September 4, 2001

2CAN090102

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station OP1-17
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
30 Day Response to NRC Bulletin 2001-01 for ANO-2; Circumferential Cracking of VHP Nozzles

Gentlemen:

On August 3, 2001, the Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." The bulletin requested information regarding the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles, inspections and repairs that have, or will be, completed to satisfy applicable regulatory requirements, and the basis for concluding that plans for future inspections will ensure compliance with applicable regulatory requirements.

Entergy Operations, Inc. (Entergy) recognizes the need to identify and correct any concerns with potential leakage through the reactor coolant pressure boundary to prevent long-term safety concerns and overall weakening of the boundary itself. Entergy is committed to ensuring the safe operation of all of its units and therefore will provide the appropriate level of attention and oversight to the issue. To date, Entergy has expended substantial resources in researching the existing Alloy 600 VHP cracking concern on Arkansas Nuclear One, Unit 1 which includes evaluating the structural acceptability of worst case expected flaws in these penetrations for ensuring safety. Even though our evaluation does not indicate there is an immediate safety concern, we agree that the concern must be addressed. The ultimate resolution of Alloy 600 cracking will require a dedicated and well-planned program for all reactor coolant system applications in our nuclear fleet. To this end, Entergy is currently working with Westinghouse to develop a weld overlay mitigation technique, which appears to be very promising in resolving future concerns with primary water stress corrosion cracking (PWSCC) initiated flaws at the wetted surface of the VHP nozzles.

As part of our research to address this concern, we have performed detailed analyses and calculations using the best tools available in the industry to determine whether an immediate safety concern might exist as a result of the inspection findings to date. Alloy 600 material, while susceptible to cracking, is an inherently tough material. The analysis shows that significant cracking can occur in a circumferential direction with the nozzle still having the ability to retain substantial safety margin. Due to significant hardships in the removal or modification of the reactor vessel head and nozzle insulation on Arkansas Nuclear One, Unit 2 (ANO-2), Entergy cannot reasonably perform a visual inspection of the area around the VHP nozzles as requested by the NRC bulletin. However, Entergy believes that an acceptable alternative would be to perform a sample inspection of the nozzles using either surface examination of the wetted surface or a demonstrated volumetric examination. Entergy has not identified any leaking VHP nozzles on ANO-2. Even though our inspection approach does not meet the full expectations of the NRC's bulletin, Entergy believes that this alternative inspection will provide a high confidence that VHP penetration leakage will be detected. Entergy will continue to evaluate VHP inspection findings within the industry and will use this information to determine the need to modify our inspection plans.

Attachment 1 provides the specific information requested by the NRC staff in NRC Bulletin 2001-01 for ANO-2. In addition, Attachment 2 provides our basis for why the regulatory requirements cited in the bulletin continue to be met. The next scheduled outage for ANO-2 will be in the spring of 2002.

This letter is submitted pursuant to 10 CFR 50.54(f) and contains information responding to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," for Arkansas Nuclear One, Unit 2. Commitments made in this letter are identified in Attachment 3.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 4, 2001.

Very truly yours,

[Original signed by Robert Bement
for Craig Anderson]

CGA/dbm
Attachments

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Response to NRC Bulletin 2001-01
Arkansas Nuclear One, Unit 2

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

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

ANO-2 Response

By letter dated August 21, 2001, the Nuclear Energy Institute (NEI) submitted EPRI Report TP-1006284 (MRP-48) [Ref. 1] on behalf of the industry to the NRC. This report provided an industry response to the information requested in Item 1a of the bulletin. Arkansas Nuclear One, Unit 2 (ANO-2) has been ranked for the potential for primary water stress corrosion cracking (PWSCC) of the reactor pressure vessel (RPV) top head nozzles using the time-at-temperature model and plant-specific input data reported in MRP-48.

Using the criteria stated in NRC Bulletin 2001-01, ANO-2 falls into the NRC category of plants with greater than 5 effective full power years (EFPYs) and less than 30 EFPYs until reaching the Oconee 3 time at temperature.

The vessel head temperature data used in determining the susceptibility ranking for ANO-2 was developed from CE-NSPD-1074, CEOG Task 953, "Evaluation of Reduction in Fluid Temperature in the Upper Plenum Due to Increased Bypass Flow," March 1997. CEOG Task 953 was conducted for Combustion Engineering (CE) units and determined the mixed-mean plenum temperature crediting bypass flow and mixing in the upper head region for all of the Combustion Engineering (CE) plants. The results of the computer code which modeled the thermal-hydraulic response of the head region, showed that the degree of cooling relative to the reactor outlet temperature varied from plant to plant as a function of their configuration and the amount of bypass flow expected. For ANO-2, the results estimated the mixed mean plenum temperature to be about 15°F below the reactor outlet temperature.

It is recognized that reactor head effective temperatures can vary based on the amount of bypass flow, head flow characteristics and flow channels in and around the head. Variations in bypass cooling can result in locally higher head temperatures. As part of the ANO-2 power uprate efforts currently in progress, Entergy had performed initial studies that indicated the potential for effective head temperatures being greater than that reported in Task 953. However, these studies were based on advanced modeling techniques that require additional benchmarking to validate the results and do not provide a consistent basis for industry susceptibility comparison. As discussed in NRC Bulletin 2001-01, the purpose of the susceptibility ranking is to provide a starting point for assessing the potential for VHP nozzle cracking in PWR plants.

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

ANO-2 Response

ANO-2 has 90 total reactor pressure vessel (RPV) head nozzles made up of 80 control element drive mechanism (CEDM) nozzles, nine incore instrumentation (ICI) nozzles and one head vent nozzle. The requested nozzle information is provided in Table 2-3 of MRP-48.

