cliffs 1 (CC1) found three nuts on an incore instrumentation flange that were corroded by boric acid, resulting in a leak. During a subsequent inspection, three more nuts on another incore instrumentation flange were also corroded by the same mechanism. On 3/7/94, and while a 100 % power, TMI1 was trying to eliminate a leak of a pressurizer spray valve by tightening a bonnet stud, when the leak suddenly increased to 3 gpm. Other studs completely failed. CC1 thought that the corrosion rate from the leakage was acceptably low in 6/93, and elected to defer the corrective actions for the flanges until the 1994 refueling outage. Other parts of the IN recap earlier problems with boric acid corrosion.	None required.
GL 88-05	Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants	3/17/88	The principal concern is whether the affected plants continue to meet the requirements of GDC 14, 30, and 31 of Appendix A when the concentrated boric acid solution or boric acid crystals, formed by evaporation of water from the leaking reactor coolant, corrode the reactor coolant pressure boundary. The GL cites Turkey Pt. 4, Salem 2, San Onofre 2, ANO-1 and FTC. The GL cites BL 82-2 as not requiring the licensees to institute a systematic program for monitoring small primary coolant leakages and to perform maintenance before leakages could cause significant corrosion damage. Because of this deficiency in the BL, the GL requests 4 actions to be taken by licensees.	(1) Determine the principal locations where leaks that are smaller than the allowable TS limit can cause degradation of the primary pressure boundary by boric acid corrosion, (2) establish procedures for locating small coolant leaks, (3) establish methods for conducting examinations and performing engineering evaluations once a leak is located, and (4) corrective actions to prevent recurrence of this type of corrosion. Responses are required within 60 days of the date of the GL.
IN 90-10	Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600	2/23/90	Alert licensees to potential problems related to PWSCC of Inconel 600 that has occurred in pressurizer heater thermal sleeve and instrument nozzles at several domestic and foreign PWR plants. During the 1989 refueling outage at CC2, visual examination detected leakage in 20 pressurizer heater penetrations and 1 upper level pressure tap instrument nozzle. Leakage was indicated by the presence of boric acid crystals. The heater sleeves and the instrumentation nozzles were made of Inconel 600 tubing and bar materials, respectively, supplied by INCO. All instrument nozzles were made from heat no. NX8297. On 2/27/86 a small leak was observed on a 3/4 inch diameter upper pressurizer level instrument nozzle at SONGS 3. Two foreign reactors were also cited involving Inconel 600. PWSCC was first reported by Coriou almost 30 years ago. The studies of PWSCC in Inconel 600 have been documented in numerous reports, however, the mechanism for PWSCC in Inconel 600 is still not well understood. It may be prudent for licensees of all PWRs to review their Inconel 600 applications in the primary coolant pressure boundary, and when necessary, to implement an augmented inspection program.	None required.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
IN 94-63	Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks	8/30/94	Alert licensees to the potential for significant damage that could result from corrosion of reactor system components caused by cracking of the stainless steel cladding. Severe corrosion damage of the carbon steel casing of a high head safety injection pump at North Anna 1. The damage was caused by cracks through the stainless steel cladding in the pump that allowed corrosive attack by the boric acid coolant. The corrosion had penetrated to within about 0.125 inch of the outside surface of the pump (2.5 inches long by 1.5 inches wide by 0.5 inches deep).	None required.
IN 96-11	Ingress of Demineralizers Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations	2/14/96	Alert licensees to the increased likelihood of stress corrosion cracking of PWR control rod drive mechanism penetrations if demineralizer resins contaminate the reactor coolant system. The NRC determined that the safety significance of the cracking was low because the cracks were axial, had a low growth rate, and were in a material with an extremely high flaw tolerance (high fracture toughness). Accordingly, the cracks were unlikely to propagate very far. In December 1991, after cracks were found in a CRDM penetration in the reactor head at a French plant (Bugey 3), an NRC action plan was implemented to address PWSCC at all U. S. plants. The NRC asked the Nuclear Management and Resources Council (NEI) to coordinate future industry actions because the issue was applicable to all PWRs. Each owners group submitted individual safety assessments, dated February 1993, through NEI to the NRC on the CRDM cracking issue. In July 1993, the NEI submitted to the NRC proposed acceptance criteria for flaws identified during inservice examination of CRDM penetrations. On the basis of owners group analyses and the European experience, the NRC concluded that there was a high probability that CRDM penetrations at U.S. plants may contain similar axial cracks caused by PWSCC. In 1994, an inspection for PWSCC at a reactor in Spain identified cracks which were apparently initiated by high sulfate levels in the reactor coolant system. 16 of 17 spare penetrations showed stress corrosion cracking, and 4 of 20 active penetrations showed stress corrosion cracking.	None required.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
GL 97-01	Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations	4/1/97	This GL requests licensees (1) to describe their program for ensuring the timely inspection of PWR control rod drive mechanism and other vessel closure head penetrations and (2) require that all addresses provide to the NRC a written response to the requested information. Beginning in 1986, leaks have been reported in several Alloy 600 pressurizer instrument nozzles at both domestic and foreign reactors from several different NSSS vendors. In 1989, PWSCC was an emerging technical issue, after cracking was noted in Alloy 600 pressurizer heater sleeve penetrations at a domestic facility. The NRC staff determined that the cracking was not of immediate safety significance because the cracks were axial, had a low growth rate, were in a material with an extremely high flaw tolerance (high fracture toughness) and, accordingly, were unlikely to propagate very far. These factors also demonstrated that any cracking would result in detectable leakage and the opportunity to take corrective action before a penetration would fail. European and Japanese utilities have taken steps to detect and mitigate the PWSCC damage and to detect the leakage at an early stage. European and Japanese utilities have inspected most of the CRDM nozzles and repaired the nozzles or replaced the vessel heads as appropriate. In Japan, the three most susceptible vessel heads are being replaced, even though no cracks were found in the nozzles of these heads. In France, Electricite de France (EDF) is planning on replacing all vessel heads as a preventative measure. Removable insulation on the vessel head and leakage monitoring systems are installed at French and Swedish plants for early detection of leakage. The NRC staff concluded that VH penetration cracking does not pose an immediate or near term safety concern. A 11/19/93 NRC safety evaluation is referenced which states that the staff recommends that NUMARC (NEI) consider enhanced leakage detection by visually examining the reactor vessel head until either inspections have been completed showing absence of cracking or on-line leakage detection is installed in the head area. The staff believes that it is prudent for NUMARC (NEI) to consider the implementation of an enhanced leakage detection method for detecting small leaks during plant operation. On 3/5/96, NEI submitted a white paper entitled "Alloy 600 RPV Head Penetration.	Regarding inspection activities: 1.1 A description of all inspections of CRDM nozzle and other VH penetrations performed to the date of this generic letter, including the results of these inspections. 1.2 If a plan has been developed to periodically inspect the CRDM nozzle and other VH penetrations, a) provide the schedule for first, and subsequent, inspections of the CRDM nozzle and other VH penetrations, including the technical basis for this schedule, b) provide the scope for the CRDM nozzle and other VH penetration inspections, including the total number of penetrations (and how many will be inspected), which penetrations have thermal sleeves, which are spares, and which are instrument or other penetrations. 1.3 If a plan has not been developed to periodically inspect the CRDM nozzle and other VH penetrations described above, provide the analysis that supports the selected course of action as listed in either 1.2 or 1.3 above. In particular, provide a description of all relevant data and/or tests used to develop crack initiation and crack growth models the methods and data used to validate these models, the plant-specific inputs to these models, and how these models substantiate the susceptibility evaluation. Also, if an integrated industry inspection program is being relied on, provide a detailed description of this program. 2. Provide a description of any resin bead intrusions, as described in IN 96-11, that have exceeded the current EPRI PWR Primary Water Chemistry Guidelines recommendations for primary water sulfate levels, including the following information: 2.1 Were the intrusions cation, anion, or mixed bed? 2.2 What were the durations of these intrusions? 2.3 Does the plant's RCS water chemistry Technical Specifications follow the EPRI guidelines? 2.4 Identify any RCS chemistry excursions that exceed the plant administrative limits for the following species: sulfates, chlorides or fluorides, oxygen, boron, and lithium. 2.5 Identify any conductivity excursions which may be indicative of resin intrusions. Provide a technical assessment of each excursion and any follow-up actions. Respond within 30 days. 2.6 Provide an assessment of the potential for any of these intrusions to result in a significant increase in the probability for IGA for VH penetrations and any associated plan for inspections.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
IN 2001-05	Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3	4/30/01	Alert licensees to the recent detection of through-wall circumferential cracks in two of the control rod drive mechanism penetration nozzles and weldments at the Oconee Nuclear Station, Unit 3. The circumferential crack in the #56 CRDM nozzle was through-wall, and the #50 nozzle had pin hole through-wall indications. These cracks followed the weld profile contour, and were nearly 165 degrees in length. Root cause of the cracking was PWSCC. The nozzles were shrink fit by cooling to at least minus 140 degrees F, inserted into the closure head penetration, and then allowed to warm to room temperature (70 degrees F minimum). The CRDM nozzles were tack-welded and then permanently welded to the closure head using 182-weld metal. The recent identification of significant circumferential cracking of two CRDM nozzles at Oconee 3 raises concerns about a potentially risk-significant condition affecting all domestic PWRs. Further, the environment in the CRDM housing annulus will likely be far more aggressive after any through-wall leakage, because potentially highly concentrated borated primary water will become oxygenated, increasing crack growth rates. The Oconee 3 cracking reinforces the importance of examining the upper PWR RPV head area (e.g., visual under-the-insulation examinations of the penetrations for evidence of borated water leakage or volumetric examinations of the CRDM nozzles) and of using appropriate NDE methods to adequately characterize cracks.	None required.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
BL 2001-01	Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles	8/3/01	The purpose of the bulletin is to request that addresses provide information related to the structural integrity of the reactor pressur vessel head penetration nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements, and require that all addresses provide to the NRC a written response. The Bulletin recaps thru-wall circumferential cracking experienced at Oconee 3. As a remedial measure, the RPV head may have to be cleaned at a prior outage for effective identification of new deposits from VH penetration nozzle cracking if new deposits cannot be discriminated from existing deposits from other sources. The recently identified CRDM nozzle degradation phenomena raise several issues regarding the resolution approach taken in GL 97-01: 1) Cracking of Alloy 182 weld metal has been identified in CRDM nozzle J-groove welds for the first time. The finding raises an issue regarding the adequacy of cracking susceptibility models based only on the base metal conditions. 2) Cracking at ANO 1 raises an issue regarding the adequacy of the industry's GL 97-01 susceptibility model. 3) circumferential cracking of CRDM nozzles, located outside of any structural retaining welds, has been identified for the first time. This concern raises concerns about the potential for rapidly propagating failure of CRDM nozzles and control rod ejection, causing a loss of coolant accident. 4) Circumferential cracking from the CRDM nozzle OD to the ID has been identified for the first time. This finding raises concerns about increased consequences of secondary effect of leakage from relatively benign axial cracks, 5, Circumferential cracking of CRDM nozzles was identified by the presence of relatively small amounts of boric acid deposits. This finding increases the need for more effective inspection methods to detect the presence of degradation in CRDM nozzles before the nozzle integrity is compromised. The Bulletin cites several GDC criteria (14, 31, 32), 10CFR50.55a, and Appendix B, Criteria V, IX, and XVI that may not be fully adhered to.	Requests the following: 1. All addressees: 1a) the plant-specific susceptibility ranking using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2 report, 1b) a description of the VH penetration nozzles, including the number type, inside and outside diameter, materials of construction, and the minimum distance between VH penetration nozzles, 1c) a description of the RPV head insulation type and configuration, 1d) a description of the VH penetration nozzle and RPV head inspections (type, scope, qualification requirement, ad acceptance criteria) that have been performed in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations. 2. If your plant has previously experienced either leakage from or cracking in VH penetration nozzles, provide the following: 2a) a description of the extent of VH penetration leakage and cracking, including the number, location, size and nature of each crack detected, 2b) a description of the additional or supplemental inspections (type, scope, qualification requirements, and acceptance criteria), repairs and other corrective actions you have taken in response to identified cracking to satisfy applicable regulatory requirements, 2c) plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule, 2d) basis for concluding that the inspections identified in 2c will assure that regulatory requirements are met. Include the following: 2d(1) If your future inspections plans do not include performing inspections before 12/31/01, provide your basis for concluding that the regulatory requirements will continue to be met until the inspections are performed, 2d(2) If your future inspection plans do not include volumetric examination of all VH penetration nozzles, provide your basis for concluding that the regulatory requirements will be satisfied, 3) If the susceptibility ranking for your plant is within 5 EFPY of ONS3, addresses are requested to provide the following: 3a) plans for future inspections and the schedule, 3b) basis for concluding that the inspections identified in 3a will assure that regulatory requirements are met. Include the following specific information: 3b(1) If your future inspection plans do not include performing inspections before 12/31/01, provide your basis for concluding that the regulatory requirements will continue to be met until the inspections are performed, 3b(2) If your future inspection plans include only visual inspections, discuss the corrective actions that will be taken, including alternative inspection methods if leakage is detected.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
BL 2001-01 (cont.)				4. If the susceptibility ranking for your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following: 4a) plans for future inspections and schedule, 4b) basis for concluding that the inspections identified in 4a will assure that regulatory requirements are met. Include the following specific information : 4b(1) If your future inspection plans to not include a qualified visual examination at the net scheduled refueling outage, provide your basis for concluding that the regulatory requirements will continue to be met until the inspections are performed, 4b(2) Corrective actions that will be taken, including alternative inspection methods if leakage is detected. 5) Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage: 5a) a description of the extent of VH penetration nozzle leakage and cracking detected at your plant, including the number, location size, and nature of each crack detected, 5b) if cracking is identified, a description of the inspections, repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
IN 2002-11	Recent Experience with Degradation of Reactor Pressure Vessel Head	3/12/02	To inform addressees about findings from recent inspections and examinations of the reactor pressure vessel head at Davis-Besse Nuclear Power Station. Recaps previous generic communication information about boric acid on the RPV head at Davis-Besse. Visual inspections in 1998 showed an even layer of boric acid deposits scattered over the RPV head (including deposits near CRDM nozzle 3). This indicated to the licensee that the boric acid evident on the head flowed downward from leakage in the CRDM flanges. During a refueling outage in 2000, the licensee also performed visual inspections of the CRDM flanges and nozzles. Above the RPV head insulation, those inspections revealed five CRDM flanges with evidence of leakage, including one flange that was the principal leakage point. All of the leaking flanges were repaired by replacing their gaskets. Visual inspections performed below the RPV head insulation during the 2000 refueling outage indicated some accumulation of boric acid deposits on the RPV head. No visible evidence of CRDM nozzle leakage (i.e., leakage from the gap between the nozzle and the RPV head) was detected. The licensee described that the RPV head area was cleaned with demineralized water to the greatest extent possible, while trying to maintain the dose as low as reasonably achievable (ALARA). Subsequent video inspection of the partially cleaned RPV head and nozzles was performed for future reference. A subsequent review of the 1998 and 2000 inspection video tapes in 2001 confirmed that there was no evidence of leakage from the RPV head nozzles, although many areas of the RPV head were not accessible because of persistent boric acid deposits that the licensee did not clean because of ALARA issues (including the region around nozzle 3). The inspections in 2002 did not reveal any visual evidence of flange leakage from above the RPV head. However, three CRDM nozzles had indications of cracking (identified by ultrasonic testing of the nozzles), which could result in leakage from the RPV to the top of the RPV head.	None required.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
BL 2002-01	Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity	3/18/02	The purpose of the bulletin is to require PWR addressees to submit (1) information related to the integrity of the reactor pressure boundary including reactor pressure vessel head and the extent to which inspections have been undertaken to satisfy applicable regulatory requirements, and (2) the basis for concluding that plants satisfy applicable regulatory requirements related to the structural integrity of the reactor coolant pressure boundary and future inspections will ensure continued compliance with applicable regulatory requirements, and (3) a written response to the NRC if they are unable to provide the information or they can not meet the requested completion dates. Recaps past generic communications and experience at Davis-Besse. A past model where boric acid crystals are assumed to accumulate on the RPV head, the deposits were assumed to cause minimal corrosion while the reactor was operating because the temperature of the RPV head is above 500 °F during operation, and dry boric acid crystals are not very corrosive. Therefore, wastage was typically expected to occur only during outages when the boric acid could be in solution, such as when the temperature of the RPV head falls below 212°F. These findings at Davis-Besse bring into question the reliability of this model. Inspections performed to date at plants with high and moderate susceptibility have generally confirmed the ability of the model to predict a plant's relative susceptibilities, however, a plant with a ranking of 14.3 effective full power years from the Oconee 3 condition (at the time when circumferential cracking was identified at Oconee 3 in March 2001) identified three nozzles with cracking, other plants with fewer effective full-power years from the Oconee 3 condition did not identify cracking. Some inspection and repair methods may not have been capable of identifying the presence of a void in the carbon steel head adjacent to the cladding interface.	1. Within 15 days of the date of the bulletin, all PWR addressees are required to provide the following: A) a summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at their plants, B) an evaluation of the ability of their inspection and maintenance programs to identify degradation of the RPV head including, thinning, pitting, or other forms of degradation such as the degradation of the RPV observed at Davis-Besse, C) a description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1A that could have led to degradation and the corrective actions taken to address such conditions, D) schedule, plans, and basis for future inspections of the RPV head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria, and E) conclusions regarding whether there is reasonable assurance that regulatory requirements are currently being met. If the evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met, discuss plans for plant shutdown and inspection. If the evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements will continue to be met until the inspections are performed. 2. Within 30 days after plant restart following the next inspection of the RPV head to identify any degradation, all PWR addressees are required to submit to the NRC the following information: A) the inspection scope and results, including the location, size, and nature of any degradation detected, and B) the corrective actions taken and the root cause of the degradation. 3. Within 60 days of the date of this bulletin, all PWR addressees are required to submit to the NRC the following information related to the remainder of the reactor coolant pressure boundary: A) the basis for concluding that their boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and this bulletin. If a documented basis does not exist, provide your plans, if any for a review of your programs. Within 7 days of the date of the bulletin, a PWR addressee is required to submit a written response if they are unable to provide the information or they can not meet the requested completion dates. Alternative courses of action and their basis must be provided.

