APS

10 CFR 50.54(f)

Palo Verde Nuclear
Generating Station

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References: 1. APS Letter No. 102-04702- GRO/SAB/RJR, "APS' Response to the Information Requested by NRC Bulletin 2002-01, Item 3.A." dated May 17, 2002, from Gregg R. Overbeck, APS to USNRC.

2. NRC letter to APS dated November 21, 2002, "Bulletin 2002-01, 'Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,' 15-Day and 60-Day Responses for Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Request for Additional Information."

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
APS' Response to NRC Request for Additional Information dated November 21, 2002 (TAC Nos. MB4563, MB4564, and MB4565).

By letter dated March 18, 2002, the U.S. Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2002-01 "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." Arizona Public Service Company (APS) provided the requested 60-day response in Reference 1. The staff evaluated the 60-day response and in Reference 2 requested additional information to support the evaluation. APS' response to this request for additional information is provided in the Enclosure to this letter.

No new commitments are being made to the NRC by this letter.

Should you have any questions, please contact Thomas N. Weber at (623) 393-5764.

Sincerely,

Gregg R. Overbeck

A095

GRO/SAB/RJR/kg

Enclosure APS' Response to Request for Additional Information Regarding NRC Bulletin 2002-01

cc:

E. W. Merschoff	(NRC Region IV)
J. N. Donohew	(NRR Project Manager)
N. L. Salgado	(NRC Resident Inspector)

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, Gregg R. Overbeck, represent that I am Senior Vice President – Nuclear, that the foregoing document has been signed by me on behalf of Arizona Public Service Company with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

Gregg R. Overbeck
Gregg R. Overbeck

Sworn To Before Me This 31st Day Of January, 2003.

Cassandre Justiss
Notary Public

OFFICAL SEAL
Cassandre Justiss
NOTARY PUBLIC - STATE of ARIZONA
MARICOPA COUNTY
MY COMM. EXPIRES October 30, 2006

Notary Commission Stamp

ENCLOSURE

APS' Response to Request for Additional Information Regarding NRC Bulletin 2002-01

APS' Response to Request for Additional Information to NRC Bulletin 2002-01

This is the Arizona Public Service Company (APS) response to Nuclear Regulatory Commission (NRC) letter dated November 21, 2002 requesting additional information related to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002.

The responses in this enclosure provide information about the Palo Verde Nuclear Generating Station (PVNGS) Boric Acid Corrosion Control Programs as it pertains to the Reactor Coolant Pressure Boundary (RCPB). The inspections/examinations of the upper Reactor Pressure Vessel Head (RPVH) and the Steam Generator Tube/Tube Sheet Inspection Program are not included in these responses. The following information demonstrates compliance with Generic Letter 88-05, as well as the current requirements of 10 CFR 50.55(a), Section XI of the American Society of Mechanical Engineers (ASME) Code, and the PVNGS Technical Specifications.

PVNGS has two programs that address boric acid corrosion detection and control including Alloy 600 pressure boundary materials and dissimilar metal Alloy 82/182 welds and connections. These programs are the Inservice Inspection (ISI) Program and the Boric Acid Corrosion Prevention Program.

NRC Required Information

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

APS Response

ISI Program

The ISI program requires all ASME Class 1 components to receive a VT-2 visual examination once every refueling outage. The Class 1 pressure boundary piping is visually examined at the end of the refueling outage at normal operating pressure. The scope of the examination includes all Alloy 600 nozzles, all Alloy 82/182 dissimilar metal welds, and all Class 1 bolted connections within the RCPB. Additionally, the Pressurizer Heater and Hot leg Nozzles are also visually examined for leakage during cold shutdown. This additional examination is due to plant experience with this particular nozzle arrangement.

The ASME Section XI Code allows visual examinations to be performed with the insulation installed. ISI required visual examinations are performed by certified VT-2 examiners. However, when there are specific areas of interest due to operating plant experience, (e.g., pressurizer heater sleeves, alloy 600 nozzles in the hot leg piping) insulation modifications have been made to allow for bare-metal examinations. If boric acid residue is found during an examination, the ISI Program requires the source of the leakage to be determined and an engineering evaluation to be performed of the impact of the leakage.

APS procedure 73TI-9ZZ78, Visual Examination for Leakage, states that "for rejected pressure tests a work order and/or CRDR shall be generated." The work order (WO) or condition request/disposition request (CRDR) would identify the condition and document the engineering evaluation performed (per the station corrective action program) to meet the ISI Program. Procedure 73TI-9ZZ78 contains the following examples of rejectable conditions:

- Pressure boundary leakage.
- Evidence of leakage at pressure retaining bolted connections.
- Components with general corrosion (from boric acid residue) that reduces the wall thickness by more than 10%.