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

ANO-2 Response

As reported in Table 2-1 of MRP-48, ANO-2 has reflective contoured RPV head insulation.

The insulation on the ANO-2 RPV head is comprised of metal reflective panels and pliable insulation covered with glass cloth. The RPV head insulation is designed to conform to the curvature of the head, and is located underneath the CEDM cooling shroud orifice plate, which is also designed to match the curvature of the RPV head (see Figure 1). A plenum is provided at the outside diameter of the cooling shroud orifice plate which forces cooled air from the CEDM cooling system across the RPV head insulation and up through orifices in the shroud plate where the CEDM assemblies penetrate the cooling shroud. Removable insulation is only provided in the head flange and stud region, which does not interface with the CEDM or incore instrumentation nozzles.

Nozzle Insulation - Each of the 81 CEDM nozzles and 8 incore instrument nozzles are insulated with a prefabricated, pliable insulation collar which fits snugly around the base of each nozzle on the top of the RPV head. The head vent is encapsulated by the metal reflective insulation and does not have a collar. Each collar is custom fit to contour to the RPV head, and is comprised of Pittsburgh Corning TempMat insulation covered with fiberglass cloth lagging and (in some cases) stainless steel (SS) wire. The inside diameter of the collar insulation is comprised of a continuous cylinder of 24-gauge SS. The nozzle insulation collars terminate approximately 3/4" below each CEDM nozzle omega seal weld lip. The height of the insulation collars range from 1-15/16" to 11-7/16" depending on the location of the nozzle on the RPV head dome region.

Head Dome Insulation - The 89 insulation collars described above are enclosed by prefabricated SS reflective insulation panels, which insulate the RPV head. The openings in the panels are sized to fit just over the CEDM nozzle insulation collars. The panels are provided with lap joints to provide an airtight seal over the head region. The single panel at the top of the dome is approximately 56" in diameter and interfaces with 21 CEDM nozzles. Two additional tiers of panels are provided on the dome region of the head, with the upper tier consisting of 8 panels (outside diameter of 98.5”), and the lower tier consisting of 8 panels (outside diameter of 129.5”).

Insulation/cooling shroud interface - The CEDM cooling shroud is installed on top of the RPV head above the insulation components. The bottom plate conforms to the head curvature, and has orifices, which allow the CEDM and incore instrumentation nozzles to pass through. The CEDM cooling shroud orifice plate openings around the 81 CEDM nozzles range from 9.41” to 9.86” in diameter, while the outside diameter of the CEDM assembly at the orifice elevation is slightly greater than 7 inches. The tops of the nozzle insulation collars are approximately 7 to 8 inches below the bottom of the orifice plate. The maximum diameter of the CEDM coil stack assemblies is 10.25”, and the bottom of the coil stack assemblies is approximately 2 to 3 inches above the top of the orifice plate. Access to the nozzle collars from above the orifice plate is restricted due to the size and location of the orifice plate openings. Below the orifice plate, the lane between the insulated nozzles is approximately 2.5” wide, which makes the area inaccessible.

Inspection options and inherent difficulties – In order to provide access to the CEDM nozzles for “bare metal” inspections, several alternatives have been assessed which involve the removal/replacement and/or modifications to the RPV head insulation and/or CEDM cooling shroud.

The alternatives were evaluated and determined to be infeasible due to a combination of each of the following:

- The RPV head insulation, CEDM cooling shroud, and CEDM assembly installation sequence was such that the insulation and shroud were constructed prior to CEDM assembly installation. As such, the cooling shroud cannot be removed, or lifted more than 2 or 3 inches, with the CEDM coil stack assemblies in place. Without the cooling shroud removed, the majority of the insulation components could only be destructively removed, since the metal reflective insulation panels were designed to be installed over the nozzles prior to CEDM assembly and cooling shroud installation. Assuming existing insulation panels and nozzle collars could be destructively removed, a suitable insulation replacement system would be very difficult to design which would meet the insulation design requirements, and/or facilitate re-installation with the cooling shroud and CEDM assembly components in place.
- Access to the insulation collars is extremely restricted by the close proximity of the CEDM assemblies, the presence of the cooling shroud orifice plate, and the narrow lane between the nozzle insulation collars (about 2.5 inches). Additional access ports on the exterior of the cooling shroud would not improve access to perform inspections.
- Cutting a gap in the insulation collars would invalidate an assumption in the CEDM thermal analysis that assumes perfect insulation (no air gap). The CEDM thermal and stress analysis would have to be evaluated to assess the effects on the CEDM components. Additionally, CEDM cooling would likely be significantly impacted by introducing convection paths from the RPV head to the cooling shroud envelope via gaps between the nozzles and insulation.

- Due to the assembly sequence of the CEDM insulation components and the CEDM assemblies, and the configuration of the cooling shroud, removal and replacement of damaged insulation collars would not be possible without disassembly of the cooling shroud and CEDM assembly components.
- Due to the configuration of the CEDM nozzles, there is little or no allowance to redesign the RPV dome/CEDM nozzle insulation and CEDM cooling shroud configuration in a manner which could effectively change the current configuration and improve access to the RPV head for CEDM nozzle inspections.
- A review of the CEDM cooling shroud orifice plate design calculation indicates that modifications to the insulation and CEDM cooling shroud would be logistically difficult, very dose intensive, expensive, and would not result in any significant access improvement.