Table 5.1-1 (cont.)
Summary of Documented Leaks from NRC Bulletins

NRC Generic Communications Involving Boric Acid Leakage and Corrosion Issued from 1980 Through the First Quarter of 2002

Generic Com	Title	Issue Date	Abstract	NRC Information Requests
IN 2002-13	Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradation	4/4/02	To alert addressees to possible indicators of RPV boundary degradation including degradation of the RPV head material. These indicators include unidentified reactor coolant system leakage and containment air cooler and radiation element filter fouling. Containment air coolers cleaning of boron deposits greatly increased. The licensee noticed that deposits removed from CAC 1 exhibited a rust-like color. The licensee attributed the discoloration to migration of the surface corrosion on the CACs into the boric acid deposits and to the aging of the boric acid deposits. During the 2002 outage, fifteen 5-gallon buckets of boric acid were removed from the CAC ductwork and plenum A flow from the CACs also resulted in boric acid deposits elsewhere within containment including on service water piping, stairwells, and other areas of low ventilation. The radiation element filters accumulate particulates and may need to be changed to ensure acceptable system operation. Licensee records correlate RB filter changes with past RCS leakage increases. In March 1999, RE filter clogging from boric acid deposits was identified and attributed to the pressurizer relief valve modification. In November 1999, after identifying yellowish brown deposits in the filters, the licensee obtained a chemical analysis of the filter particulates which identified the presence of ferric oxide in addition to boric acid crystals. Around that time, the licensee began changing the filters every one-to-three weeks. By November 1999, the frequency of filter changes had again increased.	None required.