All piping welds that have Alloy 82/182 welds are Surface and Volumetric examined at different periods during the 10-year ISI program Interval. At PVNGS, the following welds are Alloy 82/182.

Component	Total Nozzles	Nozzle Size	Weld ID
PRESSURIZER Unit 1			
Pressurizer Spray	1	4"	005-33
Pressurizer Safeties	4	6"	005-29, 005-30, 005-31, 005-32
Pressurizer Surge	1	12"	005-34
PRESSURIZER Unit 2			
Pressurizer Spray	1	4"	005-33
Pressurizer Safeties	4	6"	005-29, 005-30, 005-31, 005-32
Pressurizer Surge	1	12"	005-34
PRESSURIZER Unit 3			
Pressurizer Spray	1	4"	005-33 (IEIN82-09)
Pressurizer Safeties	4	6"	005-29, 005-30, 005-31, 005-32
Pressurizer Surge	1	12"	005-34

When the Alloy 82/182 welds are Surface or Volumetric examined, the insulation is removed to allow access to the surface of the weld. As previously discussed, any rejectable condition of these welds would require an engineering evaluation to be performed. All ISI required Surface or Volumetric examinations are performed by certified examiners in the respective process.

Boric Acid Corrosion Prevention Program

Procedure 70TI-9ZC01, Boric Acid Corrosion Prevention Program, requires a complete Boric Acid Walkdown (BAW) inspection to be performed at each refueling outage. Additional BAW inspections are performed during other plant shutdowns. A complete BAW inspection is required to be performed any time the unit at the time of shutdown has been in power operation for more than three months or 90 Effective Full Power Days (EFPD) since the last inspection. Additionally, a determination to perform a limited BAW inspection is required after each reactor trip or controlled shutdown if less than three months or 90 EFPD have elapsed since the last inspection. However, if the unit will be in Mode 3 or below for more than approximately 14 days, a complete BAW is required. The technical basis for the BAW inspection frequency is PVNGS and industry experience. A complete BAW inspection every refueling outage is the most frequent interval that the inspection can be performed without a mid-cycle shutdown. Industry experience has demonstrated that a complete and effective inspection every refueling outage, combined with an effective corrective action program, is sufficient to prevent significant degradation of the RCPB due to boric acid leakage and corrosion. The three-month/90 day inspection requirement for other plant shutdowns is based on PVNGS experience as a reasonable, conservative compromise between the risk and consequences of boric acid leaks and ALARA considerations.

The BAW inspection is usually conducted by zones. Typically zones outside the bio-shield are inspected in Mode 1 just prior to the start of the refueling outage and zones inside the bio-shield are inspected in Mode 3 at the beginning of the refueling outage at normal operating pressure and temperature. The inspection includes observation of the components identified in procedure 70TI-9ZC01 as principle potential leak locations. The listing of potential leak locations include all Alloy 600 penetrations of the RCPB, with the exception of the reactor vessel bottom head in-core instrumentation penetrations. The technical basis for inclusion of the Alloy 600 penetrations in the procedure listing as potential boric acid leak sites is that these penetrations are susceptible to primary water stress corrosion cracking (PWSCC). The specific inspection technique and coverage for each Alloy 600 penetration is discussed below:

RCS Hot Leg Instrument and Sampling Nozzles

The design of each PVNGS Unit originally consisted of 27 Alloy 600 penetrations of the hot legs. The penetrations include pressure taps, sampling line nozzles and spare and in-service RTD thermowell nozzles. Procedure 70TI-9ZC01 requires a bare-metal inspection (i.e., insulation removed) of these penetrations. Coverage is considered to be 100%. The technical basis for the bare-metal inspection requirement is the high susceptibility to PWSCC of Alloy 600 nozzles exposed to hot leg temperatures and the resultant potential exposure of the hot leg carbon steel base material to boric acid. PVNGS has been systematically replacing the Alloy 600 hot leg penetration nozzles with nozzles fabricated from corrosion-resistant Alloy 690 material. Spare RTD nozzles have been plugged with Alloy 690 material. During the most recent U1R10 refueling outage, PVNGS completed the replacement of all hot leg Alloy 600 instrument nozzles with Alloy 690, leaving no PWSCC-susceptible Alloy 600 material on the Unit 1 hot legs. The Alloy 600 hot leg penetration nozzles in Unit 3 are scheduled to be replaced during that Unit's next refueling outage (U3R10-Spring 2003) and in Unit 2 during the 2R12 outage (Spring 2005).