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

ANO-2 Response

As reported in Table 2-1 of MRP-48, ANO-2 has not performed enhanced visual RPV head and nozzle inspections.

ANO-2 has performed inspections developed in accordance with Generic Letter (GL) 88-05 and GL 97-01 requirements. Due to the construction of the head, head insulation, and the closure head assembly as discussed above, inspection of the ANO- 2 nozzles cannot be readily performed without significant cost and hardship to Entergy. The inspections are performed by viewing the top of the insulation and the orifice plate. The engineer looks for boric acid coming from the insulation pinched between each orifice plate and nozzle. Leakage from a nozzle could be forced through existing openings in the insulation or create an opening through the insulation and appear at the nozzle. Boric acid from a leak could also migrate down the top surface of the head to the outside perimeter of the head. This would cause boric acid to appear from beneath insulation around the head. The perimeter of the insulation around the head for boric acid is also inspected. These inspections are conducted immediately after shutdown from every refueling outage. No nozzle leakage has been identified from performing these inspections.

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

ANO-2 Response

General Description:

The CEDMs attach to the top of the head at the CEDM nozzles (see Figure 1). The CEDM housings are supported by the nozzles. The cables to the CEDM housings are fed from the cable trays on either side of the refueling canal over to the connections on the bulkheads on the maintenance structure. The cables are supported by the maintenance structure as they travel down and connect to the top of the CEDM housings at the top of the CEDM structure. The missile shield is located above the maintenance structure and is supported by the secondary shield walls. Containment chilled water is supplied to the CEDM coolers and fans located on top of the missile shield. CEDM cooling air ducts run from the coolers on top of the missile shield down to the cooling shroud just above the reactor vessel head.

Missile Shield:

The missile shield for ANO-2 is fabricated in one piece and is movable. It is 2’-0” thick, 28’6” wide (spanning the refueling canal), and is 28’-0’ long. It is centered above the reactor vessel and is supported by the secondary shield walls. It rolls on eight crane wheels (four on each side) that travel on crane rails anchored on the top of the secondary shield walls. The missile shield is made of reinforced concrete poured within a steel beam frame enabling it to span the refueling canal opening. The missile shield also supports the CEDM cooling air system of chillers, blowers, and ductwork. The shield is rolled to the end of the refueling canal during refueling operations. The bottom of the missile shield is at elevation 426’-6 1/2”.

CEDM Housings and Their Support/Restraint System

The CEDM housings are attached to the top of the CEDM nozzles at the top of the reactor vessel head. The nozzles provide all of the support for the housings. The housings extend up through the head lifting structure and end at various heights above the platform level approximately elevation 400’-9”. The CEDM housings are free standing in that there are no other supports or restraints for the housings within the head lifting structure or at the platform level.

Other Components:

The containment chilled water piping is supported off the south secondary shield wall. The supply and return lines are 6” diameter and the centerline of the pipes is at elevation 420’-2” before they turn up at the edge of the missile shield and connect to the flanges for the CEDM coolers on top of the missile shield.

The reactor head venting piping comes out of the CEDM nozzle area at elevation 382’-6 ¼” through two flanges until it intersects the south wall of the refueling canal. There it turns upward to elevation 384’-0” and runs west along the wall until it turns upward to exit the refueling canal. This line connects with the pressurizer vent line.

Other components in the area are the communication and lighting conduits and connections along the south and north refueling canal wall for access from the maintenance structure and refueling bridge. They are located at elevations 422'-11" and 412'-0" respectively. The connections provide lighting and communication for refueling operations.

Other Structures:

Other structures within the refueling canal above the reactor vessel include the cable trays carrying the cables to the reactor vessels and cooling ducts. The cable trays extend along the top of the refueling canal on both sides and are supported off the secondary shield walls. The bottom of the cable trays are located at approximately elevation 421'-6".

The CEDM cooling ducts exit the missile shield to the CEDM cooling shroud and are supported off the missile shield and the refueling canal floor. These ducts are round and connect to the rectangular ducts coming out of the CEDM coolers and fans supported on top of the missile shield. These ducts supply cooling air for the CEDM housings and nozzle area. Two ducts run down on the east side of the missile shield and one duct runs down the west side of the missile shield. The vertical ducts are removed and stored out of the way for refueling operations.

There are rectangular cooling ducts running along both sides of the refueling canal wall. They supply cooling air for the refueling canal area and are supported off the secondary shield walls. The bottom of the cooling ducts is at elevation 416'-1".

The maintenance structure is suspended above the service structure and reactor vessel head. Steel beams are embedded into the secondary shield walls and cantilever into the refueling canal at elevation 416'-0" (top of steel) to support the maintenance structure. The maintenance structure is constructed of steel beams and columns and provides platforms intended to provide maintenance access to the CEDMs and cables. The maintenance structure provides supports for the cables to the reactor vessel and provides access to the connections at the cable trays at the top of the structure and the top of the CEDMs. Guide structures for the maintenance structure are supported off the secondary shield walls on each side of the refueling canal to position the maintenance structure as it is installed/removed for refueling.

Other Cabling

The CEDM control and power cables enter the refueling canal area in the cable trays that extend along the top of the secondary shield walls. Also contained in these cable trays are the incore instrumentation cables and other instrumentation cabling going to the reactor vessel head area. These cables enter the maintenance structure at the top at the bulkheads. From the bulkhead area the cables are supported by the maintenance structure down to the connections at the top of the CEDM service structure.