ATTACHMENT 5.2

ALLOY 600 AND ALLOY 82/182 POTENTIAL LEAK LOCATIONS IN THE PRIMARY COMPONENTS OF WESTINGHOUSE UNITS

Table 5.2-1
Alloy 600 and Alloy 82/182 Locations in Westinghouse Primary Pressure Boundary Components, Prioritized on the Basis of Susceptibility to Boric Acid Corrosion Wastage (see Figure 5.2-1)

Alloy 600/Alloy 82/182 Location Description (See Figure 5.2-1)	Affected Carbon Steel Component Susceptible to BAC	Wastage Potential (H/M/L)	Comments
CRDM motor housing	Vessel upper head	H	
CRDM nozzles to RV head welds	Vessel upper head	H	
Head vent pipe	Vessel upper head	H	
Monitor tube	Vessel head/shell	L	These are isolated from the RCS by the inner o-ring, and therefore should be low due to low consequences.
Core support block	N/A	L	
Instrument tubes	Vessel bottom head	M	
RV nozzle-pipe weld	Vessel nozzle	H	
Surge nozzle-pipe weld	Pressurizer lower head & surge nozzle	H	
Spray nozzle-pipe weld	Pressurizer upper head & spray nozzle	H	
Safety & relief nozzle-pipe welds	Pressurizer upper head & safety & relief nozzles	H	
Heat transfer tubing	N/A	L	
Tubesheet (TS) cladding	Steam generator bottom channel head & tube sheet	L	
Tube -TS cladding weld	N/A	L	
Partition plate & welds	N/A	L	
Primary nozzle closure rings & welds	Steam generator bottom channel head & nozzle	H	
Bottom channel head drain tube & welds	Steam generator bottom channel head	M	Not applicable to all Westinghouse vessels*
SG nozzle-pipe weld	Steam generator bottom channel head & nozzle	H	Not applicable to all Westinghouse vessels*
*Note: Based on individual plant design and vintage, these locations may not have Alloy 600 or Alloy 82/182 weld materials.			

ATTACHMENT 5.3

ALLOY 600 AND ALL0Y 82/182 POTENTIAL LEAK LOCATIONS IN THE PRIMARY COMPONENTS OF COMBUSTION ENGINEERING UNITS

Attachment 5.3
Potential BAC Susceptible Alloy 600 and Alloy 82/182 Leak Locations in the Primary Components of Combustion Engineering PWR Plants

In the Combustion Engineering (CE) design plants, Alloy 600 nozzles are generally located in the reactor vessel top heads, in the steam generator primary head, in the pressurizer top head, bottom head and lower shell, and in the main coolant piping. There are some minor deviations in these locations. In 1 plant there are no Alloy 600 nozzles in the main coolant piping, and in 3 other plants there are Alloy 600 nozzles in the reactor vessel bottom head. These nozzles are attached to the components by partial penetration welds with Alloy 182 or 82 weld metals. For all nozzles, the welds were originally at the component inside surfaces.

In addition to the nozzles with partial penetration welds, Alloy 600 weld metals 82 and/or 182 were used in various bi-metallic welds in the primary system. These are located in the main coolant piping to reactor coolant pump safe-ends, in the various tributary lines that have stainless steel safe-ends that are welded to carbon or low-alloy steel nozzles, and between the pressurizer safety and relief valve nozzles and the valves.

Beginning in 1986, some of the Alloy 600 nozzles in CE plants developed leakage as a result of Primary Water Stress Corrosion Cracking (PWSCC). The leaking nozzles were replaced with similar nozzles of Alloy 600. The replacement nozzles also developed PWSCC, which led to the use of Alloy 690 for future replacement nozzles. Originally, the weld metals remained the same but later Alloy 152 or Alloy 52 weld metals (more similar to the Alloy 690) replaced the 82/182 welds. Although more resistant to PWSCC (no documented occurrences in the field or in the laboratory of PWSCC cracks in Alloy 690), Alloy 690 applications are currently treated the same as Alloy 600 applications with respect to borated water leakage and the potential for boric acid corrosion (BAC). Locations where significant numbers of replacements of Alloy 600 with Alloy 690 are discussed below.

The following paragraphs discuss the potential for boric acid corrosion as a result of leakage through Alloy 600 nozzles and welds.

1. Corrosion Potential in Pressurizers

The pressurizers were fabricated from low alloy steels and included Alloy 600 instrumentation nozzles in the top head (typically 4), lower shell (typically 1) and bottom head (typically 2), and heater sleeves (30 to 120, depending on the plant and the size of the heaters) in the bottom head. The pressurizers have a history of nozzle and heater sleeve leakage. Because the operating temperatures are significantly higher in the pressurizers, PWSCC would be expected in the pressurizers before any other components in the Reactor Coolant System (RCS). This has been the case. Leakage from a nozzle would exit the crevice and deposits boric acid on the pressurizer outside surface. If the deposits were wet, or if water from the escaping steam were to collect on the pressurizer surfaces, boric acid corrosion could occur. Laboratory data indicate that significant corrosion could occur. In one instance at Arkansas Nuclear One (ANO) Unit 2, leakage from a heater sleeve resulted in a cavity approximately 1.5 inches in diameter and 0.75-inch deep being created in a relatively short period of time in the bottom head of the pressurizer. There have been numerous other events of leakage from nozzles and heater sleeves, but similar boric acid corrosion degradation of the pressurizer has not occurred. Nevertheless, the pressurizer has the greatest potential for corrosion of any RCS component because of the high operating temperature.

Because of leakage from nozzles or sleeves, nozzles or sleeves have been replaced or other mitigative actions taken. In three plants, all instrumentation nozzles were replaced. In four additional plants, the nozzles were replaced or Mechanical Nozzle Seal Assemblies (MNSAs) applied. In three plants, some of the nozzles were replaced. In one plant, heater sleeves were replaced and in two plants, the heater sleeve Inside Diameter (ID) surface in the area of the weld were nickel plated to reduce the potential for PWSCC. The nickel-plating process does not reduce the potential of leakage through the welds, but does reduce the potential for PWSCC in the sleeves. These plants all have reduced potential for leakage, and thus reduced potential for boric acid corrosion of the pressurizer.

Bi-metallic weld locations in the pressurizer include those between the surge nozzle and surge line, the spray nozzle and spray line, and the safety and relief valve nozzles and valves. The lines or valves are austenitic stainless steels. The high temperatures in the pressurizer would indicate the bi-metallic welds in the pressurizer have the greatest potential for cracking, although cracking has occurred in only one such application to date. Any leakage from the weld could wet the nozzle material, resulting in corrosion. As a result, the potential for corrosion is considered high.

2. Reactor Vessel Upper Head

In the CE plants, the Control Rod Drive Mechanism (CRDM)/Control Element Drive Mechanism (CEDM) nozzles, the In-Core Instrumentation (ICI) nozzles, and the vent lines are Alloy 600 and are attached to the reactor vessel head by Alloy 82/182 partial penetration welds. In CE plants, there have not yet been any occurrences of leakage as a result of PWSCC in any of these nozzles or welds. Volumetric inspections of over 500 CEDM nozzles, 35 ICI nozzles, 6 vent lines and surface examinations of approximately 59 CEDM welds, and visual examinations of the outer surfaces of 4 additional heads have not detected leakage from any nozzles and only shallow cracks below the partial penetration welds in 3 nozzles. The inspection data suggest reduced potential for PWSCC in upper head nozzles in CE plants as compared to some other designs. However, the large number of smaller diameter Alloy 600 nozzles, which have also cracked, suggests that eventually PWSCC could occur in the head nozzles. The heads were fabricated from low alloy steels and would be susceptible to boric acid corrosion if leakage were to develop. For this reason, the potential for boric acid corrosion of the upper head regions in CE plants is considered high.

There are 2 bi-metallic welds in some of the CEDM housings in CE plants. These welds are between the lower and upper end fittings (stainless steel or Alloy 600) and the motor tube buttering. The temperatures are so low that PWSCC is not an issue. Therefore, the potential for boric acid corrosion is very low.

3. Reactor Vessel Lower Head

In 3 CE plants, there are 61 ICI nozzles in the bottom head. These are Alloy 600 procured to the same requirements as other Alloy 600 nozzles and are attached to the bottom head by partial penetration welds with Alloy 182/82 weld metals. These nozzles are exposed to temperatures at or near cold-leg temperatures (currently 555°F). For this reason, the potential for PWSCC is significantly reduced. There have not been any occurrences of Alloy 600 nozzle PWSCC at cold-leg temperatures. Although cracking at such low temperatures is a low-probability event, it cannot be completely discounted. Should PWSCC of the ICI nozzles occur, the bottom head, which is also low alloy steel, would be wetted by borated water and boric acid corrosion of the head could occur. However, because of the low probability of PWSCC of the nozzles. the potential for boric acid corrosion is considered low.

4. Steam Generator Primary Heads

The primary heads of the steam generators in CE plants that were supplied by CE have 4 Alloy 600 instrument nozzles. Replacement steam generators may not have these nozzles or the nozzles may be Alloy 690. The Alloy 600 nozzles were procured to the same requirements as the other Alloy 600 nozzles and were attached to the primary head by partial penetration welds with Alloys 182/82 weld metal. These nozzles are exposed to cold-leg temperatures and less susceptible to PWSCC than nozzles at higher temperatures. The primary heads were fabricated from low alloy steels.

Since the potential for PWSCC is reduced in the nozzles, the potential for boric acid corrosion of the primary heads is considered low.

5. Main Loop Piping

The main loop piping in all but 1 CE plant is carbon steel that is clad with stainless steel on the ID surfaces. There are varying numbers of Alloy 600 nozzles in the piping of the different plants. For example, Calvert Cliffs Unit 1 has 10 Resistance Temperature Detector (RTD) nozzles and 9 pressure measurement or sampling nozzles in the hot legs and 12 RTD nozzles in the cold legs (total of 31). These are attached to the inside surfaces of the piping by partial penetration welds with Alloy 182/82 weld metal. One CE plant has stainless steel piping and no Alloy 600 nozzles in the piping.

Numerous nozzles in the hot legs at several plants have leaked as a result of PWSCC. There have been no cold-leg failures. Two plants have replaced all hot- and cold-leg nozzles with Alloy 690 nozzles and Alloy 52/152 weld metals. At cold-leg temperatures, PWSCC is a low potential event. Therefore, boric acid corrosion is a low potential event. In the unlikely event that leakage does occur, boric acid corrosion is a possibility.

At hot-leg nozzle locations that have not been replaced, there is a high potential for leakage as a result of PWSCC and, therefore, a high potential for some level of boric acid corrosion.

Within the main coolant piping in all CE units (except Fort Calhoun), the only bi-metallic welds are between the cold-leg piping and the reactor coolant pump nozzle safe-ends. Because they are exposed to cold-leg temperatures, PWSCC of these welds is unlikely because they experience cold-leg temperatures. Primary coolant leakage from the welds could wet the carbon steel piping, resulting in boric acid corrosion. However, since leakage from the welds is a low-potential event, corrosion of the carbon steel piping is also considered to be low potential.

