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Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
30 Day Response to NRC Bulletin 2001-01 for ANO-1; 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, including the extent of nozzle leakage and cracking that has been found to date, inspections and repairs that have been completed to satisfy applicable regulatory requirements, and the basis for concluding that plans for future inspections will ensure compliance with applicable regulatory requirements.

As you are aware, Arkansas Nuclear One, Unit 1 (ANO-1) experienced leakage through one of the control rod drive mechanism (CRDM) nozzles in our most recent refueling outage and appropriate repairs were made. This was reported to the NRC in a licensee event report dated May 8, 2001. 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 address the VHP nozzle cracking concern, Entergy has performed detailed analyses and calculations using advanced analytical tools 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 analyses show that significant cracking can occur in a circumferential direction with the nozzle still having the ability to retain substantial safety margin. The current ANO-1 inspection program for performing bare head visual inspections is believed to be satisfactory to identify any flaws well before they could become an actual safety concern. However, Entergy will continue to evaluate VHP nozzle inspection findings within the industry to determine whether new information would alter our safety conclusions indicating the need to modify our inspection plans.

Even though our evaluation does not indicate there is an immediate safety concern, we agree that the Alloy 600 cracking issue must be addressed. The ultimate resolution of this issue will require a dedicated and well-planned program for all reactor coolant system applications. To this end, Entergy is also 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 CRDM nozzles.

The ANO-1 response to NRC Bulletin 2001-01 is provided in Attachment 1. The next scheduled outage for ANO-1 will be in the fall of 2002. Since ANO-1 experienced a leak at a CRDM nozzle, which was discovered during an outage in early 2001, ANO-1 is being requested to justify not performing a volumetric inspection of all nozzles prior to the end of 2001. As described in the attachment, Entergy believes that the ANO-1 visual inspection capability used in previous, as well as future inspections, meets regulatory requirements. Additionally, the structural analysis performed for ANO-1 on Alloy 600 nozzles demonstrates that any flaw that would develop during the operating cycle will not represent a safety concern during the period between refueling outage inspections. If the NRC desires, Entergy is willing to discuss this matter with the NRC either by phone or at the NRC offices.

Attachment 2 provides the ANO-1 perspective for complying with regulatory requirements cited in NRC Bulletin 2001-01. Commitments associated with the bulletin response are contained in Attachment 3 to this letter.

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 ANO-1. 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/sab
Attachments

cc: Mr. Ellis W. Merschoff
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Arkansas Nuclear One
P.O. Box 310
London, AR 72847

Mr. William Reckley
NRR Project Manager Region IV/ANO-1
U. S. Nuclear Regulatory Commission
NRR Mail Stop O-7 D1
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Response to NRC Bulletin 2001-01
Arkansas Nuclear One, Unit 1

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-1 Response

By letter dated August 21, 2001, the Nuclear Energy Institute (NEI) submitted EPRI Report TP-1006284, *PWR Materials Reliability Program Response to NRC Bulletin 2001-01, August 2001* (MRP-48) on behalf of the industry to the NRC. This report provided an industry response to the information requested in Item 1.a of the bulletin.

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-1 Response

Arkansas Nuclear One, Unit 1 (ANO-1) is a B&W design, which has 69 reactor pressure vessel head penetrations containing 68 control rod drive mechanisms (CRDMs) and one vessel reactor vessel level instrument which is the center nozzle. The ANO-1 head does not contain thermocouple nozzles. The requested nozzle information is provided in Table 2-3 of MRP-48. In addition, the CRDM nozzles at B&W plants were fabricated from the same product form (hot finished seamless tubing) and from only 13 individual heats of Alloy 600. The materials were ordered to ASME Code, Section II, Specification SB-167 and relevant Section III requirements. Additional information on the B&W designed nozzles can be found in topical report BAW-2301, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, which was submitted to the NRC on July 25, 1997 in response to Generic Letter 97-01.

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

ANO-1 Response

ANO-1 is typical of other B&W facilities, which have metal reflective insulation that is located on a horizontal plane above the head. The lowest clearance for inspection of the nozzles is at the top of the head, which has an approximate 2-inch space between the upper most nozzles and the insulation (see Figure 1).

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-1 Response

ANO established a CRDM nozzle inspection program based on the guidance of Generic Letters 88-05 and 97-01. The ANO program has been developed and enhanced based on lessons learned from ongoing inspections and improved techniques to observe boric acid pressure boundary leaks at the nozzle-to-head interface on the exterior. The Reactor Vessel Head Service Structure support contains more than 20 openings, which allow inspection in the base of the skirt around the vessel head. Additionally, a larger inspection opening was added in the latest refueling outage (1R16) to support repair of CRDM Nozzle 56. Previous inspections have utilized a videoscope (boroscope) that is conducted by knowledgeable personnel within the ANO Systems Engineering organization. The boroscope has low light detection capability and includes its own light source to enhance visual clarity. The ANO-1 RV head was cleaned at 1R14 to establish a baseline for subsequent inspections. Video recordings of the baseline inspection and subsequent inspections for ANO-1 have been retained. Following the CRDM Nozzle 56 repair in 1R16, the head was cleaned again and a new baseline established. Video recordings of the baseline inspections and all subsequent inspections for ANO-1 have been retained.

A remotely operated (robotic) camera was developed prior to 1R16 for ANO, which allowed better access to the head with reduced dose to the inspection team. The video quality of the robotic camera was improved over that of the videoscope. The robotic camera also has its own light source, which can easily illuminate boric acid deposits around these nozzles similar to the types of boric acid deposits experienced during 1R16. The robotic camera has only minor limitations for the inspection of the upper nine nozzles on the top center of the head due to reduced access from insulation (see Figure 1) and from insulation supports. However, if necessary, additional boroscopic exam of this area can also be performed. This inspection meets the intent of a qualified visual inspection per NRC Bulletin 2001-01 (see the response under item 2.d). Either the robotic camera or the boroscope is considered acceptable for detecting boric acid deposits on the head. A copy of a VHS formatted presentation that shows the robotic inspection capability was provided to the NRC in Entergy letter dated August 23, 2001 (1CAN080103).