Pressurizer Instrument Nozzles

The design of each PVNGS pressurizer originally consisted of seven Alloy 600 instrument penetrations. The penetrations include pressure/level taps and a temperature element. All of the instrument penetrations originally fabricated of Alloy 600 material have been replaced with Alloy 690 in all three Units. Accordingly, procedure 70TI-9ZC01 requires only an inspection of each nozzle location (i.e., a bare-metal inspection is not required). Such an inspection consists of an observation of the component location, looking for abnormal conditions that might indicate a boric acid leak. Such conditions would include boric acid residue on or at the base of the nozzle where it protrudes from the base metal, or residue at seams in the insulation. The technical basis for not requiring a bare-metal inspection of these nozzles is that Alloy 690 material is resistant to PWSCC. Weld material used in the replacements was E82 for Unit 1, and E52 for Units 2 and 3.

Pressurizer Heater Sleeves

The bottom head of each PVNGS pressurizer includes 36 Alloy 600 heater sleeve penetrations. Due to heater sleeve susceptibility to PWSCC (exposed to pressurizer operating temperature), the insulation has been modified to allow for a complete visual inspection of each heater sleeve.

Reactor Head Nozzles

Each reactor head includes 97 Alloy 600 penetrations for Control Element Drive Mechanisms (CEDM) nozzles and one head vent nozzle. During the performance of the BAW inspections the visible surfaces of the CEDM and head vent nozzles, and other vessel head appurtenances, are examined for visual evidences of boric acid leakage. The procedure does not require a bare-metal inspection because the vessel head insulation is a close-fitting type not designed to be routinely removed. A specific inspection program for reactor head penetration nozzles is being employed per the requirements of NRC Bulletin 2001-01. Palo Verde Units 1 and 2 have undergone an extensive inspection program of the CEDM nozzles as discussed in Reference 2. The examinations yielded no detectable defects, no visual indications of leakage on 24 peripheral nozzles examined, no visual indications of leakage on the reactor vessel head (RVH) vent line, and no detection of any leak path indication between the nozzle outside diameter and shell bore interface. Unit 3 inspections are scheduled for the spring of 2003.

Cold Leg and Steam Generator Primary Instrument Nozzles

The design of each PVNGS Unit includes 28 Alloy 600 penetrations of the cold legs and steam generator cold leg primary shell. The penetrations include pressure taps and in-service RTD thermowell nozzles. The procedure requires an inspection of these locations for abnormal leakage. However, a bare-metal inspection is not required. The technical basis for not requiring a bare-metal inspection of these nozzles is the lower susceptibility to PWSCC of Alloy 600 materials when exposed to cold leg operating temperatures (typically 555° F). A time-at-temperature model developed by the Materials Reliability Project (MRP) for the RPV head penetrations shows that PWSCC of Alloy 600 is highly sensitive to temperature. The model ranks plants based on the number of effective full power years (EFPY) of operation required for that plant to reach the same EFPYs as Oconee 3, normalized for any differences in head temperature. Using this approach for comparison purposes, the cold leg and steam generator primary instrument nozzles with a T_{cold} of 565 °F, would reach susceptibility to PWSCC in an estimated 66.7 EFPY. As operating hours accumulate, or as deemed prudent based on industry experience, these low susceptibility locations may become more susceptible to PWSCC and in the future require more conservative examination methods such as a bare-metal inspection. However, the current inspection technique has been determined to be the appropriate balance between the relatively low susceptibility to PWSCC of these nozzles and ALARA principles in light of current circumstances.

Reactor Bottom Head Nozzles

The design of each PVNGS Unit includes 61 Alloy 600 penetrations of the reactor bottom head for In-core instrumentation (ICI). During the first 10-year ISI program interval, a relief request was granted to monitor sump levels to determine if there was RCS leakage under the reactor vessel in lieu of visual examinations.

The ISI second interval program requires visual examination of the bottom of the reactor head, ICI penetrations, and ICI tubing following each refueling outage during the Mode 3 RCS pressure test and visual inspection for leakage per procedure 73TI-9ZZ78. Due to the high dose rates normally found under the reactor bottom head (very high radiation area (VHRA)), the inspection is performed using a remote, robotic video camera. Removal of insulation is not required per this procedure.

PVNGS has documented in our corrective action program refueling water leaking into the ICI chase from the original refueling pool seal and the ICI seal table that is flooded during refueling operations. There is no leakage onto the bottom reactor vessel head. The ICI tubing and the support spherical bearings are stainless steel and are not affected by boric acid deposits. The ICI penetrations are the low point of the reactor vessel and, therefore, leakage is expected to be visually detectable at this low point.

Due to this history of known leakage, ISI performs a visual examination of the ICI tubing and penetrations while the refueling pool is flooded during the refueling outage. This visual examination documents the ICI seal table leakage and gives a reference for the required examination. During Mode 3 at normal operating pressure (NOP) the visual examination is performed to meet the ASME Section XI requirements. Both of these examinations use certified VT-2 examiners to perform the inspection. The effects from the seal table leakage on the carbon steel support structures is being evaluated using our CRDR process.