The tributary lines also connect to the piping and have stainless steel safe-ends welded to carbon steel nozzles and have bi-metallic welds (182/82). The surge line and shutdown cooling system connect to the hot leg. Because of the temperature, these nozzles are considered to have a high potential for leakage because of PWSCC although there have yet to be any occurrences of PWSCC in these welds. If leakage were to occur, the carbon steel nozzle, and possibly the main coolant piping, could be wetted and experience boric acid corrosion. As a result, these locations are considered to have a high potential for boric acid corrosion.

The other tributary lines (safety injection, drain, charging/letdown, and spray) connect to the cold leg, and the potential for PWSCC is low.

Table 5.3-1 (see Figure 5.3-1)
Potential for Boric Acid Corrosion in CE Plants by Location

Component	Location	Part	Potential
Pressurizer	Top Head	Instrument Nozzles	High
		Inst. Nozzle Partial Penetration (PP) Welds	High
		Safety Valve Welds	High
		Spray Nozzle Welds	High
		Relief Valve Welds	High
	Side Shell	Instrument Nozzles	High
	Bottom Head	Instrument Nozzles	High
		Heater Sleeves	High
		Inst. Noz. & Sleeve PP Welds	High
		Surge Nozzle Welds	High
	All	Alloy 690 Replacement Noz.	Very Low
		Alloy 52/152 PP Welds	Very Low
Reactor Vessel	Top Head	CEDM/ICI Nozzles	Mod High
		Vent Line	Mod High
		PP Welds	Mod High
		CEDM Housing Welds	Very Low
	Bottom Head	ICI Nozzles	Low
	(3 Plants Only)	PP Welds	Low
Steam Generator		Instrument Nozzles	Low
		PP Welds	Low
Main Loop Piping	Hot Leg	Instrument Nozzles	High
		PP Welds	High
		Surge Nozzle Weld	High
		SD Cooling Nozzle Welds	High
	Cold Leg	SI Nozzle Welds	Low
		Drain Nozzle Welds	Low
		Charging/Letdown Noz Welds	Low
		Spray Nozzle Welds	Low
	All	A690 Replacement Nozzles	Very Low
		52/152 Welds	

ATTACHMENT 5.4

ALLOY 600 AND ALLOY 82/182 POTENTIAL LEAK LOCATIONS IN THE PRIMARY COMPONENTS OF BABCOCK AND WILCOX (B&W) PWR PLANTS

Attachment 5.4
Potential BAC Susceptible Alloy 600 and Alloy 82/182 Leak Locations in the Primary Components of Babcock and Wilcox PWR Plants.

Ranking Criteria Utilized

PWSCC Susceptibility

High = Relative time-to-10% cracking (t/t_{ref}) ≤ 65

Medium = $t/t_{ref} \leq 300$ (generally) but some locations ~1000

Low = remainder

(Reference: "Alloy 600 PWSCC Susceptibility Model," FRA-ANP Document 51-5001951-01, December 1998, B&W Owners Group Proprietary.)

Boric Acid Corrosion Potential

A boric acid corrosion potential exists, since carbon or low-alloy steel material is utilized in the vicinity of potential leakage from PWSCC of the nearby Alloy 600 component item or weld.

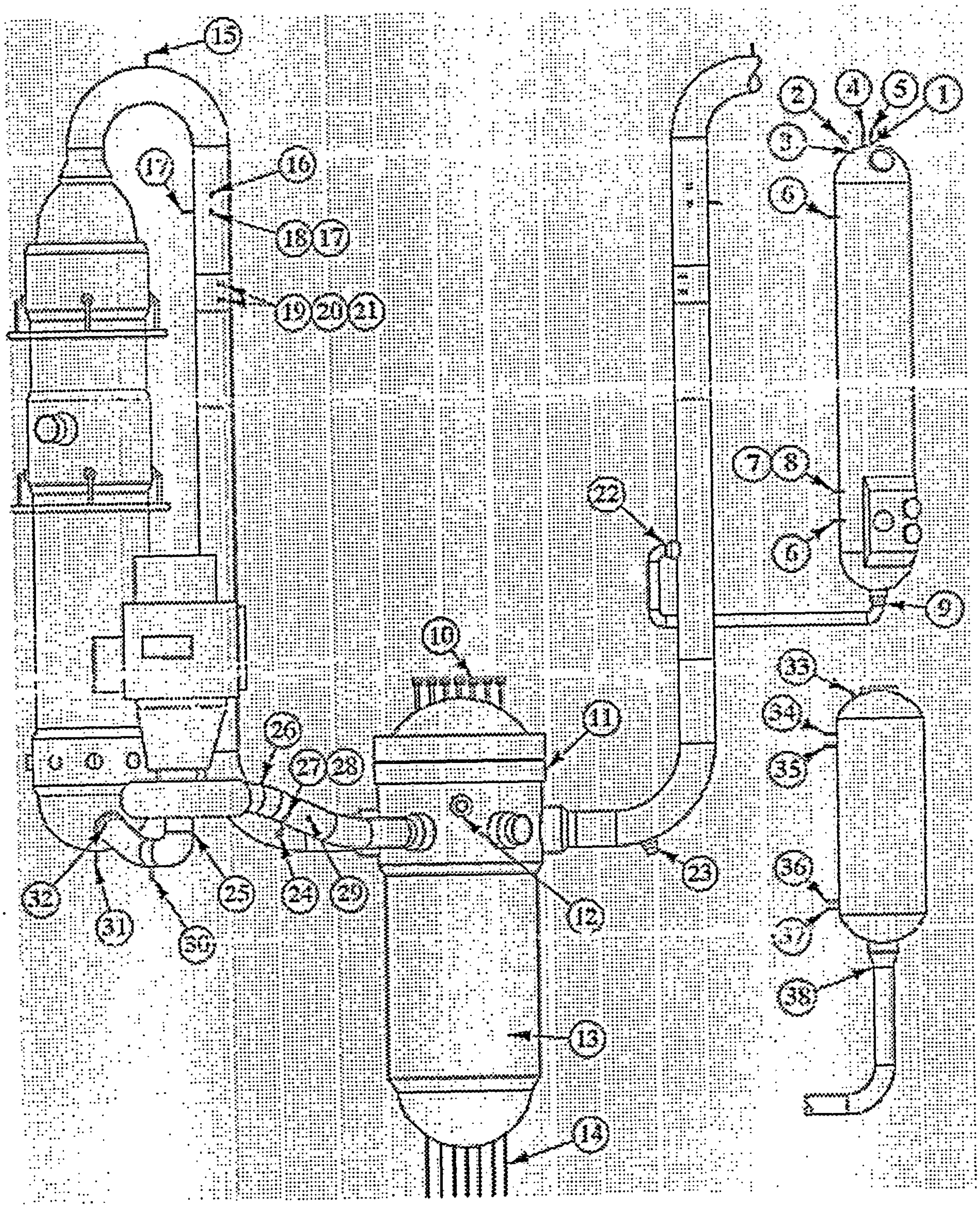


Figure 5.4-1. General Locations of Alloy 600 Type Materials in the B&W (177-FA Design) Reactor Coolant System (Prepared by DEI)

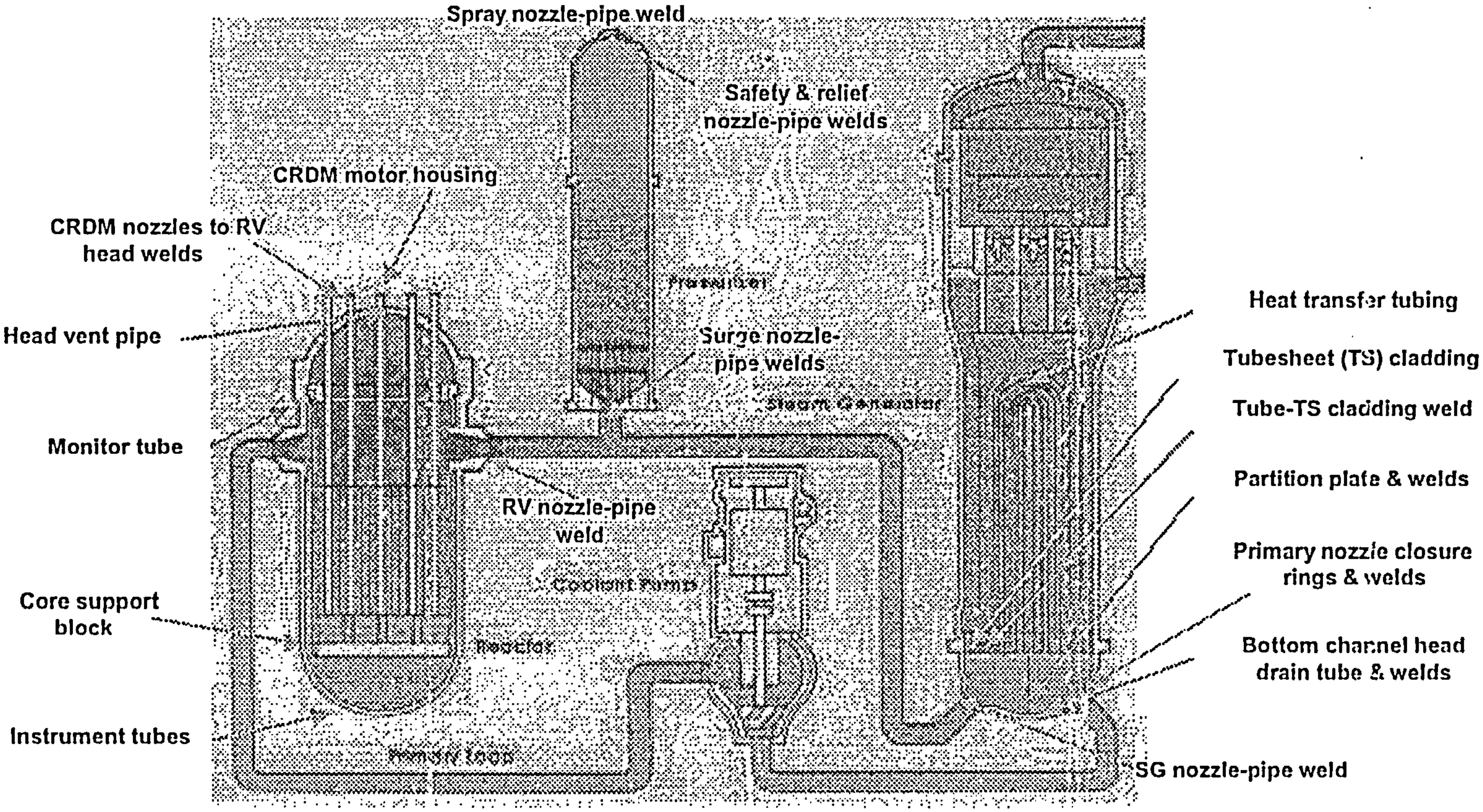


Figure 5.2-1. Alloy 600 and Alloy 82/182 Locations in the Primary Pressure Boundary Components of Westinghouse PWR Units