Immediately after shutdown for a refueling outage, ANO performs a complete inspection of the CRDM nozzles to determine whether boric acid is present that would indicate nozzle leakage. Nozzle leakage must be distinguished from CRDM flange leakage, CRDM stator cooling water leakage, or other activities on the reactor vessel head that can create boric acid deposits. Boric acid deposits can occur during normal power operation, shutdown conditions, or plant refueling. Based on experience from both Oconee Units 1 and 3 as well as ANO-1, CRDM nozzle leakage can now be better distinguished from other types of leakage.

During 1R15 a small “kernel” of boron was observed at the annulus of CRDM nozzle 56. A careful comparison of the 1R14 and 1R15 videos was performed of the areas around the nozzle of CRDM 56. It was concluded that the boron kernel found in that outage appeared to be boron residue that had fallen into the annulus of CRDM nozzle 56 from above the nozzle. If the boron in the annulus resulted from a leak, it was expected that the boron crystals would have been continuous around the circumference of the annulus. In addition, bare metal was found between some of the residue of boron in the annulus. Therefore, it was concluded that the boron was only partially removed by cleaning in 1R14 and that the unique kernel of boron on CRDM 56 had likely dripped down from the insulation above.

In 1R16, with knowledge of the Oconee Nuclear Station (ONS-1) experience, and since this nozzle had exhibited a small kernel of boron in the annulus on top of the head during 1R15, CRDM nozzle 56 was closely re-examined in the 1R16 inspection. An indication of boric acid leakage was found on top of the reactor vessel head around CRDM nozzle 56. This increase in boron was found with the robotic camera in a first-look inspection. After this initial inspection and with the head still on the reactor vessel, the visual examination was completed on all 69 CRDM nozzles using remote video equipment. No other leakage was observed from the 1R16 inspection. Even though there was boron present during the 1R15 inspection, it is inconclusive that nozzle 56 was leaking during the previous cycle. However, even if the nozzle/weld had experienced throughwall cracking, the flaw had not progressed beyond a limited axial flaw. See the discussion in the response to 2.a regarding the flaw characterization.

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-1 Response

General Description: The CRDMs are located above the reactor vessel head and are attached to the head at the CRDM nozzles (see Figure 1). They are housed within the service structure attached to the top of the reactor head. At the top of the service structure is a structural grid that provides support for the stabilizer plates and the tops of the CRDM housings. The stabilizer plates also provide access to the CRDM cable connections and support for the CRDM cables.

Additional components that are located above the reactor vessel head and below the missile shield within the refueling canal include cooling ducts for the refueling canal area and the service structure, CRDM and instrumentation cabling, cooling water piping for the CRDM thermal barriers, miscellaneous electrical power cables, and communication cables. Other structures in this area are the cable support platforms.

Missile Shield: The missile shield above the reactor vessel is made up of four reinforced concrete blocks, each having dimensions of 7’-0” wide, 2’-6” thick, and 31’-0” long. The concrete blocks span across the refueling canal, are supported on the secondary shield walls, and are centered over the reactor vessel. Each shield block is anchored at each end by two 1½” diameter bolts during plant operation. Bottom elevation of the missile shield blocks is 426’-6”. The missile shield blocks are removed by the Polar Crane for refueling activities and are stored inside the Reactor Building.

CRDM Housings and Their Support/Restraint System: The CRDM housings are supported off the tops of the nozzles coming out of the reactor vessel head by a bolted flange and by the stabilizer plate at the top of the service structure. The service structure is a ¾” thick steel shell which is mounted on the service structure support and mounting flange that is attached to the reactor vessel head. The stabilizer plates are bolted to steel channel structures supported on the top of the service structure. The stabilizer plates support personnel access during outages, the top of the CRDM housings, and the cables for the CRDMs and instrumentation.

Other Components: The intermediate cooling water (ICW) piping is supported off the south secondary shield wall. The supply and return lines are 3” in diameter and bottom of pipe (BOP) is at elevation 406’-6” and 405’-6” before they turn downward and connect to the flanges on the service structure. The pipes continue down to the thermal barriers just above the reactor vessel head.

The reactor vessel head high point vent comes off of the top of the reactor axial power shaping rod CRDM housing at elevation 401’-11 ¾”. From that location the piping turns horizontal at elevation 403’-0 3/16” to a valve, an orifice, and splits into two horizontal 1” diameter pipes. All of the supports for the vent system are attached to the top of the service structure adjacent to the ICW piping attachments to the structure.

Other components in the area are the communication and lighting conduits and fixtures along the south refueling canal wall. They generally are located at elevation 418’-6” and provide lighting and communication for refueling operations.

Other Structures: The only other structures in the area inside the refueling canal and above the reactor vessel are the cable tray structures, the CRDM cooling ducts, and the refueling canal area cooling ducts.

The cable support structures (Bat Wings) are two triangular shaped steel platforms that support the cables as they extend from the connections at the cable chase to the tops of the CRDM housings. These platforms are attached to the CRDM service structure and off the north wall of the refueling canal wall by resting on the refueling machine rails. There is one platform on the east and west sides of the support structure top platform. They are made of steel channel and beam shapes with decking for a walking surface.

There are two round cooling ducts that run along the south side of the refueling canal wall above elevation 412’. One of these ducts splits into two ducts that (along with the other duct), turns downward to elevation 391’ to ring ducts which supply cooling air to the inside of the service structure at the CRDM flange area.

On the north side of the refueling canal is a rectangular cooling duct at approximately elevation 414’ that supplies cool air into the canal area through grills along its length.

Other Cabling: The major cabling in the area of the refueling canal up to the missile shield consists of the CRDM power and control cables, and other instrument cables for the area around the reactor vessel head. The cables extend from the cable trays on the north side of the refueling canal at elevation 401’-6”. All of these cables are supported by the service structure as they proceed to the reactor vessel head area.