The Boric Acid Corrosion Prevention program does not currently specify the Alloy 600 ICI penetrations on the bottom of the reactor vessel as principle potential leak sites. The technical basis for not including ICI penetrations in the inspection program is the lack of industry or PVNGS experience of leaks at these locations. Over the past 10 years, approximately 20 units have had volumetric exams performed on the base metal (no attachment j-welds had been inspected). The units inspected include 17 EdF units, 2 Electrabel units and 1 Ringhals unit. These inspections have resulted in no cracking in the 20 units, with over 1,000 tubes inspected. In the US, visual inspections at Davis Besse show boron, which appears to be coming from the top head, however, the investigation is on going.

Alloy 600 nozzles exposed to at or near cold leg temperatures (including reactor bottom head penetrations) are considered to be of low susceptibility to PWSCC and therefore do not currently require bare-metal inspection at PVNGS. The technical basis is time-at-temperature models developed within the industry for Alloy 600 materials in Pressurized Water Reactor (PWR) environments, as well as the lack of industry or PVNGS experience of leaks at these locations. As operating hours accumulate, these low susceptibility locations may become more susceptible to PWSCC and require more conservative examination methods.

To be consistent with the inspection of other leak sites subject to cold leg (or near cold leg) temperatures, the BAW procedure is being revised to include the bottom of the reactor vessel as a potential leak site.

Bolted Connections

All bolted connections within the RCPB are listed as principle leak locations in Procedure 70TI-9ZC01. Insulation is not required to be removed. The visual inspection consists of an observation of the component location looking for abnormal conditions that might indicate a boric acid leak. Such conditions would include boric acid residue at uncovered locations or at seams in the insulation. The technical basis for the performance of a visual inspection of all bolted connections within the RCPB is that PVNGS and industry experience indicates bolted connections are the most likely source of boric acid leaks. Insulation removal is not required based on industry experience. The nature of boric acid leaks is such that leaks are readily detected by visual inspection, even with insulation installed. Boric acid residue and other evidence of leakage, such as wetted insulation, is detectable by an effective visual inspection prior to the onset of significant boric acid corrosion.

Personnel Qualifications

BAW inspections are performed by personnel trained and qualified to the requirements of procedure 70TI-9ZC01. These requirements do not include a specific VT-2 qualification. PVNGS experience has demonstrated that the ability to recognize very small volumes of boric acid leakage is not dependent on VT-2 qualification. Rather the BAW qualifications concentrate on the performance of a thorough inspection of the principle potential leak locations and evaluation of identified leakage for potential degradation of components susceptible to boric acid corrosion.

All ISI required visual examinations are performed by certified VT-2 examiners. APS personnel are certified in accordance with 73DP-9EE16, Qualification and Certification of NDE Personnel, which meets ASME and ASNT CP-189.

NRC Required Information

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

APS Response

The requirements for insulation removal during refueling outages, as specified in the Boric Acid Corrosion Prevention Program procedure, are based on the relative susceptibility of the component to PWSCC. Nozzles and penetrations that have been replaced with Alloy 690 material are not considered to be susceptible to PWSCC and, therefore, do not require insulation removal. Alloy 600 nozzles exposed to hot leg or higher temperatures are considered to be highly susceptible to PWSCC and, therefore, a bare-metal inspection is performed. During non-refueling outages, the insulation is not typically removed when performing a BAW unless required to facilitate a thorough inspection of a suspected leak location.

Alloy 600 nozzles exposed to at or near cold leg temperatures (including reactor bottom head penetrations) are considered to be of low susceptibility to PWSCC and, therefore, do not currently require bare-metal inspection. The technical basis is time-at-temperature models developed within the industry for Alloy 600 materials in PWR environments. As operating hours accumulate, these low susceptibility locations may become more susceptible to PWSCC and require more conservative examination methods.

Procedure 70TI-9ZC01 provides specific guidance for a thorough inspection of identified leak sites. It includes the removal of insulation and deposited boric acid crystals as necessary to determine the leak source, quantify the extent of the leakage, the run-off path, and exposure of adjacent components to boric acid leaks, drips or spray.

The ISI program, as well as the ASME Section XI Code, allows visual examinations to be performed with the insulation installed. If boric acid residue is found during a visual examination, the source of leakage must be determined and an engineering evaluation performed on the impact of the leakage. The source of leakage determination would include insulation removal, as appropriate. Procedure 73TI-9ZZ78 is used for the visual examination for leakage, which states "for rejected pressure tests a work order and/or CRDR shall be generated." The work order or CRDR would document the engineering evaluation that is required to meet the ISI Program and ASME Section XI Code required evaluation. Procedure 73TI-9ZZ78 has examples of rejectable conditions:

- Pressure boundary leakage.
- Evidence of leakage at pressure retaining bolted connections.
- Components with general corrosion (from boric acid residue) that reduces the wall thickness by more than 10%.