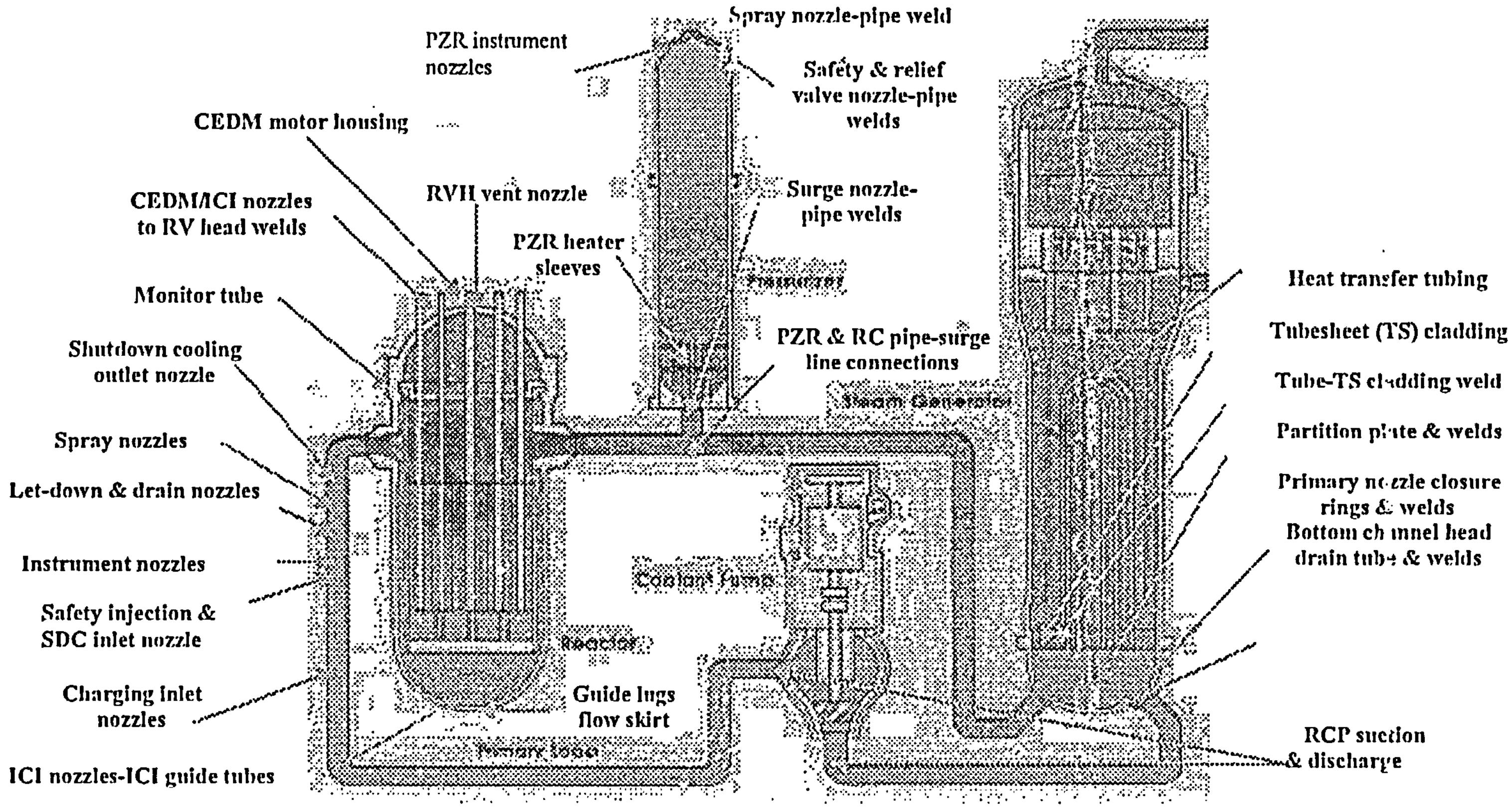


Figure 5.3-1. Alloy 600 and Alloy 82/182 Locations in the Primary Pressure Boundary Components of CE PWR Units

Table 5.4-1
Boric Acid Corrosion Potential from PWSCC of Typical Alloy 600 and Alloy 82/182 Component Locations in the Reactor Coolant System in B&W Plants

Component	Location (See Figure 5.4-1)	Component Item Description[a]	Alloy 600/82/182 PWSCC Susceptibility (H/M/L)	Carbon or Low-Alloy Steel Nearby?
Reactor Vessel	Closure Head (10)	CRDM Nozzles/Welds	High	Yes
	Closure Head (10)	Thermocouple Nozzles/Welds	High	Yes
	Closure Head (10)	CRDM Motor Tube Cladding/Welds	Low	Yes
	Lower Head (14)	In-core Monitoring Instrumentation Nozzles/Welds	Medium	Yes
	Lower Shell (13)	Core Guide Lugs/Welds	Low	N/A[c]
	Upper Shell (12)	Core Flood Nozzle-SE Welds	Medium	Yes
	Upper Shell (11)	Monitor Tap Welds	Medium	Yes
Steam Generator	Lower Head (31)	Primary Drain Nozzles/Welds	Medium	Yes
	Lower Head (32)	Cold Leg Nozzle Dam Rings/Welds	Low	N/A
	Upper and Lower Head	Tube Sheet Clad	High	N/A
	Upper and Lower Head	Tube Sheet - Tube Welds	High	N/A
Pressurizer	Upper and Lower Shell (3, 5-8)	Instrument Nozzles/Welds	High	Yes
	Upper and Lower Shell (3, 5-8)	Instrument Nozzle Safe Ends/Welds	High	Yes
	Upper Head (1)	Vent Nozzles/Welds	High	Yes
	Upper Head (1, 2)	Vent and Spray Nozzle Safe Ends/Welds	High	Yes
	Upper Head (4)	Pressure Relief Nozzle Welds	High	Yes

Table 5.4-1 (cont.)
Boric Acid Corrosion Potential from PWSCC of Typical Alloy 600 and Alloy 82/182 Component Locations in the Reactor Coolant System in B&W Plants

Component	Location (See Figure 5.4-1)	Component Item Description[a]	Alloy 600/82/182 PWSCC Susceptibility (H/M/L)	Carbon or Low-Alloy Steel Nearby?
	Lower Head (9)	10" Surge Nozzle-SE Welds	High	Yes
	Heater Bundle	Diaphragm Plates/Welds	High	Yes
	Heater Bundle	Heater Sleeves/Welds	High	N/A
Reactor Coolant Piping	Upper Cold Leg (27)	Instrument Safe Ends/Welds	Medium	Yes
	Upper Cold Leg (29)	HPI/MU Nozzle Welds	Medium	Yes
	Upper Cold Leg (26)	Piping-RC Pump Welds	Medium	Yes
	Lower Cold Leg	Instrument Safe Ends/Welds	Medium	Yes
	Lower Cold Leg	Instrument Nozzles/Welds	Medium	Yes
	Lower Cold Leg (30)	Drain Nozzle Safe Ends/Welds	Medium	Yes
	Lower Cold Leg (30)	Drain Nozzles/Welds	Medium	Yes
	Lower Cold Leg (28)	RTE Mounting Bosses/Welds	Medium	Yes
	Lower Cold Leg (25)	Piping-RC Pump Welds	Medium	Yes
	Hot Leg (15, 17-21, 24)	Instrument Safe Ends/Welds	High	Yes
	Hot Leg (15, 17-21, 24)	Instrument Nozzles/Welds	High	Yes
	Hot Leg (16)	RTE Mounting Bosses/Welds	High	Yes
	Hot Leg (23)	Decay Heat Nozzle Welds	Medium	Yes
	Hot Leg (22)	Surge Nozzle Welds	High	Yes

Table 5.4-1 (cont.)
Boric Acid Corrosion Potential from PWSCC of Typical Alloy 600 and Alloy 82/182 Component Locations in the Reactor Coolant System in B&W Plants

Component	Location (See Figure 5.4-1)	Component Item Description[a]	Alloy 600/82/182 PWSCC Susceptibility (H/M/L)	Carbon or Low-Alloy Steel Nearby?
Core Flood Tank	Top/Bottom (34-37)	Instrument Nozzle Safe Ends/Welds	Low	Yes
	Top/Bottom (34-37)	Instrument Nozzles/Welds	Low	Yes
	Top (33)	Pressure Relief Nozzle Safe Ends/Welds	Low	Yes
	Bottom (38)	Outlet Nozzle Welds	Low	Yes

Notes:

a. Each of these component items and welds are not present at all of the individual 177-FA plants.

b. Individual plant component item and weld rankings vary.

c. Not applicable; component item and/or weld is internal to the RCS and is not a pressure boundary location.

ATTACHMENT 5.5

TYPICAL EXAMPLES OF POTENTIAL LEAK LOCATIONS IN THE AUXILIARY SYSTEMS OF WESTINGHOUSE UNITS

Attachment 5.5
Potential Leak Locations with Boric Acid Corrosion Wastage Significance in the Auxiliary System Components of Westinghouse Design PWR Units

Evaluation Basis

Concentrated boric acid is contained in a limited number of subsystems and components in Westinghouse Pressurized Water Reactor (PWR) plants. Furthermore, most plants have reduced the concentration of boric acid in these systems to minimize the amount of heat tracing required and the problems associated with system maintenance. A representative three-loop Westinghouse PWR plant was selected as the basis of this evaluation. However, the boric acid systems and equipment on which the review was based are typical of PWRs and the results are considered to be generally applicable to the Westinghouse fleet of plants.