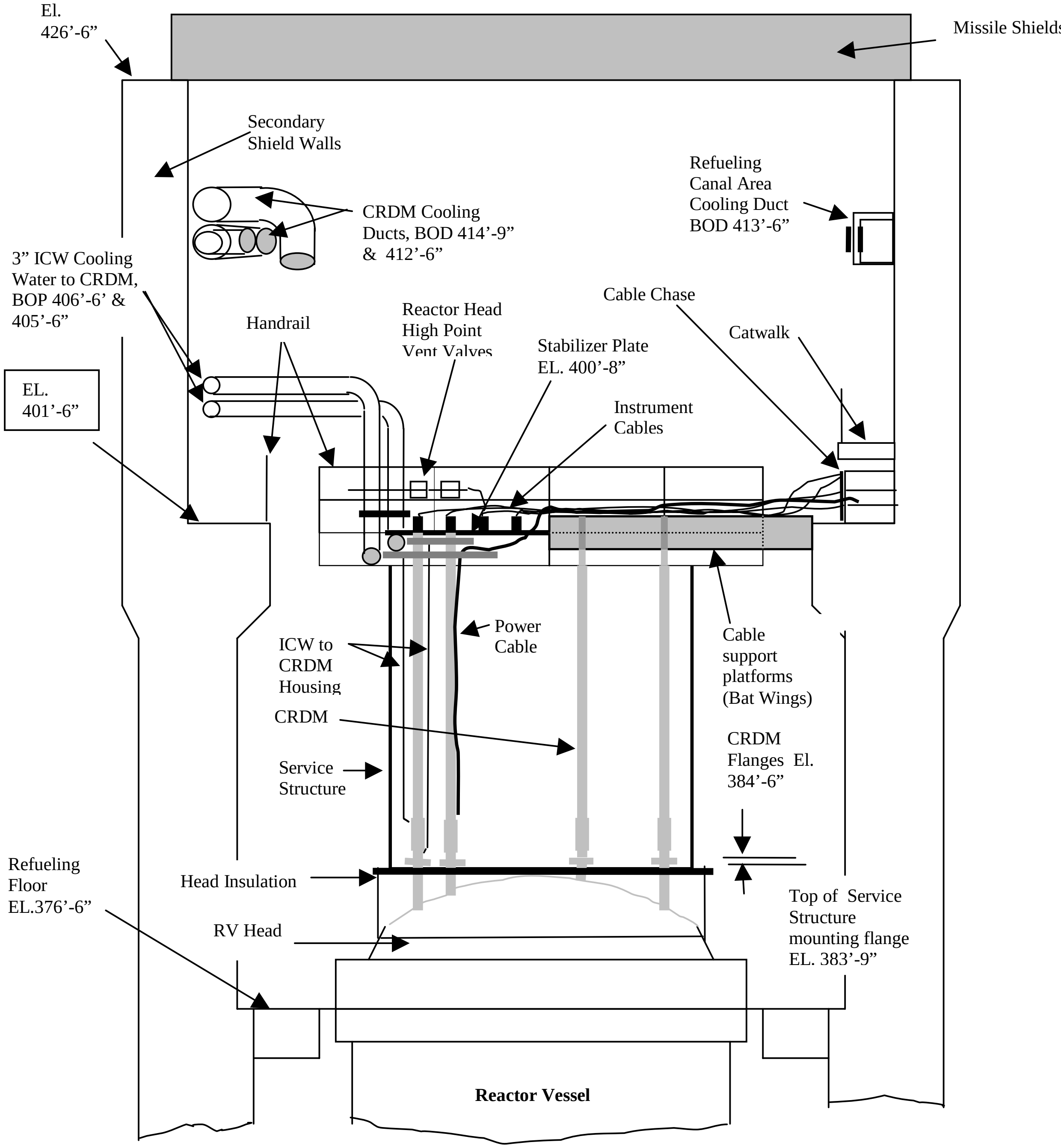

FIGURE 1

General View of ANO-1 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:***

2.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;

ANO-1 Response

During the 1R16 refueling outage on March 23, 2001 and again on March 24, 2001, penetrant examinations (PT) were performed on the Alloy 600 J-groove weld-to-nozzle 56 from beneath the head. These PT examinations tested three areas: (1) the outside of the nozzle extension under the head, (2) the J-groove weld, and (3) the inside bore of the nozzle for approximately 8" up from the nozzle end. These examinations were performed after light surface preparation of the weld area by an orbital sander and scotch-brite scrubbing of the bore with solvent.

The PT examinations found a rejectable crack on the outer diameter (OD) of the nozzle beneath the weld (below the pressure boundary) on the downhill side. Figure 2 shows the characterization of the flaw identified in 1R16. This crack contained a circumferential segment at a location 0.4" from the weld fusion line, and it then curved into the axial direction. The total circumferential extent of this crack was approximately 0.7" and extended from the 340° to 0° (bottom dead center of lower hillside). The crack branched twice in a "Y" shape, with all three crack segments extending toward the weld fusion line. These crack segments did not propagate into the weld. The PT examinations did not detect any indications on the inside surface of the nozzle.

On March 24, 2001 Framatome ANP performed an automated ultrasonic examination using a "top-down" probe delivery system through the nozzle bore. The top-down tool was positioned with the "Y" axis (axial) zeroed at the top of the flange with the positive direction extending down the nozzle. The ultrasonic examination (UT) confirmed a crack in the location of the linear PT indication. The UT data indicated the sub-surface dimensions of the crack extended in a circumferential direction below the weld from approximately 339° to 0° to 30° and in an axial direction through the weld at the fusion zone to a termination point approximately 1.3" above the weld. The maximum flaw depth dimension was estimated to be 0.2" into the nozzle wall. Therefore, the UT inspection confirmed that there was a leak path at CRDM 56 that extended from an OD crack that propagated partially throughwall past the weld to the nozzle annulus.