The insulation installed on the reactor coolant system (RCS) is the reflective insulation that is routinely removed to meet the inspection requirements of the ISI program welds.

NRC Required Information

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

APS Response

There are no areas of the RCPB considered to be inaccessible. There are two areas that have limited access that impacts the routine performance of the BAW and ISI inspections. These two areas include the reactor vessel bottom head, due to radiation exposure circumstances, and the reactor vessel head due to the close fitting insulation design. The details regarding the inspection of the reactor vessel bottom head are discussed in the response to Questions 1 and 5. References 1, 2, 3, and 4 discuss the upper head inspections as a result of NRC Bulletins 2001-01 and 2002-01.

NRC Required Information

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

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

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

APS Response

As discussed previously, the BAW inspection is usually conducted by zones per Procedure 70TI-0CZ03. Typically zones outside the bio-shield are inspected in Mode 1 and zones inside the bio-shield are inspected in Mode 3 at normal operating pressure and temperature. Leakage sites, including those from mechanical joints, are identified on a detailed checklist. A thorough inspection of all identified leak sites is conducted. Removal of insulation and deposited boric acid crystals may be performed to allow for the determination of leak source, run-off path, and potentially impacted carbon steel components. Each leak is characterized as to whether the leakage is active or inactive, whether carbon steel components are or potentially affected, and the extent of the leakage quantified as slight, moderate, or heavy.

Any leakage site that is identified as having a potential impact on carbon steel components is classified as a material deficiency. A deficiency (DF) work order is generated to correct the leakage and perform an evaluation of the impacted carbon steel. The BAW procedure requires a careful cleaning of the affected component and an evaluation of the extent to which the component has been affected. The procedure identifies that wall thickness, diameter measurements, localized corrosion depths etc. of the affected area may be required to determine the extent of the degradation and its potential impact on component operability and structural integrity. Typically the work order to correct the leakage will involve re-packing a valve, replacing a bonnet seal, or otherwise correcting the source of the leak.

Evaluation of impacted carbon steel would involve denoting the degree of degradation and replacement of the component (such as a packing gland eyebolt). If it is determined that potentially impacted carbon steel components cannot be replaced, an evaluation is performed to demonstrate the acceptability of continued operation. The evaluation would determine the current degree of degradation and anticipated future degradation due to continued exposure to the boric acid leak. A corrosion rate assessment may be performed as necessary based on industry corrosion rate data at the operating temperature of the affected components, utilizing the EPRI Boric Acid Corrosion

Guidebook, or other similar industry or operating experience, as necessary. A structural evaluation would be performed to demonstrate that structural integrity would be maintained and compliance with ASME Code requirements.

Mitigating measures, such as coating of the carbon steel components with an approved grease proven to protect carbon steel parts subjected to wet boric acid, or installation of a drip catch to re-direct the leak path away from the carbon steel component, may be taken as necessary. Monitoring actions may also be implemented as necessary. Monitoring actions have included periodic at-power containment entries for visual inspection or installation of a video camera. This information, as well as information discussed above, are typical examples of what would be specified by and documented in the DF work order.

In rare cases in which boric acid leakage has resulted in significant degradation of carbon steel RCPB components, procedure 70TI-9ZC01 requires a CRDR be initiated to document and trend the condition. The CRDR process ensures that an Equipment Root Cause of Failure Analysis (ERCFA) will be performed, as appropriate. As part of the ERCFA program, corrective actions are identified to prevent recurrence. A CRDR would also be initiated if more serious leakage conditions are identified but degradation has only just started.

For leakage sites that have been identified as not affecting carbon steel components, corrective maintenance (CM) work orders are initiated to correct the leakage. Typically these actions would involve tightening of packing, replacing gaskets, or re-packing valves. Since carbon steel components have not been affected, evaluations of the affect of the leakage are not performed. As an added inherent benefit of the documentation process of the program, however, these leakage sites can be tracked and trended, such that proactive preventative measures may be implemented if deemed prudent.

Those leakage sites that are characterized as slight, inactive, and not affecting carbon steel components would not require corrective action. Such sites may be annotated as "clean only" in the BAW work order. The technical basis for continued operation is operating experience at PVNGS and previous evaluations of the BAW program. PVNGS experience has shown that the majority of leaks are identified as packing leaks on small (~ 1 inch and smaller) manual vent, drain, and instrument root valves. Packing leakage exhibited by these valves has been almost exclusively slight, inactive, and had no carbon steel impact.