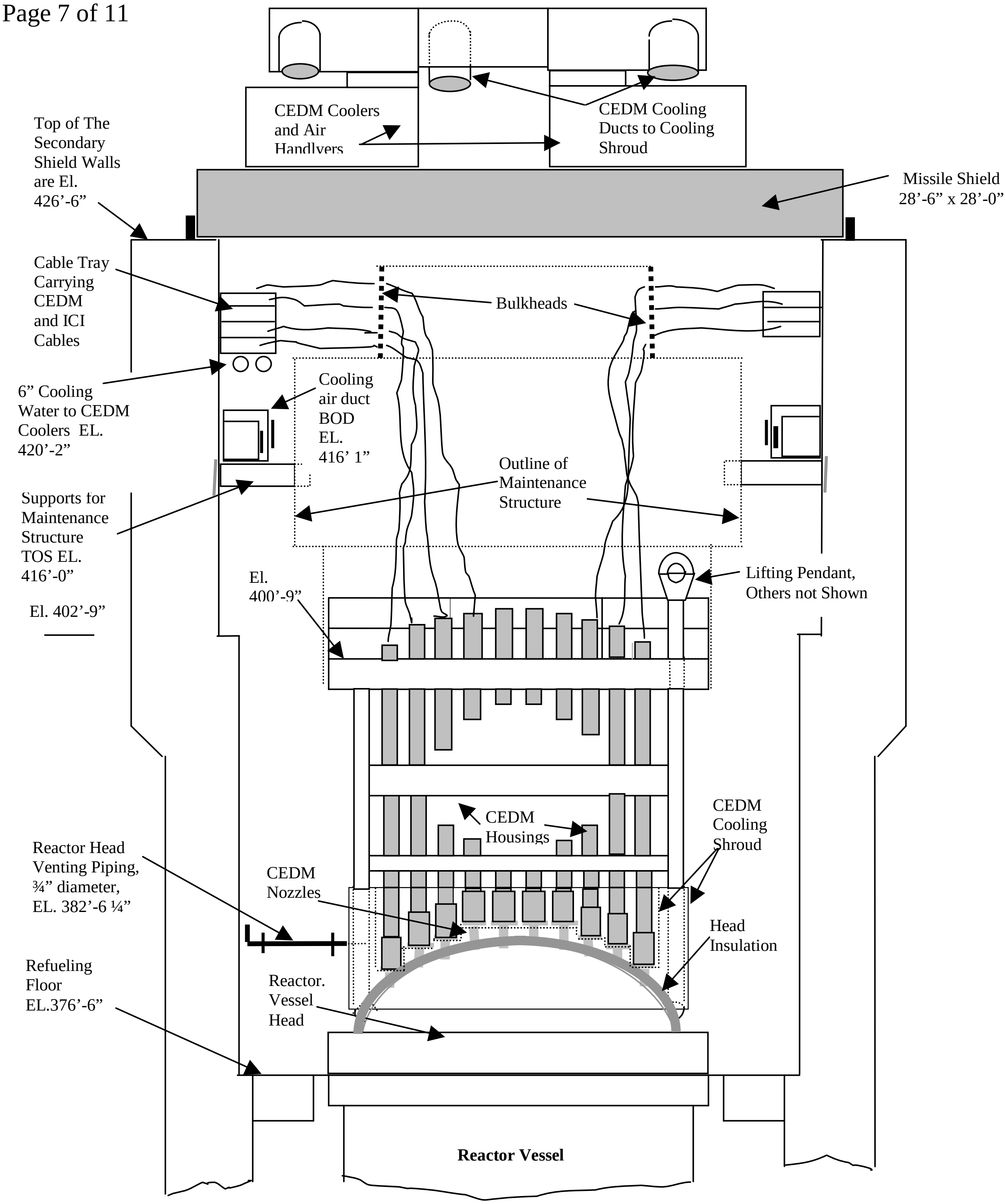

FIGURE 1

General View of ANO-2 Structures and Components Above the Reactor Vessel

Looking West (Not to Scale)

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

Not Applicable to ANO-2

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

Not Applicable to ANO-2

NRC Request 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 information:

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

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

ANO-2 Response

Type and Scope of Inspections

Due to the significant dose impact, monetary costs, and hardships that would be incurred by removing or modifying the head insulation, a sample of 25% (22 penetrations) of the CEDM/ICI nozzle penetrations will be examined in 2R15 (spring 2002). The method of inspection will be a demonstrated surface examination of the wetted surface or a demonstrated volumetric examination method that is capable of detecting axial and circumferential cracks. The locations will be those judged most susceptible to PWSCC. If conditions are discovered in the first sample set that require repair based on the acceptance criteria below, then an additional number of penetrations equal to the initial sample set will be examined. If any additional unacceptable flaws as described above are found in the expanded sample, Entergy will examine the remaining CEDM/ICI penetrations. The scope of surface or volumetric examination need not be expanded when a flaw is located below the J-groove weld.

Acceptance Criteria

Flaws identified by nondestructive examination methods, which are beyond current requirements, will be evaluated in accordance with the flaw evaluation rules for piping contained in Section XI of the ASME Code. This approach has been accepted by the NRC for other similar applications. Flaws not meeting requirements for the intended service period will be repaired before returning them to service.

The NEI and NRC staff acceptance criteria from GL 97-01 will be used for inside diameter (ID) connected axial flaws found in a non-leaking VHP. These criteria require NRC notification following the identification of any circumferential flaw that may be left in place.