Also, on March 24, 2001 Framatome personnel performed an automated eddy current (ET) examination of the nozzle bore. The ET examination indicated two clusters of crazing (shallow cracking) and a scratch near the uphill side of the nozzle bore at 180°. The estimated depth of the crazing was less than 2 mm (0.079"). These indications were not detected by the PT inspection, indicating that they were very shallow. This was confirmed by the UT examination.

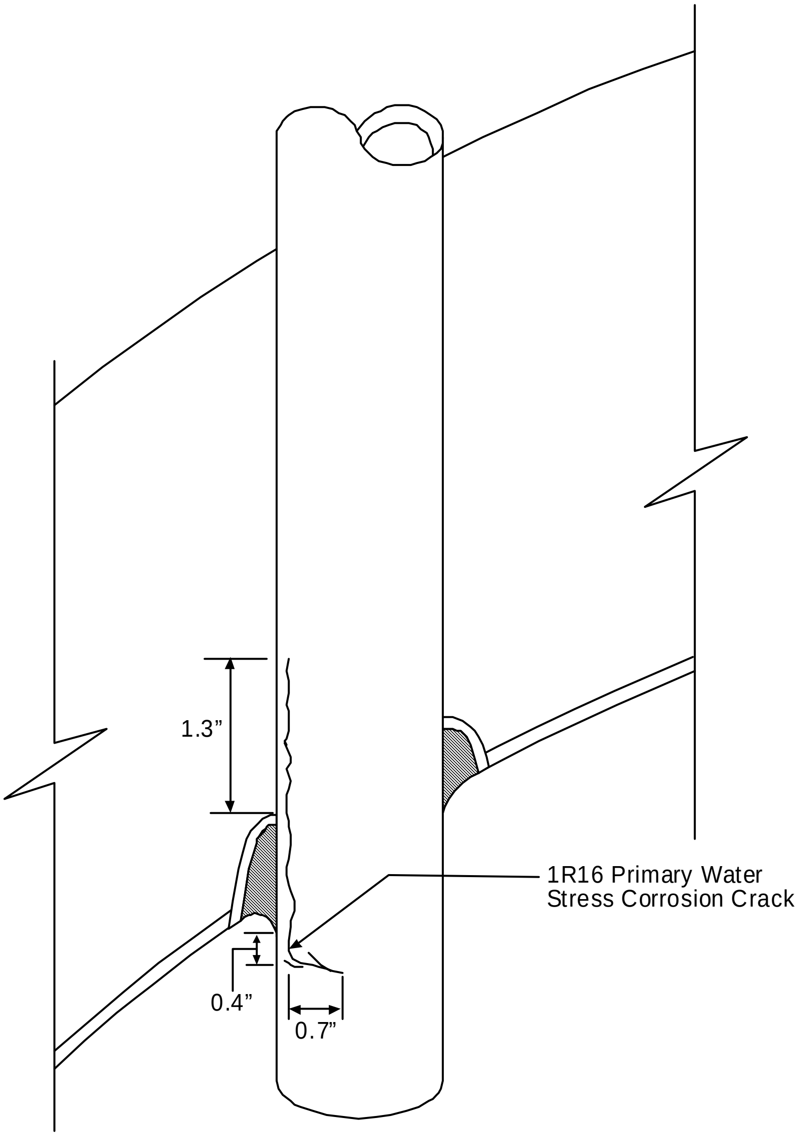

Figure 2

Characterization of ANO-1 CRDM Flaw Identified in 1R16

2.b a description of the additional or supplemental inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken in response to identified cracking to satisfy applicable regulatory requirements;

ANO-1 Response

Since the flaw identified during the 1R16 outage was limited to the interface between the nozzle and the J-groove weld, Entergy chose to perform an embedded flaw weld repair. Initially the circumferential portion of the flaw was removed by severing the nozzle just above its circumferential extent. The axial portion of the flaw was removed in both the existing J-groove weld and on the OD of the nozzle (below the butter interface) by grinding. The final ground finish at the butter interface did not reveal any radial crack in the weld other than a non-reportable spot at the nozzle to J-weld interface. The only remaining portion of the flaw was above the weld at or near the outside diameter of the penetration nozzle. The excavated cavity wall was then built back with an Alloy 690 compatible weld material. Progressive PT examinations were performed during welding per ASME Code requirements. The remaining flaw was isolated from the RCS.

After completing the weld repair of the J-weld, Framatome ANP performed an automated UT and ET examination of the CRDM nozzle using the “top-down” inspection tool as was performed for the initial UT and ET examinations. Results of these examinations confirmed that the remaining embedded flaw was unaffected by further welding activities.

Entergy evaluated the need to perform additional inspections of other CRDM nozzles during the 1R16 outage. It was concluded that additional head penetration examinations during the 1R16 outage were not necessary for the following reasons:

- Bounding fracture mechanics and flaw growth evaluations concluded that adequate safety margin exists to ensure that no adverse structural concern would exist between cycles assuming significant initial flaws.

- The as-found indications of the ANO-1 CRDM nozzle and the reported indications at ONS-1 and ONS-3 if applied to an ANO-1 CRDM nozzle are bounded by the fracture mechanics analysis discussed above.

- Safe operation of the B&W-design plants would not be affected given the visual inspection criteria currently in place that will detect leakage prior to any potential degradation of the structural integrity of the CRDM nozzles. Thus, the potential for cracking of CRDM nozzles did not present a near-term safety concern.

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

ANO-1 Response

Inspection Plan

ANO-1 will perform a *qualified visual* examination of the upper surface of the reactor vessel head during 1R17 (the next refueling outage scheduled for the fall of 2002), and contingency plans and preparations will be made for volumetric examinations, if necessary.

Based on the information provided in this response and other actions being taken by the MRP, Entergy believes that the safety significance of primary water stress corrosion cracking (PWSCC) of CRDM nozzles at ANO-1 can be adequately managed by *qualified visual* examinations and associated repairs. However, Entergy also believes that for long term management of this condition more proactive actions are necessary to prevent the potential recurrence of leakage caused by PWSCC. Therefore, in addition to the plans described in this response, an effort is in progress to develop a mitigation technique that would apply a weld overlay of corrosion resistant material to the wetted surface of the CRDM nozzle and J-groove weld using remote automated tooling. This technique, once developed, could be applied as a repair or preventative action for cracking of CRDM penetrations. Although not a commitment, the goal of the mitigation development effort is to begin using the technique in the fall 2002 refueling outage.