ISI Procedure 73TI-9ZZ78 states that "if evidence of leakage is detected at pressure retaining bolted connections the pressure test shall be rejected". The next procedure step states "bolting at leaking connections shall be either removed and VT examined per IWA-5250, or evaluated in accordance with Appendix B of this procedure." Appendix B is the ISI NRC approved Relief Request Number 12 and it defines the steps necessary to perform an evaluation of bolted connections.

NRC Required Information

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

APS Response

The ISI second interval program requires visual examination of the bottom of the reactor head, ICI penetrations, and ICI tubing following a refueling outage during the Mode 3 RCS pressure test and visual inspection for leakage per procedure 73TI-9ZZ78. Due to the high dose rates normally found under the reactor bottom head (VHRA) the inspection is performed using a remote, robotic video camera. Removal of insulation is not required per this procedure.

It is expected that even low levels of leakage will result in detectable evidence of boric acid deposits. For example, the EPRI Boric Acid Corrosion Guidebook, Figure 6-3, shows that 50 days of 0.01 gallons per minute (gpm) leak-rate for 1500 ppm borated water will result in 720 gallons of leakage volume and 45 pounds of boric acid deposits. The area of deposits was calculated to be 870 square inches. APS believes our current inspection method will detect low levels of leakage such as this. Previous studies related to top of head CEDM nozzle leakage have been submitted to the NRC by Owners Groups demonstrating that low levels of borated water leakage will not result in an immediate safety concern due to carbon steel wastage.

If boric acid residue is found during a bottom head ICI visual examination, the source of leakage must be determined and an engineering evaluation performed on the impact of the leakage. The source of leakage determination would include insulation removal if necessary. Procedure 73TI-9ZZ78 is used for the visual examination for leakage, which states in corrective actions "for rejected pressure tests a work order and/or CRDR shall be generated." The work order or CRDR would document the engineering evaluation that is required to meet the ISI Program and ASME Section XI Code required evaluation. The leakage path would also be evaluated for impacted components, which in this case would include the effects of the leak on the carbon steel supports.

As previously discussed, Alloy 600 nozzles exposed to at or near cold leg temperatures (including reactor bottom head penetrations) are considered to be of low susceptibility to PWSCC and, therefore, do not currently require bare-metal inspection at PVNGS. The technical basis is time-at-temperature models developed within the industry for Alloy 600 materials in PWR environments, as well as the lack of industry or PVNGS experience of leaks at these locations. Over the past 10 years, approximately 20 units have had volumetric exams performed on the base metal (no attachment j-welds had been inspected). As operating hours accumulate, these low susceptibility locations may become more susceptible to PWSCC and require more conservative examination methods.

Therefore, APS believes the current examination method is sufficient to detect leakage and provides assurance that pressure boundary leaks and boric acid residue will not have an adverse impact on the structural integrity of the bottom of the reactor vessel head material.

NRC Required Information

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

APS Response

The capability to detect low levels of reactor coolant pressure boundary leakage is provided for by the combined implementation of the ISI program and the Boric Acid Corrosion Prevention program. The visual examination of identified potential leakage sites, via the BAW examination, performed every refueling outage in each unit, and under certain circumstances at other plant shutdowns, and of the RCPB via the ISI program has proven to be effective in detecting leakage prior to significant degradation of the pressure boundary due to boric acid corrosion.

In addition to visual examinations during refueling outages and shutdowns, the Boric Acid Corrosion Prevention program employs the use of five sampling methods during operating cycles to aid in the identification of potential small primary leaks. These methods are:

- Containment atmosphere particulate radioactivity monitors, including evaluation of monitor filters for iron oxide and boric acid fouling
- Containment atmosphere gaseous radioactivity monitors
- Containment relative humidity readings
- Containment sumps - level rate of change and discharges
- Reactor coolant system water inventory balance measurements

An early warning of pressure boundary leakage or unidentified leakage is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level.

Industry practice has shown that water flow changes of 0.5 gpm to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump monitor consists of instrumentation used to monitor containment sump level and flow (pump run time). The containment sump used to collect unidentified leakage is instrumented to alarm if the rate of level increase corresponds to a sump inflow greater than 1 gpm for 1 hour. This sensitivity is acceptable for detecting increases in unidentified leakage.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} µCi/cc radioactivity for particulate monitoring and of 10^{-6} µCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities, because of their sensitivities and responses to RCS leakage.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature or relative humidity measurements can thus be used to monitor increasing humidity levels of the containment atmosphere as an indicator of potential RCS leakage.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem.

Air temperature and pressure monitoring methods may also be used to infer unidentified leakage to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing sizable leakage in containment.