If during the conduct of the described examinations, conditions not attributed to leakage are identified, they will be evaluated using the acceptance criteria of IWB-3523, "Standards for Examination Category B-O, Pressure Retaining Welds in Control Rod Drive Housings" as supplemented below. Even though this criteria is not specific to the intended application it is considered appropriate. Any indications determined to be indicative of throughwall leakage will be corrected in accordance with IWA-5250. Partial throughwall flaws not meeting the criteria of IWB-3523 may be evaluated and dispositioned in accordance with IWB-3600.

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

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

Bases for the ANO-2 Inspection Plan

- Examination of a sample of the CEDM/ICI nozzles using a demonstrated surface or volumetric method is considered an acceptable alternative to an effective visual examination of 100 percent of the nozzles. A sample inspection using a demonstrated method provides a high degree of confidence that cracking will be detected, if present.

- Alloy 600 CEDM penetrations are similar to reactor coolant austenitic piping in that the material has high fracture toughness thereby making it extremely flaw tolerant. Field experiences (the large crack at Duane Arnold plant as well as numerous SCC cracks at BWRs and PWRs in Alloy 600) and fracture mechanics analyses have verified the flaw tolerance of Alloy 600. To date, fracture mechanics evaluation of the largest circumferential crack found at Oconee has shown that a significant time period exists for the flaw to grow to an extent that ASME safety margins are reached and an even longer time period to reach instability.

- A probabilistic fracture mechanics evaluation is in progress by the EPRI Materials Reliability Program that will provide an estimate of the likelihood of a pipe rupture in the CEDM penetrations. The evaluation approach will use the failure probability for SCC using the PRAISE and SARA computer codes.

- The assumption that an initiating event frequency of 1 for a rupture of a CEDM penetration made by the NRC is extremely conservative. Historically, complete pipe breaks have been estimated from data at a frequency of about $1X10^{-5}$ /reactor year. CEDM penetrations made from Alloy 600 and having PWSCC cracks have been in service for about 10 years in both domestic and foreign reactors. To date, no pipe ruptures associated with Alloy 600 have ever occurred and leakage from PWSCC penetration cracking has been found well in advance of when a rupture might be expected. Therefore, based on expert opinion, a pipe rupture frequency on the order of 10^{-2} - 10^{-3} /reactor year is considered a more reasonable yet conservative estimate. Therefore, a more realistic core damage frequency estimate resulting from a CEDM nozzle ejection is estimated to be in 10^{-5} - 10^{-6} /reactor year which is consistent with the NRC safety goal.

- CEDM penetration cracks can be evaluated using limit load methodology similar to that used to evaluate austenitic stainless steel degraded piping. Therefore, the selection of sampling criteria modeled after that contained in the ASME Boiler and Pressure Code, Section XI is appropriate for this application.

- For determining a sample size using binomial statistics (Ref. 2) for a large population, the number of tests required with a confidence level of 90% and a reliability goal of 0.9, would require testing twenty-two (22) parts. In addition, using the susceptibility ranking data of reference 1 and the Weibull statistics test plan matrix for zero failures, a sample size representing a 90% confidence level is 15 (β = 2.0). Both of these analyses demonstrate that a twenty five percent (25%) initial sample size (22 penetrations) provides sufficient assurance to ensure the detection of degraded condition.

- A comparison of the axial stress magnitude between the ANO-1 analysis (Ref. 4) and for a typical CE CEDM (Ref. 3) indicates that the stresses in the peripheral CEDM nozzle is substantially lower than those obtained for the ANO-1 peripheral nozzle. The axial stress magnitude in the ICI nozzle for the CE design is lower than that for the peripheral ANO-1 nozzle but the distribution is opposite to that of the ANO-1 nozzle. Entergy has a project in progress to determine the residual stress distributions for both the ANO-2 and Waterford-3 design utilizing the same methodology that was utilized to determine the residual stress distribution for the ANO-1 CRDM. The differences in the modeling method are expected to simulate the joint in a more realistic manner and therefore provide a better estimate of the residual stress profile. Previous experience gained from the ANO-1 analysis effort indicates that there would be a reduction in the magnitude of the stresses. The ANO-1 analysis is summarized in letter dated September 4, 2001, which was also provided in response to NRC Bulletin 2001-01.

- The fracture mechanics analysis performed for the ANO-1 nozzle, using the finite element analysis results for a refined mesh, showed that for a throughwall circumferential flaw the stress intensity factor decreased as the flaw extended around the circumference. This trend is expected since the stress distribution at the uphill location was higher and that at the downhill location was significantly lower. The experience gained from the ANO-1 analysis indicates that a similar trend would be expected for the CE nozzles except the higher stress intensity factor would be at the downhill location. Since the magnitude of the stresses are lower for the CE nozzles, the magnitude for the stress intensity factor is likewise expected to be lower. The reduced stress intensity factor for ANO-2, together with a lower normalized EFPY for ANO-2 compared to ANO-1, suggests that any significant degradation of the CEDM nozzles is unlikely at this time. Hence, a reasonable sample size would be adequate to ensure the detection of degraded condition. The structural analysis information will be used to select the penetration locations for the inspection samples.

Entergy has also addressed each of the specific regulatory requirements cited in NRC Bulletin 2001-01 in Attachment 2 of this letter.

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

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

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

ANO-2 Response:

Entergy will provide the requested information for ANO-2 or indicate that no leakage was identified within 30 days after plant restart following the next refueling outage, which is currently scheduled to begin in spring 2002.

References

1. "PWR Materials Reliability Program Response to NRC Bulletin 2001-01. {MRP-48}"; EPRI TR-1006284; Electric Power Research Institute; August 2001.