Type and Scope of Inspections

A *qualified visual* examination will be performed on essentially 100% of the outer bare metal surface of the control rod drive penetrations for evidence of leakage. If evidence of leakage is found, additional examinations of the penetration will be performed to characterize the nature and extent of cracking and disposition as required by IWA-5250 of the ASME Section XI Code. Decisions on additional inspections beyond those identified as leaking will be based on the nature of the observed cracking, the extent and severity of cracking, the dose rates, the availability of non-destructive examination equipment including trained and qualified workforce.

2.d. your basis for concluding that the inspections identified in 2.c 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 performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

ANO-1 Response

Timeliness of Corrective Actions

In 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, the licensee is first expected to ensure the public health and safety and second to restore the systems, structures and components (SSCs) to the current licensing basis of the plant with an acceptable level of safety. It further clarifies that when a degraded or nonconforming condition of an SCC subject to Appendix B to 10 CFR Part 50 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 this criteria, ANO-1, upon discovery of the leak during 1R16, initiated a condition report promptly identifying the finding. Under the auspices of this corrective action document, the leaking nozzle was repaired, the safety significance determined, immediate corrective actions identified and taken, effects on operability concluded, and root cause determination completed. In all of these actions, Entergy considered current information from other licensees who were experiencing similar conditions. At the conclusion of the root cause determination, there were no discoveries to indicate that the actions taken during 1R16 were inadequate to ensure the continuance of public health and safety for the next cycle of operation.

For the subpopulation of plants that have previously experienced either leakage or cracking in VHP nozzles, the NRC Staff's expectation is that the suggested examination would be completed before December 31, 2001. This schedule cannot be accomplished within ANO-1's refueling schedule and would require an unplanned shutdown. The degree of ASME Code permissible cracking is sufficient to use *qualified visual* examinations on an outage frequency basis with actions taken upon discovery of leakage. Based on the safety significance of this issue (flaw tolerance of the nozzles, and the ability to detect leakage before structural integrity is compromised), and by what is necessary to implement this portion of the corrective action, this appears to be inconsistent with the guidance of Generic Letter 91-18. As presented later in this response, Entergy believes that the robustness of the CRDM nozzles is adequate to sustain substantial cracking before challenging the nozzles structural integrity.

Additionally, the visual inspections performed in 1R16 were recently concluded in the spring of this year. There has been less than six months of plant operation and it is reasonably expected that additional inspections would not identify any new evidence of leakage.

During 1R14 and again in 1R16, the reactor head and CRDM connections were cleaned, visually inspected and a video recorded baseline established. Inspection of the reactor head and CRDM connections has been repeated and video taped during 1R15 and 1R16. The visual inspection performed during 1R16 discovered the leak at CRDM 56 indicating that the conditions of the head are such that leakage, when present, can be identified and accurately characterized. As discussed later, analysis of as-built information also demonstrates that, during operation, sufficient clearances between the CRDM nozzles and the RV closure head exist to permit accumulation of boron that would indicate throughwall leakage on the head surface. Therefore, the following observations are made:

1. There is reasonable assurance that throughwall leakage in CRDM nozzles at ANO-1 will manifest into leakage that can be visually detected on top of the RV closure head as discussed under 2.d (2).

2. Visual examinations of sufficient quality to detect leakage have been performed for the last three outages with only one leaking CRDM identified. Based on the recent leakage findings at ANO and at Oconee, the evaluation sensitivity for minor leakage has improved.

3. The leaking CRDM nozzle did not contain any circumferential cracking above the weld.

4. Because leakage is a precursor to the initiation of OD circumferential cracking located at or above the J-grove weld, there is reasonable assurance that cracks of this nature did not exist when the unit was returned to operation from 1R16.

5. Based on the fracture mechanics information discussed later, there is reasonable assurance that if a leak were to initiate on the day the plant returned to service from 1R16, and a circumferential crack immediately initiated coincident with the leak, that significant margin to limit load conditions exist for more than one operating cycle (18 months).

Combined with the inspection history, the known ability to detect leaks, and the understanding of how PWSCC flaws affect the CRDM nozzle’s structural integrity, there is adequate evidence to conclude that the overall condition of the ANO-1 RV head does not represent a condition of safety significance that warrants an unplanned shutdown of the unit.

Additionally, the mobilization of vendor resources to support volumetric examination would currently challenge the abilities of the NDE industry. The MRP has reported to the NRC (EPRI Interim Report TP-1001491, Part 2 (MRP-44), that the current capabilities of the three vendors providing acceptable NDE services, at best, would only support 3 to 5 units for the fall 2001 outages. The MRP has collected information that indicates that there are approximately 10 plants in the 3 upper susceptibility populations scheduled for refueling outages in the fall of 2001. While all of these plants are not performing volumetric examinations, most will be making some level of contingency plans in the event repairs to their vessel head are required. These contingencies typically include the ability to perform volumetric NDE of the repaired area. The tooling, NDE equipment and NDE personnel to perform examination of a repaired CRDM nozzles are the same that would be used for an overall head examination. As recognized by the NRC, these inspections can result in large personnel exposures and require careful planning and tooling optimization to keep the personnel exposure as low as reasonably achievable (ALARA). Entergy believes that the NDE technology is advancing to improve both the quality of the NDE itself as well as the delivery systems that will reduce the associated personnel exposure.

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

Therefore, Entergy concludes that for the reasons stated above, as supported by other information contained in this response, that the inspections scheduled for 1R17 meet the guidance of Generic Letter 91-18 for timeliness of corrective actions.

2.d.(2) If your future inspection plans do not include volumetric examination of all VHP nozzles, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will be satisfied.