If any one or more of the above sampling methods were to identify an abnormal condition potentially indicating a primary coolant leak, System and/or Maintenance Engineering, as well as Plant Management, would be notified of the sampling results. An initial qualitative assessment would be performed by Engineering utilizing data trends and comparisons to qualify the potential significance of the result. A CRDR would be initiated if deemed appropriate. An at-power containment walkdown may be performed if this assessment determined it was warranted. Once the leakage source is identified, an evaluation in accordance with the Boric Acid Corrosion Prevention procedure would be performed.

Any leakage site that is identified as having a potential impact on carbon steel components would be classified as a material deficiency. A DF work order would be generated to correct the leakage and an evaluation of the impacted carbon steel would be performed. Per procedural requirements, the run-off path and surrounding area would be examined. The procedure requires a careful cleaning of any affected component and an evaluation of the extent to which the component has been affected. Insulation would be removed if necessary to perform this evaluation. The procedure identifies that measurements such as wall thickness, diameter measurements, localized corrosion depths of the affected area may be required to determine the extent of the degradation and its potential impact on component operability and structural integrity.

NRC Required Information

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

APS Response

Presently, none of the aspects of the PVNGS Boric Acid Corrosion Control program make use of formal susceptibility or consequence models. Relative susceptibility to PWSCC based on exposure temperature of Alloy 600 materials is employed within the Boric Acid Corrosion Prevention program's specification of examination techniques. Those locations that are considered most susceptible due to exposure to hot leg or higher temperature receive a bare-metal inspection during refueling outages. Locations exposed to cold leg or near-cold leg temperatures do not require a bare-metal inspection due to the lower susceptibility to PWSCC. The technical basis for the use of relative susceptibility is time-at-temperature models developed within the industry for Alloy 600 materials in a PWR environment and industry experience.

NRC Required Information

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

APS Response

At the request of Combustion Engineering (CE) NSSS Licensees, Westinghouse reviewed Combustion Engineering and ABB-CE databases and communications to determine what recommendations had been made to CE plant owners on visual inspections of Alloy 600/82/182 materials in the reactor coolant pressure boundary. The results of this review were provided in Westinghouse Letter WOG-02-223/CEOG-02-259, "Vendor Recommendations for Visual Inspections of Alloy 600/82/182 Component Locations (MUHP-5035, CEOG 2046)" dated December 13, 2002. The recommendations made to CE plants are:

(1) inspect pressurizer small diameter Alloy 600 nozzles and heater sleeves during each refueling outage for signs of primary coolant leakage,

(2) inspect with the insulation in-place or removed (either approach is acceptable). The presence of boric acid deposits or corrosion products should be assumed to be an indication of leakage until proven otherwise and appropriate actions taken to stop the leakage,

(3) inspect low alloy steels exposed to boric acid and promptly repair primary coolant leaks.

The current APS Boric acid corrosion control programs are considered to be consistent with the vendor summary of recommendations.

<u>NRC Required Information</u>

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

<u>APS Response</u>

At Palo Verde, the Boric Acid Corrosion Prevention Program's boric acid walkdown and the Inservice Inspection Program's visual examination for leakage are performed as part of a refueling outage. The plants Technical Specification (TS) 3.4.14 on RCS operational leakage is applicable in Modes 1, 2, 3, and 4. As such, they are not directly used to comply with the Plant's TSs during normal plant operations. However, these programs do provide added assurance that systems and components forming the reactor coolant pressure boundary are free of pressure boundary leaks and boric acid residue.

During plant life, RCS joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational Leakage LCO (3.4.14) is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage. 10 CFR 50, Appendix A, GDC 30, requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the identified leakage from the unidentified leakage is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection. LCO 3.4.14 deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a Loss Of Coolant Accident (LOCA).

Except for primary to secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary leakage as the initial condition. Primary to secondary leakage is a factor in the dose releases outside containment resulting from a Steam Line Break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a Steam Generator Tube Rupture (SGTR).

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm primary to secondary leakage in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50 or the staff approved licensing basis (i.e., a small fraction of these limits).

RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36 (C)(2)(ii).

RCS operational leakage shall be limited to:

a. <u>Pressure Boundary Leakage</u>

   No pressure boundary leakage is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB. Leakage past seals and gaskets is not pressure boundary leakage.

b. <u>Unidentified Leakage</u>

   One gallon per minute (gpm) of unidentified leakage is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

c. Identified Leakage

Up to 10 gpm of identified leakage is considered allowable because leakage is from known sources that do not interfere with detection of unidentified leakage and is well within the capability of the RCS makeup system. Identified leakage includes leakage to the containment from specifically known and located sources, but does not include pressure boundary leakage or controlled Reactor Coolant Pump (RCP) seal leakoff (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

The PVNGS Technical Specifications states the following:

LCO 3.4.14 RCS operational leakage shall be limited to:

a. No pressure boundary leakage;

b. 1 gpm unidentified leakage;

c. 10 gpm identified leakage; and

d. 150 gallons per day primary to secondary leakage through any one SG.