Scope of Review

The following systems were reviewed as part of this evaluation:

- Chemical and Volume Control System (CVCS)
  - Boric acid batching and storage
  - Reactor makeup control
  - Emergency boration
  - Charging pump suction
- Boron Recycle System (BRS)

Most PWRs have eliminated or bypassed the Boron Injection Tank (BIT) and the BIT recirculation subsystem in the Safety Injection System (SIS) as permitted by relaxed steamline break accident acceptance criteria. Therefore, the BIT was not included in the scope of this review. Other plant systems contain more dilute boric acid solutions that pose less severe problems in the event of leakage. Therefore, these systems also have been excluded from the review.

Summary of Results

The results of this evaluation are summarized below for each plant subsystem that was reviewed.

1. Chemical and Volume Control System

   The CVCS includes equipment that enables the concentrated boric acid solution to be prepared, stored, and delivered to the Volume Control Tank (VCT) or directly to the suction of the Centrifugal Charging Pumps (CCPs). Concentrated boric acid solution is then injected into the Reactor Control System (RCS) for reactor shutdown, reactivity control, or compensation for xenon transients due to power changes. It has been assumed that concentrated boric acid solution that reaches the suction of the CCPs will be diluted by additional flow from the VCT. Therefore, the CVCS leakage review was terminated at the CCP suction connection. The CVCS components and piping that contain boric acid are fabricated of austenitic stainless steel. Most piping and fitting connections in the CVCS are welded. Welds in Safety-Class piping are periodically inspected for any indication of weld degradation that could result in leakage.

The locations in the CVCS that are susceptible to the leakage of concentrated boric acid solution are summarized in Table 5.5-1. The table includes the following items:

- Identification of the component in which the leak can occur
- A description of the leak location on the component
- An assessment of the potential for leakage to occur (High/Medium/Low)
- The potential for carbon steel or Inconel subcomponents in the immediate vicinity to be exposed to the leakage from the identified leak location (Yes/No)
- A drawing reference for the component

Table 5.5-1 contains a tabulation of potential leakage locations in the CVCS equipment that is used to batch, store, and deliver concentrated boric acid solutions. The majority of the potential leak locations that were identified are flanged connections on the components. The existence of adjacent carbon steel components or piping on which leakage from the CVCS equipment could impinge is dependent on the plant layout. Performing a plant layout review to identify other affected components is beyond the scope of this evaluation.

Tables 5.5-2 and 5.5-4 contain a tabulation of potential leakage locations in the valves and Instrumentation and Control (I&C) equipment of the CVCS and BRS that contain concentrated boric acid solution. Again, most of the potential leak locations that were identified are flanged and bolted connections on the components.

It is noteworthy that the Boric Acid Transfer Pump (BATP) that is included in Table 5.5-1 is a canned motor pump in most plant designs. This design precludes the occurrence of shaft seal leakage through a mechanical seal that is typically found on common centrifugal pump designs. However, the potential for leakage from the flanged suction and discharge pump connections has been noted in Table 5.5-2.

Table 5.5-3 contains a tabulation of various types of valves in the CVCS that contain potential leakage flow paths. However, the packless elastomer diaphragm valves that are used extensively in the concentrated boric acid system are designed to eliminate external leakage and are assumed not to leak.

2. Boron Recycle System

The BRS includes equipment to collect and reprocess borated reactor coolant effluents, relief valve discharges, and equipment drains and leakoffs. These dilute boric acid solutions are reprocessed into concentrated boric acid solution and condensate by the use of the recycle evaporator package. The separated, processed species are then recycled back to the CVCS Boric

Acid Tank (BAT) and Reactor Makeup Water Storage Tank (RMWST) for reuse as reactor coolant makeup.

The BRS components and piping that contain boric acid are fabricated of austenitic stainless steel. Most piping and fitting connections in the BRS are welded. Welds in Safety-Class piping are periodically inspected for any indication of weld degradation that could result in leakage.

The recycle evaporator is a skid-mounted package that includes a steam-powered evaporator drum to boil off some of the water that enters with the effluent feed. The concentrated boric acid solution is pumped out of the evaporator drum, filtered, and returned to the BAT. The portion of the evaporator package and connecting equipment and piping in the BRS that contains the boric acid concentrate was included in the scope of this review.

The locations in the BRS that are susceptible to the leakage of concentrated boric acid solution are summarized in Table 5.5-2. Table 5.5-2 contains a tabulation of potential leakage locations in the BRS equipment that is used to reprocess and transfer concentrated boric acid solutions. The majority of the potential leak locations that were identified are flanged connections on the components. The existence of adjacent carbon steel components or piping on which leakage from the BRS equipment could impinge is dependent on the plant layout. Performing a plant layout review to identify other affected components is beyond the scope of this evaluation.

Table 5.5-3 contains a tabulation of various types of valves in the BRS that contain potential leakage flow paths. However, the packless elastomer diaphragm valves that are used extensively in the recycle system are designed to eliminate external leakage and are assumed not to leak.

Table 5.5-1
Potential Leak Locations in the Westinghouse Auxiliary Systems
(CVCS System)

CONCENTRATED BORIC ACID SYSTEM
Potential Leak Location Evaluation
Based on Typical 4 wt. % B.A. System
(Shearon Harris Plant selected as representative)

System: CVCS (Boric Acid Batching/Storage, Emergency Boration, Reactor Makeup Control and Charging subsystems)

Tag or SPIN No.	Location Description	Potential Leaking Component	Drawing Reference	Leak Potential (H/M/L)	Potential Carbon Steel or Inconel Exposure (Y/N)	Comments
BAT	Overflow connection	3" flange	BMT Dwg. No. 77-D113815-1	M	Undetermined	1. BAT non-W scope 2. Pressure = BAT Elev. Head
BAT	Shell manhole	24" flange	BMT Dwg. No. 77-D113315-1	M	Undetermined	1. BAT non-W scope 2. Pressure = BAT Elev. Head
BABT/ CSATBB	Fill connection	Lid	W Dwg. 1141E15	L	Y (CS Supports)	Location above normal BABT water level
BABT/ CSATBB	Mixer port	6" flange	W Dwg. 11[illegible]1E15	L	Y (CS Supports and Bolts)	Location above normal BABT water level
BABT/ CSATBB	Overflow connection	3" flange	W Dwg. 1141E15	L	Y (CS Supports)	Location above normal BABT water level
BATP/ CSAPBA	Boric Acid Trans. Pump Suction connection	2" flange	Crane Dwgs. B62239, B63000, B63003	L	Y (B7 bolting)	Pressure = BAT (or BABT) Elev. Head
BATP/ CSAPBA	Boric Acid Trans. Pump Discharge connection	1" flange	Crane Dwgs. B62239, B63000, B63003	M	Y (B7 bolting)	Pressure = BAT (or BABT) Elev. Head + BATP TDH
BATP/ CSAPBA	Boric Acid Trans. Pump Bypass orifice	3/4" flange	W Dwg. 1093E63	M	Y (B7 bolting)	Orifice non-W scope
BA Filter/ CSFLBA	Top access	Lid	5Q-20467	M	N	Pressure = BAT Elev. Head + BATP TDH

Note: (Tables 5.5-1, 5.5-2, 5.5-3, 5.5-4)

The criteria employed in the (H, M and L) categorization of the potential for leakage to occur in the Westinghouse and Combustion Engineering design plants is give below:

<u>High</u>: Piping or component connections that are both subject to elevated system operating pressure and have a known history of the occurrence of leakage.

<u>Medium</u>: Piping or component connections that are either subject to elevated system operating pressure or located below the normal water level, as applicable to each defined location.

<u>Low</u>: Piping or component connections that are neither subject to elevated system operating pressure nor located below the normal water level, as applicable to each defined location.

Table 5.5-2
Potential Leak Locations in the Westinghouse Auxiliary Systems
(BRS System)

CONCENTRATED BORIC ACID SYSTEM
Potential Leak Location Evaluation
Based on Typical 4 wt. % B.A. System
(Shearon Harris Plant selected as representative)

System: BRS (Recycle Evaporator Concentrates subsystem)

Tag or SPIN No.	Location Description	Potential Leaking Component	Drawing Reference	Leak Potential (H/M/L)	Potential Carbon Steel or Inconel Exposure (Y/N)	Comments
Recycle Evap./ BREVRE	Evaporator Drum Manway	16" flange	W HTD Dwg. 731J452	L	Y (CS bolting and supports)	Location above normal water level
Recycle Evap./ BREVRE	Concentrates Pump: Suction Conn.	3" flange	Crane Dwgs. B-62133, B-66460	L	Y (B7 bolting)	1. Pump part of R.E. pkg. 2. Pressure = R.E. Evap. Drum Pressure
Recycle Evap./ BREVRE	Concentrates Pump: Discharge Conn.	1-1/2" flange	Crane Dwgs. B-62133, B-66460	M	Y (B7 bolting)	1. Pump part of R.E. pkg. 2. Pressure = R.E. Evap. Drum Pressure + RECP TDH
R.E. Conc. Filter/ BRFLCN	Top access	Lid	5EHD 10601-002-EG32	M	N	Pressure = R.E. Evap. Drum Pressure + RECP TDH

Table 5.5-3
Potential Leak Locations in the Westinghouse Auxiliary Systems
(Valves)

CONCENTRATED BORIC ACID SYSTEM
Potential Leak Location Evaluation
Based on Typical 4 wt. % B.A. System
(Shearon Harris Plant selected as representative)

Components: Valves - AOVs, MOVs, Manual Gates, Globes, Checks (See Note 1)

Tag or SPIN No.	Location Description	Potential Leaking Component	Drawing Reference	Leak Potential (H/M/L)	Potential Carbon Steel or Inconel Exposure (Y/N)	Comments
2-TM78FN	MOV in Emergency Boration line	Stem Packing, Valve Bonnet Flange	E73-020*	M	Y	1. Pressure = BAT Elev. Head + BATP TDH 2. See Note 2
8-GM72FB	MOV in charging pump suction header	Stem Packing, Valve Bonnet Flange	8376D29	M	Y	1. Pressure = BAT Elev. Head + BATP TDH 2. See Note 2
2-RA42DD	Conc. BA FCV in RMCS	Stem Packing, Valve Bonnet Flange	D-166834	M	Y	1. Pressure = BAT Elev. Head + BATP TDH 2. See Note 2
6-G72	Charging pump suction header isolation valve	Stem Packing, Valve Bonnet Flange	8378D39	M	Y	1. Pressure = BAT Elev. Head + BATP TDH 2. See Note 2
2-C58	BATP discharge line	Bolted topworks	W-D-9911-(3)*	L	Y	1. Pressure = BAT Elev. Head + BATP TDH 2. Inconel spring (See Note 3)
2-C58	R.E. concentrates line check valves	Bolted topworks	W-D-9911-(3)*	L	Y	1. Pressure = R.E. Evap. Drum Pressure + RECP TDH 2. Inconel spring (See Note 3)
2-C78	Emergency Boration line check valve	Bolted topworks	W-D-9911-(2)*	L	Y	1. Pressure = BAT Elev. Head + BATP TDH 2. Inconel spring (See Note 3)
3-C52	BABT to BATP suction line, BAT gravity drain line check valves	Bolted topworks	8378D19	L	Y	1. Pressure = BAT (or BABT) Elev. Head 2. Inconel disc arm (See Note 3)
			* Beaver Valley 2			

NOTES: 1. All packless elastomer diaphragm valves in the concentrated boric acid system are designed to eliminate external leakage and are assumed not to leak.
2. The upper structures (yoke, etc.) of MOVs and AOVs are likely to contain carbon steel.
3. Internal valve part; leakage not required for exposure to boric acid to occur.