ANO-1 Response

As described in 2.c, the ANO-1 planned inspections are scheduled to occur beginning in 1R17 which will consist of a *qualified visual* examination. Like the inspections performed during 1R15 and 1R16, these visual inspections are believed to be adequate to address the regulatory requirements because they provide appropriate measures to ensure structural integrity of the CRDM housing and are provided in a timely manner consistent with the guidance of Generic Letter 91-18.

Because ANO-1 plans on performing a 100% visual examination in lieu of volumetric, the Staff's guidance for a *qualified visual* examination is being applied. The Staff has indicated that the *qualified visual* examination should be able to reliably detect and accurately characterize leakage from cracking in VHP nozzles considering two characteristics. One characteristic is a plant-specific demonstration that any VHP nozzle exhibiting *throughwall* cracking will provide sufficient leakage to the RPV head surface (based on as-built configuration of the VHPs). Secondly, similar to the *effective visual* examination for moderate susceptibility plants, the effectiveness of the examination should not be compromised by the presence of insulation, existing deposits on the RPV head, or other factors that could interfere with the detection of leakage.

Characteristic 1, Ability to Manifest a Detectable Leak: Even though the conditions reported for CRDM 56 provide evidence that throughwall leakage can be detected by top head visual inspections, Entergy, with the assistance of Structural Integrity Inc., has performed substantial analysis of the as-built dimensional fits for the CRDM nozzles in the ANO-1 head.

The analysis consisted of a finite element model to include the upper hemispherical head, the upper closure flange and the CRDM housing nozzles. Due to the symmetrical nature of the upper head and the layout of the CRDM nozzles, only 45° of the total circumference was modeled. The finite element model for the gap analysis is being used to support leak determinations and fracture mechanics evaluations. For the gap analysis, the worst (or largest) interference values were modeled to minimize the gap opening and thus the leak rate. In addition, with only 13 of the 69 nozzles modeled, the worst interference load for the corresponding nozzle sets was used. Worst case for this analysis was the nozzle that had the greatest top or bottom interference dimension. For this evaluation, where the greatest interference fits were used, it was determined that all of the nozzles include a vertical path that will allow leakage.

Characteristic 2, Ability to Visually Identify a Leak: The visual inspection to be performed during 1R17 will be performed by personnel from multiple site disciplines including those who performed the inspections at 1R16. The inspection team will include personnel qualified to the requirements of ASME Section XI for performing VT-2 examinations. Personnel on the team will be knowledgeable in the detection and discrimination of leakage evidenced by the accumulation of boron deposits.

The knowledge gained from the flaws identified at both ANO-1 and Oconee has enhanced the knowledge of our inspection team for boric acid deposit characterization. The inspection techniques and tooling will be consistent with those that have been proven successful in the detection of boron accumulation indicative of leakage. As stated earlier, the ANO-1 head was cleaned in 1R14 and again at 1R16 where baseline conditions were recorded on videotape for comparison at subsequent inspections. This process has continued through 1R15 and 1R16 with the results of each inspection also being recorded.

Because of the ease of inspection of the ANO-1 head and the demonstration of available leakage paths between the CRDM nozzle and the RV closure head, Entergy believes that there is reasonable assurance that initial leakage that would initiate latent circumferential cracking of the CRDM nozzle at or above the J-grove weld will be detected by visual inspection.

Analysis to Support Qualified Visual Examination

Cracking of the CRDM from PWSCC can manifest itself in differing configurations and locations having varying affects on both the leak tightness and structural integrity of a CRDM nozzle. Cracks located in the portion of a CRDM nozzle that extends below the weld may lead to crack propagation into the weld or further up the nozzle wall. However, these cracks in themselves, do not affect the nozzle's leak tightness or structural integrity.

Cracks in the nozzle wall or in the nozzle wall to weld interface that are oriented in an axial direction may obtain sufficient length to permit leakage into the annulus between the CRDM nozzle and the RV head above the weld. Although this crack condition does permit leakage of reactor coolant, it does not challenge the integrity of the CRDM nozzle. However, throughwall leakage provides an environment that may lead to circumferential cracking from SCC at or above the J-grove weld.

Other circumferential cracking, either ID initiated or propagated from an OD axial crack within the weld fusion zone, is currently believed to be very unlikely due to the low axial stress on the nozzle ID and the compressive stress on the nozzle's OD in the weld fusion area. To verify this, the MRP is evaluating approximately 300 cases of differing flaw orientations, configurations and locations to clearly identify the limiting conditions that should be considered. But in the interim, as supported with limited analysis and the concurrent industry inspection results, it is reasonable to focus on circumferential cracks that initiate at or above the weld on the OD of the CRDM nozzle. For these reasons, Entergy's analytical efforts as described below are focused on the cracking considered to be most significant, which is the OD initiated circumferential crack. Conservatively, the flaw growth assessment takes no credit for initiation or incubation time following a throughwall leak.

Entergy has aggressively pursued further understanding of the degradation condition and its safety significance. During 1R16 and subsequent to 1R16, analyses and fracture mechanics evaluations of the ANO-1 CRDMs have been performed to quantify the safety significance of PWSCC on CRDM nozzles. As a result of these analyses and evaluations, as described below, it is demonstrated that the inspection plans provided under item 2.c are adequate with regard to safety significance and for compliance with the cited regulatory requirements. The detailed analysis is considered proprietary, but can be made available to the NRC upon request.

The results of the evaluation described for the fracture mechanics are based on the analysis performed to date of the nozzle at the 38.5-degree location (location of the nozzle leak at ANO-1 in March 2001). Even though not part of this bulletin response, additional analyses are in progress for two nozzle locations namely, at the center and at an intermediate location.

<u>Finite Element Analysis of Nozzle-to-Vessel Head Interface:</u> A detailed finite element model of one quarter of the vessel head including the CRDM nozzles was developed using ANO-1 as-built drawings and verified dimensions. The purpose of this analysis was to ascertain whether a physical gap existed between the nozzle outside diameter and the bore under normal operating conditions. The detailed evaluation showed that under normal operating conditions a gap did exist between the nozzle outside diameter and the bore diameter. The existence of the gap ensures that leakage can be detected by visual means on the vessel head.