2. "The New Weibull Handbook"; R. A. Abernethy; Gulf Publishing Company; Houston, TX. 1993

3. "Stress Analysis Alloy 600 CEDM and ICI Nozzles"; Prepared for Combustion Engineering Owners Group; DEI-357; Dominion Engineering Inc.; May 1993.

4. "Entergy Nuclear Southwest Design Input Description and Responsibilities for ANO Unit 1 CRDM Nozzle Flaw Evaluation"; April 2001.

Basis for Concluding That Regulatory Requirements as Cited in NRC Bulletin are Met

The NRC Bulletin 2001-01 section entitled Applicable Regulatory Requirements cites the following regulatory requirements and plant commitments as providing the basis for the Bulletin assessment:

- Appendix A to 10 CFR Part 50, General Design Criteria for Nuclear Power Plants
  Criterion 14 - *Reactor Coolant Pressure Boundary*
  Criterion 31 - *Fracture Prevention of Reactor Coolant Boundary, and*
  Criterion 32 - *Inspection of Reactor Coolant Pressure Boundary*
- Plant Technical Specifications
- 10 CFR Part 50.55a, Codes and Standards, which incorporates by reference Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code
- Appendix B of 10 CFR Part 50, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Criterion V, IX, and XVI

This section discusses how the cited regulatory requirements and plant commitments affect plant decisions relating to addressing NRC Bulletin 2001-01 requested actions and regulatory compliance.

General Design Criteria

Criterion 14 – Reactor Coolant Pressure Boundary

> *"The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."*

Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

> *"The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws."*

<u>Criterion 32</u> – Inspection of Reactor Coolant Pressure Boundary

"Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure boundary."

During licensing of the plant, Entergy demonstrated that the design of the reactor coolant pressure boundary meets these requirements. The following information demonstrates how Entergy complies with the design criteria for the cracking of RPV top head nozzles:

- Pressurized water reactors licensed both before and after issuance of Appendix A to Part 50 (1971) complied with these criteria in part by: 1) selecting Alloy 600 or other austenitic materials with excellent corrosion resistance and extremely high fracture toughness, for reactor coolant pressure boundary materials, and 2) following ASME Codes and Standards and other applicable requirements for fabrication, erection, and testing of the pressure boundary parts. NRC reviews of operating license submittals subsequent to issuance of Appendix A included evaluating designs for compliance with the General Design Criteria. The SRPs (standard review plans) in effect at the time of licensing do not address the selection of Alloy 600. They only require that ASME Code requirements be satisfied. It should be noted that the ASME Code does not consider localized forms of corrosion in design; suitability of material for these types of corrosion was left to the Owner. The only guidance regarding stress corrosion cracking was that contained in the SRP for austenitic stainless steel.
- Although stress corrosion cracking of primary coolant system penetrations was not originally anticipated during plant design, it has occurred in the RPV top head nozzles at some plants. The suitability of the originally selected materials has been confirmed. The robustness of the design has been demonstrated by the small amount of leakage that has occurred and by the fact that none of the cracks in Alloy 600 CEDM reactor coolant pressure boundary materials has rapidly propagated, encroached on ASME Code safety margins, or resulted in catastrophic failure or gross rupture. It should be noted that earlier versions of the GDCs are in terms of extremely low probability of gross rupture or significant leakage throughout its design life.

As described above, the requirements established for design, fracture toughness, and inspectability in GDC 14, 31, and 32 respectively were satisfied during ANO-2's initial licensing review, and continue to be satisfied during operation, even in the presence of a potential for stress corrosion cracking of the RPV top head penetrations.

Technical Specification Requirements

The Bulletin states:

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

10CFR50.36 contains requirements for plant Technical Specifications. Paragraphs (c)(2) and (c)(3) of 10CFR Part 50.36 are particularly relevant:

10CFR 50.36 (c)(2) Limiting Conditions for Operation

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the Technical Specifications until the condition can be met. A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one of the following criteria:

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

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

10 CFR 50.36 (c)(3) Surveillance Requirements

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

The reactor coolant system boundary provides one of the critical barriers that guard against the uncontrolled release of radioactivity and is relied upon for defense in depth in limiting risk. Therefore, ANO-2 Technical Specifications include a limiting condition for operation and associated action statements addressing reactor coolant system leakage. Per Technical Specification 3.4.6.2 the limits for reactor coolant system leakage are stated in terms of the amount of leakage: 1 gallon per minute (gpm) for unidentified leakage; 10 gpm for identified leakage; and no pressure boundary leakage. The bases for Technical Specification 3.4.6.2 state in part:

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 gpm identified leakage limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage by the leakage detection systems.

Pressure boundary leakage of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any pressure boundary leakage requires the unit to be promptly placed in cold shutdown.

Technical Specification 3.4.6.2 provides requirements for action when leakage is found. These actions are:

With any pressure boundary leakage, be in at least hot standby within 6 hours and in cold shutdown within the following 30 hours.

With any reactor coolant system leakage greater than any one of the above limits, excluding pressure boundary leakage, reduce the leakage rate to within limits within 4 hours or be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours.

Operability requirements, applicability, and actions for the leakage monitoring systems per Technical Specification 3.4.6.1 are as follows:

The following reactor coolant system leakage detection instrumentation shall be operable:

a. *One containment sump level monitor,*

b. *One containment atmosphere particulate radioactivity monitor, and*

c. *One containment atmosphere gaseous radioactivity monitor.*

Applicability: Modes 1, 2, 3 and 4.