Verifying RCS leakage to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance.

Procedures 40ST-9RC02, ERFDADS Calculation of RCS Water Inventory, 40ST-9RC05, Manual Calculation of RCS Water Inventory, and 40ST-9RC08, Backup Calculation of RCS Water Inventory are used by Operations to determine if leakage exists during plant operations (Modes 1 – 4).

The 1992 ASME Code, 1992 Winter Addenda of Section IWA-5250 Corrective Actions states the following:

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

APS procedure 73TI-9ZZ78, Visual Examination for Leakage, states that "for rejected pressure tests a work order and/or CRDR shall be generated." Components with general corrosion (from boric acid residue) that reduce the wall thickness by more than 10% are identified as a rejectable condition. This procedure implements the IWA-5250 requirement to evaluate corrosion that reduces wall thickness by more than 10%.

When performing a BAW, PVNGS procedure 70TI-9ZC01 contains the guidance and actions required when significant degradation of carbon steel RCPB components is identified. This would result in a CRDR that would, if necessary, be evaluated by personnel who are familiar with ASME Code requirements.

The steps from procedure 70TI-9ZC01 that initiate this review are provided below.

3.3 CONDUCTING EXAMINATIONS

3.3.2 ...For identified leaks, the runoff path and surrounding area should be searched. Thoroughly inspect nearby components for boric acid attack due to dripping or spraying. Verify that no damage has occurred to other adjacent components from dripping or spraying boric acid solutions.

3.3.3 Sites of observed abnormal boric acid leakage are to be identified and documented in Appendix D, BAW Summary and Results. It is important to determine if any carbon steel RCS pressure boundary components are affected by the leakage, as well as whether the leak is active or not.

3.4 Performing Evaluations

3.4.1 Leakage areas may have affected carbon steel components. If carbon and low-alloy steel components are exposed to boric acid leakage, carefully clean the components and evaluate the extent that the carbon steel has been affected. Wall thickness, diameter measurements, localized corrosion depths, etc. of the affected area may be required to determine the extent of the wastage and its potential impact on component operability and structural integrity. For significant damage, ultrasonic and dye penetrant inspections of the affected components may be required to assess the impact. An Engineering evaluation of degraded or potentially damaged components will be completed in accordance with Procedure 81DP-0DC13, Deficiency (DF) Work Order.

3.5 Corrective Actions

3.5.1 Formal corrective actions will be required for cases in which boric acid leakage has resulted in significant degradation of carbon steel RCS pressure boundary components. In these cases, a Condition Report Disposition Request (CRDR) per procedure 90DP-0IP10, Condition Reporting, is to be initiated to document and trend the condition. The CRDR process will ensure that an Equipment Root Cause Failure Analysis (ERCFA) will be performed for the condition as appropriate. As part of the ERCFA program, corrective actions are identified to prevent recurrence of the condition.

3.5.2 A CRDR should be initiated if more serious leakage conditions are identified but degradation is only starting. These conditions include: borated water dripping on a hot metal surface, and borated steam impinging on hot metal surfaces. The above conditions, if left unattended, may result in corrosion rates exceeding one inch per year. Testing performed by EPRI/Westinghouse have found concentrated boric acid solutions at around 200° F to be highly corrosive. Conversely, boric acid corrosion rates of carbon steels in deaerated conditions, regardless of boron concentration or temperature, have been found to be very low.

The implementation of the PVNGS Boric Acid Corrosion Prevention Programs and the Visual Examination for Leakage, as described in the preceding responses, demonstrates compliance with Generic Letter 88-05, as well as the current requirements of 10 CFR 50.55(a), Section XI of the ASME Code, and the PVNGS Technical Specifications.

References

1. APS Letter No. 102-04702- GRO/SAB/RJR, "APS' Response to the Information Requested by NRC Bulletin 2002-01, Item 3.A." dated May 17, 2002, from Gregg R. Overbeck, APS to USNRC.

2. Letter 102-04681-GRO/SAB/RJR, "Response to Bulletin 2002-01: Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated April 03, 2002, from Gregg R. Overbeck, APS to USNRC.

3. APS Letter No. 102-04603-CDM/SAB/RJR, "Response to NRC Bulletin 2001-01: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," Dated September 4, 2001, from Gregg R. Overbeck, APS to USNRC.

4. Letter 102-04628-GRO/SAB/RJR, "Revised Inspection Schedule in Response to NRC Bulletin 2001-01," dated December 6, 2001, from Gregg R. Overbeck, APS to USNRC