Table 5.5-4
Potential Leak Locations in the Westinghouse Auxiliary Systems
(Instrumentation and Control Systems)

CONCENTRATED BORIC ACID SYSTEM
Potential Leak Location Evaluation
Based on Typical 4 wt. % B.A. System
(Shearon Harris Plant selected as representative)

Components: Instrumentation and Control Equipment

Tag or SPIN No.	Location Description	Potential Leaking Component	Drawing Reference	Leak Potential (H/M/L)	Potential Carbon Steel or Inconel Exposure (Y/N)	Comments
TIS-100	Boric Acid Batching Tank Thermowell connection	1-1/2" flanges	W Dwg. 1141E15	L	Y - CS bolts	Pressure = BABT Elev. Head
LIS-101	Boric Acid Batching Tank Level instrument tap	3" flanges	W Dwg. 1141E15	L	Note 1	Pressure = BABT Elev. Head
FE-110	Emergency Boration flow instrument	2" flanges	W Dwg. 114E066, Sht. 3 of 4	M	Note 1	Pressure = BAT Elev. Head + BATP TDH
FT-113	BA flow to Reactor Makeup Control System (RMCS)	1" flanges	W Dwg. 114E066, Sht. 3 of 4	M	Note 1	Pressure = BAT Elev. Head + BATP TDH
FT-114	RMCS blended flow	2" flanges	W Dwg. 114E066, Sht. 3 of 4	M	Note 1	Pressure = BAT Elev. Head + BATP TDH
FE-314	Recycle Evap. Pkg. Concentrates flow instrument	1-1/2" flanges	W Dwgs. 4558D95, 731J486	M	Note 1	Pressure = BAT Elev. Head + RECP TDH

NOTES: 1. CS exposure is undetermined; however, it is assumed that "best practices" dictate the use of SS bolts with SS flanges in SS piping systems.

ATTACHMENT 5.6

TYPICAL EXAMPLES OF POTENTIAL LEAK LOCATIONS IN THE AUXILIARY SYSTEMS OF COMBUSTION ENGINEERING UNITS

Table 5.6-1
Potential Leak Locations with BAC Wastage Significance in the Auxiliary System Components of Combustion Engineering Units – Chemical and Volume Control Systems

CVCS Boric Acid Addition System						
Tag or SPIN No.	**Location Description**	**Potential Leaking Component**	**Drawing Reference**	**Leak Potential (H/M/L)**	**Potential Carbon Steel or Inconel Exposure (Y/N)**	**Comments**
CH-12	Boric Acid Batching Tank	Heater flanges	Ft. Calhoun dwg. E-23866-210-121 SH. 1	M	Yes SA-193 GR-B7 bolt material	Flanged chromalox heaters, SL2 dwg. 57724
	Downstream Boric Acid Batching Tank	1" union, relief side of relief valve CH-337	Ft. Calhoun dwg. E-23866-210-121 SH. 1	L	Unknown	Exposure potential to unknown adjacent components
CH-337	Downstream Boric Acid Batching Tank	3/4" angle relief stem packing	Ft. Calhoun dwg. E-23866-210-121 SH. 1	L	Unknown	Exposure potential to unknown adjacent components
	Downstream Boric Acid Batching Tank	3/4" union, pressure side of relief valve CH-337	Ft. Calhoun dwg. E-23866-210-121 SH. 1	L	Unknown	Exposure potential to unknown adjacent components
CH-102	Downstream Boric Acid Batching Tank	1/2" normally closed globe valve packing (local sample)	Ft. Calhoun dwg. E-23866-210-121 SH. 1	L	Unknown	Exposure potential to unknown adjacent components
CH-21	Downstream Boric Acid Batching Tank	2" strainer drain cap	Ft. Calhoun dwg. E-23866-210-121 SH. 1	L	Unknown	Exposure potential to unknown adjacent components
CH-11A	Concentrated Boric Acid Storage Tank A	16" manway flange	Ft. Calhoun dwg. E-23866-210-121 SH. 1	M	Yes SA-193 GR-B7 bolt material	SL2 Drawing 576716
CH-11A	Concentrated Boric Acid Storage Tank A	2" level indicator flange	Ft. Calhoun dwg. E-23866-210-121 SH. 1	L	No SA-193 GR-B6 bolt material	SL2 Drawing 576716

Table 5.6-1 (cont.)
Potential Leak Locations with BAC Wastage Significance in the Auxiliary System Components of Combustion Engineering Units – Chemical and Volume Control Systems

CVCS Boric Acid Addition System						
Tag or SPIN No.	Location Description	Potential Leaking Component	Drawing Reference	Leak Potential (H/M/L)	Potential Carbon Steel or Inconel Exposure (Y/N)	Comments
HCV-265	Downstream Concentrated Boric Acid Storage Tank A	3" normally closed gate (motor op.) valve packing (controlled leakoff)	Ft. Calhoun dwg. E-23866-210-121 SH. 1	L	Unknown	Exposure potential to unknown adjacent components
CH-140	Downstream Concentrated Boric Acid Storage Tanks A & B	2" normally closed globe valve packing (controlled leakoff)	Ft. Calhoun dwg. E-23866-210-121 SH. 2	L	Unknown	Exposure potential to unknown adjacent components
CH-155	Downstream Concentrated Boric Acid Storage Tanks A & B	3" swing check valve access cap	Ft. Calhoun dwg. E-23866-210-121 SH. 2	L	Unknown	Exposure potential to unknown adjacent components
CH-115	Downstream Concentrated Boric Acid Storage Tank A	4" locked-open gate valve packing	Ft. Calhoun dwg. E-23866-210-121 SH. 1	L	Unknown	Exposure potential to unknown adjacent components
CH-4A	Boric Acid Pump A Inlet	3" flange	Ft. Calhoun dwg. E-23866-210-121 SH. 1	M	Unknown, possible B7 bolt material	Exposure potential to unknown adjacent components
CH-4A	Boric Acid Pump A	Gland seal	Ft. Calhoun dwg. E-23866-210-121 SH. 1	M	Unknown	Exposure potential to unknown adjacent components, SL2 Drawing N750257
CH-4A	Boric Acid Pump A	Casing gasket	Ft. Calhoun dwg. E-23866-210-121 SH. 1	M	Unknown, possible B7 bolt material	Exposure potential to unknown adjacent components, SL2 Drawing N750257

Table 5.6-1 (cont.)
Potential Leak Locations with BAC Wastage Significance in the Auxiliary System Components of Combustion Engineering Units – Chemical and Volume Control Systems

CVCS Boric Acid Addition System						
Tag or SPIN No.	**Location Description**	**Potential Leaking Component**	**Drawing Reference**	**Leak Potential (H/M/L)**	**Potential Carbon Steel or Inconel Exposure (Y/N)**	**Comments**
CH-4A	Boric Acid Pump A Discharge	1 1/2" flange	Ft. Calhoun dwg. E-23866-210-121 SH. 1	M	Unknown, possible B7 bolt material	Exposure potential to unknown adjacent components

ATTACHMENT 5.7

LISTING OF SYSTEMS CONTAINING BORIC ACID

Attachment 5.7
Listing of Systems Containing Boric Acid

1. Reactor Coolant System
2. Chemical and Volume Control System
3. Safety Injection System
4. Residual Heat Removal/Shutdown Cooling System
5. Reactor Plant Sampling System
6. Spent Fuel Pool Cooling and Purification System
7. Containment Depressurization System
8. Containment Spray System
9. Reactor Plant Vent and Drain System
10. Liquid Waste Disposal System
11. Gaseous Waste Disposal System

ATTACHMENT 5.8

TYPICAL BACC ISSUE DOCUMENTATION FORM

(Reprinted with permission from TVA)

Attachment 5.8

Corrosion Control Program – Typical Boric Acid Corrosion Documentation Form

INITIAL EVALUATION OF BORATED WATER LEAKS

Part 1: Component Identification WID Number: ____________

Affected Component (UNID): ______________________ Date: ______

Component Description: ______________________

Component Location: ______________________

Building Row/Column Radius/Azimuth Elevation

Part 2: Initial Leak Evaluation:

Visual examination of leak performed by: ____ Craft Forman ____ Planner ____ Maint ____ Eng ____ Other

1. Does boric acid contact carbon or low-alloy steel? ____ Yes ____ No ____ Uncertain

2. Does the leak path come in contact with the reactor coolant pressure boundary? ____ Yes ____ No ____ Uncertain

3. What is the condition of the boric acid residue? ____ Wet ____ Dry ____ Uncertain

4. What is the color of the boric acid residue? ____ White ____ Rust ____ Uncertain

5. Is the leaking component covered by insulation? ____ Yes ____ No ____ Uncertain
   If yes, remove insulation for evaluation

6. Describe the flow path of the leak from the initiation point to where the residue ends. ______________________

____

7. If possible, take a picture of the leakage site and leakage path to document the as found condition.

8. Remove boric acid residue and/or crystals to allow for determination of corrosion damage.

9. Is there evidence of corrosion damage to the component?

Yes	No	Uncertain
____ If yes, forward original SPP-9.7-3 form to Engineering for evaluation	____ If no, proceed to Part 3	____ If uncertain, forward the original SPP-9.7-3 form to Engineering for evaluation

Part 3: Physical Actions Taken to Address Leakage:

1. Action Taken: ______________________

____

2. Forward a copy of the original SPP-9.7-3 form to Engineering (WID may be closed when the action from Part 3, step 1 is complete and leakage has been stopped.)