Action:

a. *With one or more containment atmosphere radioactivity monitor(s) inoperable, operation may continue for up to 30 days for each inoperable monitor provided:*

  1. *grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours, or*

  2. *a reactor coolant system water inventory balance is performed at least once per 24 hours in accordance with surveillance requirement 4.4.6.2.1.a;* otherwise, be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours.*

b. *With the containment sump level monitor inoperable, operation may continue for up to 30 days provided a reactor coolant system water inventory balance is performed at least once per 24 hours in accordance with surveillance requirement 4.4.6.2.1.a;* otherwise, be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours.*

c. *With the containment sump level monitor inoperable and one containment atmosphere radioactivity monitor inoperable, operation may continue for up to 30 days for each inoperable monitor provided a reactor coolant system water inventory balance is performed at least once per 24 hours in accordance with surveillance requirement 4.4.6.2.1.a;* otherwise, be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours.*

Technical Specification Bases 3.4.6.1 for the reactor coolant system leakage detection systems state in part:

> *GDC 30 of Appendix A to 10 CFR 50 requires means for detecting and, to the extent practical, identifying the location of the source of RCS leakage. The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" May 1973. Likewise, the actions implemented upon inoperability of a required leak detection instrument are sufficient in maintaining the diversity and accuracy needed to effectively detect RCS leaks.*
>
> *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. In addition, the reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Instrument sensitivities of 10 – 10^6 cpm for particulate and gaseous monitoring are practical for these leakage detection systems.*

Most leaks from reactor coolant system Alloy 600 CRDM penetrations have been well below the sensitivity of on-line leakage detection systems. Further, the leakage is not easily detectable with the visual examinations associated with ASME Code required examinations or pressure tests. Entergy has evaluated this condition and has determined that visual examinations are not practical for the upcoming refueling outage as an acceptable alternative, the inspections planned will be a surface or volumetric examination of a sample of the CEDM penetrations during the next scheduled refueling outage. Field experience and analysis have demonstrated that most PWSCC in Alloy 600 CEDM penetrations and its weldments are axial in nature as predicted by analysis and confirmed through observation by NDE and destructive metallurgical analyses. In those cases where the cracking has progressed to the OD of the tube and propagated as driven by the highest stresses present, the cracking could challenge the tube integrity if uncorrected. Evaluation of the most severe circumferentially oriented cracking found has demonstrated that margins exceeding those required in the ASME Code are present for tube integrity. Further, probabilistic fracture mechanics evaluations of the CEDM cracking have demonstrated that the initiating event frequency is low, and is well below the event frequency of 1 assumed by the staff in its review. Supplemental ongoing evaluations are considering a range of crack growth rates, flaw sizes and the initiation of multiple cracks on the pipe OD, although this is considered unlikely based on the stresses driving the cracking.

If through-wall pressure boundary leaks of CEDMs increase to the point where they are picked up by the on-line leak detection systems, then the leak must be evaluated per the specified acceptance criteria, and the plant be shut down if it is a pressure boundary fault.

ANO-2 has met and will continue to meet Technical Specification requirements for reactor coolant system leakage. If pressure boundary leakage from the reactor coolant system is detected or if leakage exceeds limitations during plant operation, appropriate action statements will be followed. If indications are found to be unsuitable for continued service by analysis they must be repaired before the plant goes back on line. Further, the root cause would be identified, an evaluation would be performed to define any necessary inspections and evaluation of the inspection findings, and further analyses needed to determine that there is reasonable assurance of a low probability of abnormal (significant) leakage and of loss of structural integrity over the next intended period of plant operation.

10 CFR 50.55a/ASME Code, Section XI

The Bulletin states:

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

10CFR50.55a requires that inservice inspection and testing be performed per the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, *Inservice Inspection of Nuclear Plant Components*. Section XI contains applicable rules for examination, evaluation and repair of code class components, including the reactor coolant pressure boundary.

ANO-2 is in its third Inservice Inspection Interval and is committed to the 1992 Edition with portions of the 1993 Addenda of ASME Section XI. By this Edition and Addenda of the Code, Examination Category B-E has been deleted and pressure testing with VT-2 examination is now performed under Examination Category B-P as part of the reactor vessel pressure retaining boundary. The ASME Code requires a System Leakage Test in accordance with IWB-5220 and acceptance of discovered conditions in accordance with IWB-3522. For systems borated for the purpose of controlling reactivity, the Code requires the insulation at bolted connections to be removed for the VT-2 examination. For other components (which includes the CEDMs) the Code allows the VT-2 examination to be performed without removal of the insulation by examining the accessible and exposed vessel surfaces and joints of the insulation.

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

In addition to the inspection requirements of ASME Section XI, ANO-2 performs visual inspections for evidence of leakage by examining the RPV top head surface, or the insulation per the requirements of NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*

The Code acceptance standard for the VT-2 visual examination is found in Paragraphs IWB-3522.1, *Visual Examination, VT-2* and IWA-5250, *Corrective Actions*. While the NRC Bulletin references IWB-3142 implying that the licensee may use supplemental examinations, and analytical evaluations to accept the leaking condition, IWA-5250, *Corrective Action* is the more appropriate reference which requires through wall leaks to be corrected by either repair or replacement.

Flaws identified by nondestructive examination methods, which are beyond current requirements, are evaluated in accordance with the flaw evaluation rules for piping contained in Section XI of the ASME Code. The NRC has accepted this approach. Any flaw not meeting requirements for the intended service period would be repaired before returning it to service.