Finite Element Fracture Mechanics: The region for potential OD initiated circumferential cracking is the weld root between the nozzle and the bore. Given the geometry of this intersection the potential crack plane is oblique with respect to the nozzle axis. In addition, the residual stress distribution at this location is complex and cannot be described by simple closed form equations. Because of the combination of the geometric and stress distribution complexities, finite element analysis methods were employed for both the determination of the residual stress distribution and for the evaluation of stress intensity factors as a function of various *throughwall* circumferential flaw lengths. The stress intensity factors, determined by this effort, were used to determine the flaw growth around the circumference by stress corrosion cracking. The finite element analysis performed to date, using the detailed stress distributions and the appropriate gaps for ANO-1, shows the stress intensity factor (for a throughwall flaw) to gradually diminish as the circumferential extent (flaw length) increases. The reduction in the stress intensity factor, around the circumference, reduces the bending experienced by the flaw tip. This occurs when the initial gap between the nozzle outside diameter and the bore wall is closed.

Crack Growth Rate Assessment: As noted in the bulletin, the possible existence of a more aggressive environment in the CRDM housing annulus following throughwall leakage is an issue that must be addressed. The issue is that potentially highly concentrated borated primary water could become oxygenated in the annulus and cause increased likelihood for the initiation of cracking and higher crack growth rates. Because of the uncertainties associated with this issue, a bounding correlation was developed. The available crack growth rate data for both PWSCC and for stress corrosion cracking under oxygenated conditions were evaluated. The crack growth rate behavior as a function of the stress intensity factor for crack growth in the annulus between the nozzle and the bore was developed by utilizing the Boiling Water Vessel Internals Project Alloy 600 data for normal water chemistry and by adjusting the correlation to account for the temperature difference. The correlation was developed for a temperature of 550 °F and was adjusted for a temperature of 602 °F for the ANO-1 condition. This adjustment leads to an enhancement by a factor of approximately 4.2. Reactor head effective temperatures can vary from the mixed mean reactor outlet temperature based on the head flow characteristics. The impact of these effects may require small adjustments of the head temperature. However, it is not expected that these adjustments would significantly alter the conclusions of the flaw growth assessment presented here.

Flaw Growth Assessment: A partial throughwall/partial circumferential flaw is assumed to exists simultaneously with the occurrence of a breach, at the location of interest, exposing the outside diameter of the nozzle to the reactor coolant. The existence of a partial throughwall/ partial circumferential flaw permits flaw growth analysis to commence at the instant of the breach and is conservative because no incubation period is considered. The size of the assumed partial throughwall/partial circumferential flaw was chosen to adequately accommodate multiple initiation sites. The flaw growth through the thickness was modeled using the closed form empirical solution provided in ASME B&PV Code, Section XI, Appendix "A". The finite element based stress intensity factors for partial *throughwall*/partial circumferential flaws are to be developed for the MRP project on probabilistic fracture mechanics. The empirical solution utilized in this calculation yields a more conservative stress intensity factor because: a) the stresses used in the calculation were peak axial stresses (from the finite element analysis) and were assumed to be constant around the circumference; and, b) the flaw propagation plane is assumed to be the radial-axial plane and not the oblique plane along the weld root. Once the flaw had penetrated through the wall, the finite element based stress intensity factors were used to propagate the flaw around the circumference. The allowable flaw size was computed using limit load methods with a factor of safety of three (SF= 3.0). The flaw growth rate, which reflects the expected growth rate in an oxygenated environment at reactor water temperatures, ensures that proper conservatism is maintained. Results of this analysis showed that the initial partial throughwall/partial circumferential flaw took approximately one and one quarter (1.25) operating cycles to penetrate the wall thickness. The time for subsequent growth around the circumference, to an allowable flaw size, takes approximately two and a quarter (2.25) operating cycles. Thus from the time of breach, with the conservative assumptions described above, it would take more than three operating cycles for the assumed initial flaw to reach the allowable flaw size. Therefore, there would be sufficient opportunity to detect the leak and effect proper repairs.

Summary of Analysis Performed: The detailed analysis performed clearly demonstrates that:

- The existence of a gap under normal operating conditions would facilitate discovery of a leak by appropriate visual inspection;
- It will take over three operating cycles to propagate the conservative hypothetical flaw to reach the ASME structurally significant flaw length;
- Sufficient margins exists to ensure safety at the same time provide the opportunity for timely discovery and for performing effective repairs.

Other Information to Support the *Qualified Visual* Examination Method

1. Alloy 600 CRDM 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 the 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. Fracture mechanics 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.

2. 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 CRDM penetrations. This evaluation, which would include the ANO-1 condition, is scheduled to be complete by the end of 2001. The evaluation approach will use the failure probability for SCC using the PRAISE and SARA computer codes.

3. The assumption that an initiating event frequency of 1 for a rupture of a CRDM penetration made by the NRC is extremely conservative. Historically, complete pipe breaks have been estimated from historical data at a frequency of about $1X10^{-5}$ /reactor

year. CRDM 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 have 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 CRDM penetration ejection is estimated to be in the 10^{-5} - 10^{-6} /reactor year range which is consistent with the NRC safety goal.

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

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-1

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:

Not Applicable to ANO-1

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-1 Response:

Entergy will provide the requested information for ANO-1 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 fall 2002.

ANO-1 Perspective for Compliance to Regulatory Requirements Cited in NRC Bulletin 2001-01

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

The three referenced design criteria state the following:

- 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."

- Criterion 32 – 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 met 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) 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 CRDM reactor coolant pressure boundary materials have 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.

TECHNICAL SPECIFICATION REQUIREMENTS

The Bulletin states:

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

Title 10 of the Code of Federal Regulations, Part 50.36 (10CFR 50.36) 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 pressure 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-1 Technical Specifications include a requirement and associated action statements addressing reactor coolant pressure boundary leakage. The limits for reactor coolant pressure boundary 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 leakage from a non-isolable fault in the reactor coolant system pressure boundary.