Print Name	Signature	Organization	Telephone	Date

Supervisor (Print Name and Signature)	Date:

Attachment 5.8 (cont.)

ENGINEERING EVALUATION OF BORATED WATER LEAKS

Initiating Document (WO, etc): ______________________________

Component: ______________________________
UNID Description

1.0 Description of Source and Leak Path:

Is description of source and leak path as described on SPP-9.7-3 correct? ___ Yes ___ No

If "NO" provide corrected information below (Determine entire leak path. Note any contact with carbon or low-alloy steel components or reactor coolant pressure boundary):

____

____

____

2.0 Damage Assessment:

Description of surface damage: ______________________________

____

Applicable acceptance criteria: ______________________________

Does surface damage exceed acceptance criteria? ___ Yes ___ No

PER No. ____________ ___ If Yes Above

3.0 Recommended Additional Corrective Actions:

1. If damage is within acceptance limits, recommend additional corrective actions if required.
2. If damage exceeds acceptance limits, recommend either repair or replacement.
3. For ASME Section XI components, perform a suitability evaluation for repair/replacement in accordance with SPP-9.1.

Component is ASME Section XI? ___ Yes ___ No

Comments: ______________________________

____

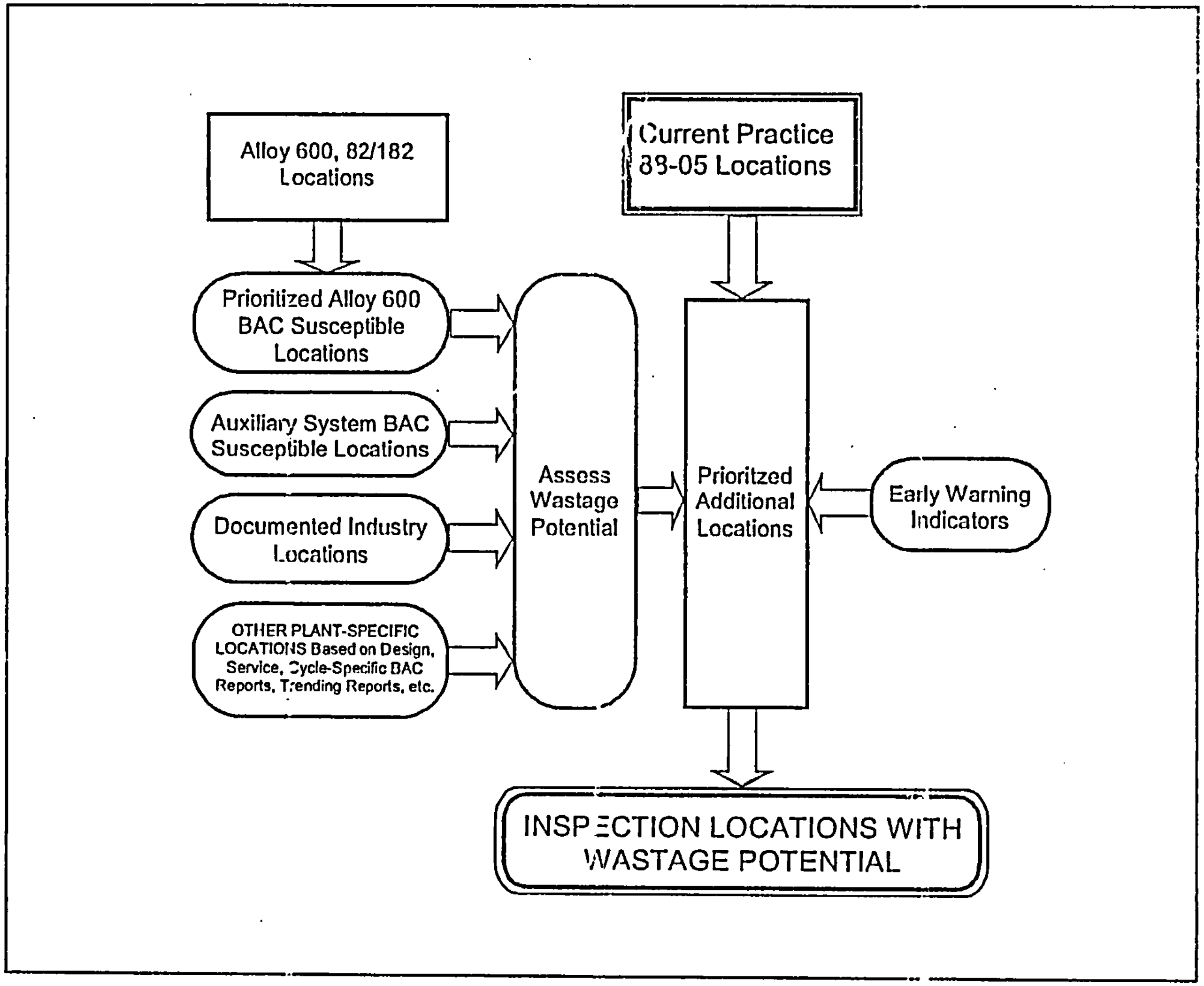
	/	/	/	/
Site Eng:	Signature	Org	Telephone	Date
	/	/	/	/
Supervisor	Signature	Org	Telephone	Date

6 FLOW CHARTS

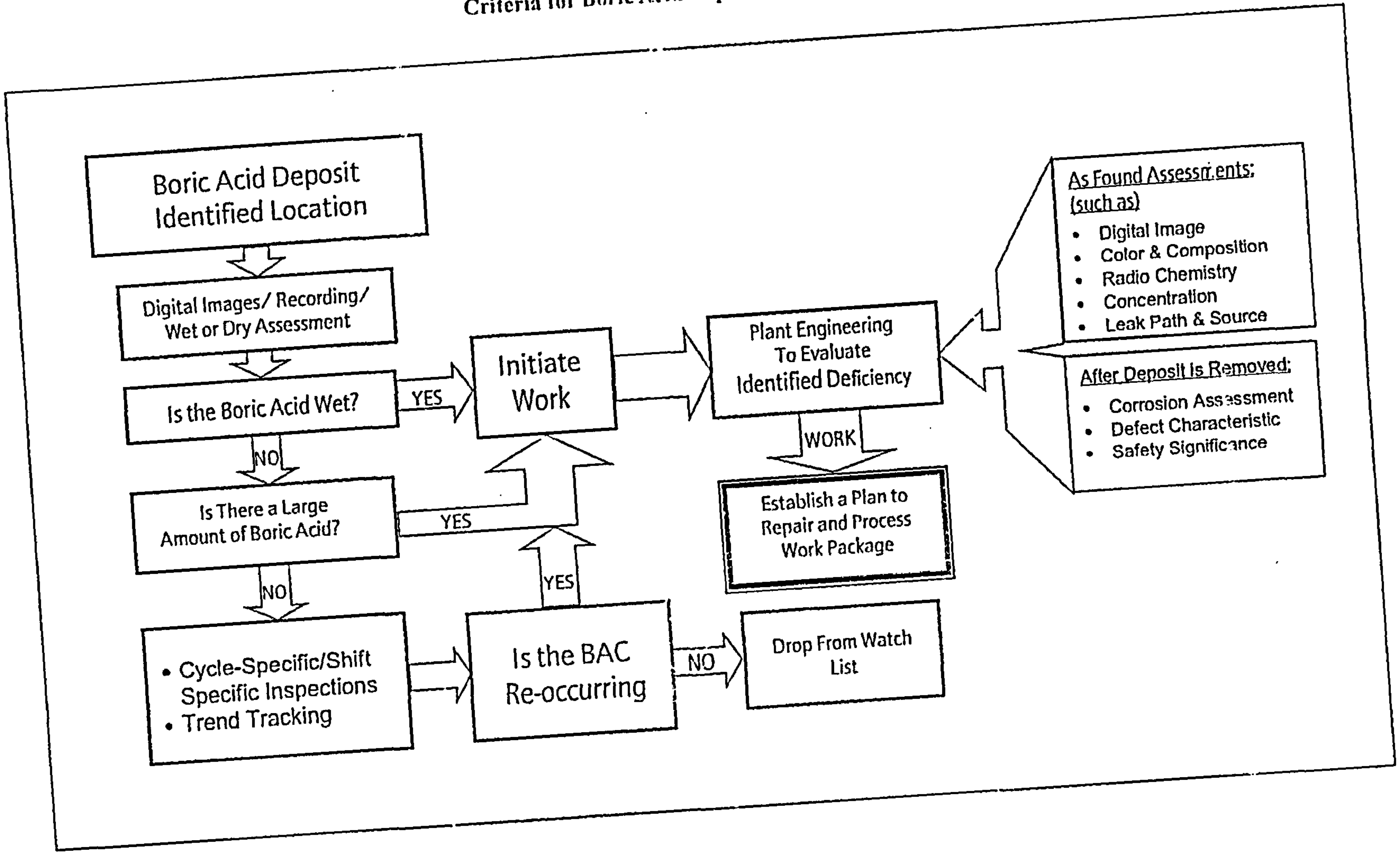
6-1 Identification of Inspection Locations with Wastage Significance

6-2 Criteria for Boric Acid Deposit Assessment

Flow Chart 6-1
Identification of Inspection Locations with Wastage Significance



Flow Chart 6-2
Criteria for Boric Acid Deposit Assessment



7 REFERENCES

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2. INPO SER 4-01, "Recent Events Involving Reactor Coolant System Leakage at Pressurized Water Reactors," July 26, 2001.

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5. NRC Bulletin 2002-02: "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," August 9, 2002.

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9. "Boric Acid Corrosion Guidebook, Revision 1: Managing Boric Acid Corrosion Issues at PWR Power Stations," TR-1000975, EPRI, Palo Alto, CA, 2001.