Entergy complies with, and will continue to comply with these ASME Code requirements through implementation of the plant's inservice inspection program. If a VT-2 examination detects the conditions described by IWB-3522.1(c) and (d), then corrective actions per IWA-5250 would be performed in accordance with the plant's corrective action program. Further, defects found from any examination of the CEDMs would be evaluated to Section XI criteria for continued service, or repaired to ASME Code requirements or with alternative repair methods approved by the NRC. No new plant actions are necessary to satisfy the cited regulatory criteria.

<u>10 CFR 50, Appendix B</u>

The Bulletin states:

> *"Criterion IX of Appendix B to 10 CFR Part 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of VHP nozzles, special requirements for visual examination would generally require the use of a qualified visual examination method. Such a method is one that a plant-specific analysis has demonstrated will result in sufficient leakage to the RPV head surface for a through-wall crack in a VHP nozzle, and that the resultant leakage provides a detectable deposit on the RPV head. The analysis would have to consider, for example, the as-built configuration of the VHPs and the capability to reliably detect and accurately characterize the source of the leakage, considering the presence of insulation, preexisting deposits on the RPV head, and other factors that could interfere with the detection of leakage. Similarly, special requirements for volumetric examination would generally require the use of a qualified volumetric examination method, for example, one that has a demonstrated capability to reliably detect cracking on the OD of the VHP nozzle above the J-groove weld."*

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Criterion IX is a forward-looking requirement such that if inspections are performed they must be controlled and accomplished by qualified personnel. No action is required to satisfy this criterion unless a new inspection is proposed. However, if a new inspection were utilized, appropriate qualification for inspection personnel would be established in accordance with Criterion IX. Sufficient analysis and demonstration of the method would be performed to demonstrate its capability.

The Bulletin further states:

> *"Criterion V of Appendix B to 10 CFR Part 50 states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Criterion V further states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual and volumetric examinations of VHP nozzles are activities that should be documented in accordance with these requirements."*

Criterion V is also a forward-looking criterion that applies should the Bulletin response identify new inspections. It does not establish criteria for when or if inspections should be performed. If new inspections are performed, they will meet Criterion V.

The last Appendix B criterion cited in the Bulletin is:

> *"Criterion XVI of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For cracking of VHP nozzles, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future cracking. These actions could include proactive inspections and repair of degraded VHP nozzles."*

Criterion XVI has two attributes that should be considered by licensees in their response to the Bulletin.

The first attribute is *that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected.* This criterion infers a licensee's responsibility to be aware of industry experience, and has been interpreted in this manner in most plant's corrective action programs. A licensee should determine if an industry experience applies to its plant and what, if any, corrective actions are appropriate. This approach is consistent with the NRC's generic communication process for an Information Notice, which reports industry experience, but does not require a response to the NRC. Licensees are expected to evaluate the applicability of the occurrence to their plant, and document a record of the plant specific assessment for possible NRC review during inspections.

Criterion XVI provides the objectives and goals of the corrective action program, but licensees are responsible for determining a specific process to accomplish these goals and objectives. With regard to the Bulletin response, Criterion XVI does not provide specific guidance as to what is an appropriate response, but rather, the licensee is responsible for determining actions necessary to maintain public health and safety. That is, the licensee must justify its actions for addressing the stress corrosion cracking of vessel head penetrations. Furthermore, the regulatory criteria of 10 CFR 50.109(a)(7) provide supporting evidence, where it states that *if there are two or more ways to achieve compliance . . . then ordinarily the applicant or licensee is free to choose the way which best suits its purposes.*

The second attribute of Criterion XVI that should be considered is that for significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. The Bulletin suggests that for cracking of vessel head penetrations, the root cause determination is important in understanding the nature of the degradation and the required actions to mitigate future cracking. As part of its corrective action program, a licensee, through its own efforts or as part of an industry effort, would determine the cause of cracks in the vessel head penetrations, if they were detected. However, if no known cracks in the heads are identified through reasonable quality assurance measures or inspection and monitoring programs, this criterion would not require specific action on the part of a licensee for remaining in compliance with the regulation.

In summary, the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CEDM nozzle cracking is clearly in compliance with the performance-based objectives of Appendix B.

Generic Letter 91-18, Revision 1 *Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions*, the Staff has clarified that in all success paths, whether specifically stated or not, are first directed to ensuring public health and safety and second to restoring the SSCs to the current licensing basis of the plant as an acceptable level of safety. It further clarifies that when a degraded or nonconforming condition of an SCC subject to Appendix B to 10CFR50 is identified, Appendix B requires prompt corrective action to correct or resolve the condition.

> *The licensee must establish a time frame for completion of the corrective action. The timeliness of this corrective action should be commensurate with the significance of the issue…the NRC will consider whether corrective action was taken at the first opportunity, as determined by safety significance, and by what is necessary to implement the corrective action. Factors that might be included are the amount of time required for design, review, approval, or procurement of the repair/modification; availability of specialized equipment to perform the repair, etc.,.*

In keeping with the criteria of Appendix B to 10 CFR Part 50, with the additional guidance of Generic Letter 91-18, for ANO-2, the inspections described in response to NRC Bulletin item 4.a are adequate to address the potential for PWSCC of the CEDMs.

Commitments Contained in this Letter

Commitment	TYPE		Scheduled Completion Date (If Required)
	One Time Action	Continuing Compliance	
ANO 2 will perform a surface or volumetric examination of a 25% sample (22 CEDM penetrations) during the next scheduled refueling outage. This will include the more susceptible penetrations.	X		Spring 2002
ANO-2 will provide the information requested in Item 5 of NRC Bulletin 2001-01 or indicate that no leakage was identified within 30 days after plant restart following the next refueling outage, which is currently scheduled to begin in Spring 2002.	X		30 days after next refueling outage