Technical Specifications provide requirements for action when leakage is found, and operability requirements for the leakage monitoring systems. Portions of the applicable ANO-1 Technical Specifications are as follows:

3.1.6.3.a *If it is determined that any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system strength boundary (such as the reactor vessel, piping, valve body, etc., except steam generator nozzles), the reactor shall be shutdown….*

3.1.6.5 *Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.*

3.1.6.7 *When the reactor is at power operation, three reactor coolant leak detection systems of different operating principles shall be in operation. Etc..*

The bases of TS 3.1, *Leakage*, help provide the intent of the TSs.

Every reasonable effort will be made to reduce reactor coolant leakage, including evaporative losses (which may be on the order of 0.5 gpm), to prevent a large leak from masking the presence of a smaller leak. Reactor building sump level, water inventory balances, radiation monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not, can be a serious problem with respect to in-plant radioactive contamination and cleanup or it could develop into a still more serious problem; and therefore, the first indication of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of 1 gpm may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage, must be considered in the safety evaluation.

Leakage monitoring during power operations and system reactor coolant pressure boundary walkdowns during outages serve to ensure that unidentified leakage from the RCS, which would indicate a potential safety concern, will be readily detected. Regarding inspections which determine leakage that cannot be determined through online monitoring, the current reactor vessel bare head inspections look for signs of boric acid deposits which would indicate throughwall boundary leakage from the CRDM nozzles. The B&WOG utilities, including ANO-1, have included plans to visually inspect the CRDM nozzle area to determine if leakage is observed on top of the RV head, which would indicate *throughwall* cracking has occurred, during their outages. In addition, walk-down inspections have been implemented in response to NRC Generic Letter 88-05 at each of the B&WOG plants. The walk-down inspections include an enhanced visual inspection of the gasket area and RV head during every refueling outage. The B&W closure head and service structure design provides access for a visual or boroscopic examination of the CRDM nozzle area, since the insulation is not resting on the RV head. If any leaks or boric acid crystal deposits are noted during inspection of the RV head area, an evaluation of the source of the leak and the extent of any wastage is performed. This program has shown to be effective, as evidenced at ONS and ANO-1. For the flaw identified during 1R16, it consisted of a minor axial indication that was not structurally significant. The boric acid deposits clearly indicated the presence of a throughwall flaw, which required repair. These visual examinations provide an acceptable level of quality and safety and are in accordance with 10CFR50.55a and General Design Criteria 30 of Appendix A to 10 CFR50.

During on-line operations, reactor coolant pressure boundary integrity can be continuously monitored in the control room by observation of the variation from normal conditions for Reactor Building Sump Level, and Reactor Building Radiation and Gaseous Activity Levels. In addition, the RCS inventory balance provides a highly sensitive means of measuring inventory reduction. 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 detectable with the visual examinations associated with ASME Code required examinations or pressure tests. ANO-1 has evaluated this condition and has determined that the appropriate inspections are enhanced bare metal visual inspections of the reactor head for boric acid deposits near the CRDM penetrations during plant shutdowns. Field experience and analysis have demonstrated that most PWSCC in Alloy 600 CRDM penetrations and its weldments are axial in nature as predicted by analysis and confirmed by observation by NDE and destructive metallurgical analyses. In those cases where the cracking has progressed to the OD of the nozzle and propagated as driven by the highest stresses present, the cracking could challenge the nozzle 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 nozzle integrity. Further, probabilistic fracture mechanics evaluations of the CRDM 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 *throughwall* pressure boundary leaks of CRDMs increase to the point where they are detected 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-1 has met, and will continue to meet, technical specification requirements for reactor coolant pressure boundary leakage. If leakage from the reactor coolant pressure boundary is detected from a non-isolable fault or if it exceeds leakage limitations during plant operation, appropriate action statements will be followed. If leakage or unacceptable indications are found from either required ASME Code visual examinations or supplemental examinations prescribed by Entergy, any throughwall leakage of the reactor coolant pressure boundary or any defect found to be unsuitable for continued service by analysis 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. Further analyses would be performed for determining 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 throughwall leakage that penetrates the pressure retaining membrane." Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit throughwall cracking of VHP nozzles.*
>
> *For throughwall 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-1 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 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 CRDMs) 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.

In addition to the inspection requirements of ASME Section XI, ANO-1 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*. In addition, enhanced bare metal visual examinations have been conducted at ANO-1.

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[1] An error 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.

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 throughwall leaks to be corrected by either repair or replacement. Upon discovery of the leaking nozzle, ANO-1 implemented welded repairs in accordance with ASME Section XI prior to returning the unit to service from 1R16. 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 ANO's corrective action program. Further, any defects found from any examination of the CRDM nozzles 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.

10 CFR 50, APPENDIX B

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 throughwall 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 constitutes 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 are 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 addition, NDE inspection techniques only determine the current condition of the nozzle and will not prevent PWSCC from occurring. Therefore, the performance of volumetric examinations will not prevent future occurrences of PWSCC cracking without mitigative actions.

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

Commitments Contained in this Letter

Commitment	TYPE		Scheduled Completion Date (If Required)
	One Time Action	Continuing Compliance	
Entergy will perform a *qualified visual* examination of essentially 100% of the upper surface of the reactor vessel head during 1R17 and contingency plans and preparations will be made for volumetric examinations if necessary.	X		Fall 2002
The visual inspection to be performed during 1R17 will be performed by personnel of multiple site disciplines including those groups who performed the inspections at 1R16. These personnel will include a VT-2 inspector who is knowledgeable in the detection and discrimination of leakage evidenced by the accumulation of boron deposits.	X		Fall 2002
Entergy will provide the NRC with the following information for ANO-1 a. A description of the extent of RPV head nozzle leakage and cracking. This information will include the number, location, size and nature of each crack detected. b. A description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs and other corrective actions taken to satisfy applicable regulatory requirements.	X		Within 30 days after plant restart following the fall 2002 refueling outage.