Entergy

Entergy Operations, Inc.
17265 River Road
Killona, LA 70066
Tel 504 464 3120

Early C. Ewing, III
General Manager, Plant Operations
Waterford 3

W3F1-2001-0081
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PR

September 4, 2001

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
30 Day Response To NRC Bulletin 2001-01 For Waterford 3;
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 significant resources in researching the existing condition of the Arkansas Nuclear One, Unit 1 (ANO-1) control rod drive mechanism penetration nozzles 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 mutually agree that the concern

A088

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.

To address the VHP nozzle cracking concern, Entergy has performed detailed analysis 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 analysis shows that significant cracking can occur in a circumferential direction with the nozzle still having the ability to retain substantial safety margin. Entergy's inspection program for VHP nozzles is believed to be satisfactory to identify any flaws well before they could become an actual safety concern. However, Entergy will continue to evaluate upcoming VHP nozzle inspection findings within the industry to determine whether new information would alter our conclusions indicating the need to modify our inspection plans.

Attachment 1 provides the specific information requested by the NRC staff in NRC Bulletin 2001-01 for the Waterford Steam Electric Station, Unit 3 (Waterford 3). In addition, Attachment 2 provides Entergy's basis for why the regulatory requirements cited in Bulletin 2001-01 are being met. As discussed in the attachments, a bare metal effective visual inspection of the Waterford 3 VHPs will be performed during the next scheduled outage scheduled for 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 Waterford 3. Commitments associated with the bulletin response are identified in Attachment 3 to this letter.

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

Very truly yours,

Early C. Ewing
General Manager, Plant Operations
Waterford 3

JTH/dbm/cbh

Attachment 1: Response To NRC Bulletin 2001-01 Waterford Steam Electric Station, Unit 3
Attachment 2: Basis for Concluding That Applicable Regulatory Requirements As Cited in NRC Bulletin 2001-01 Are Met
Attachment 3: Commitment Identification/Voluntary Enhancement Form

cc: E.W. Merschoff (NRC Region IV), N. Kalyanam (NRC-NRR), NRC Resident Inspectors Office

ATTACHMENT 1

TO

W3F1-2001-0081

RESPONSE TO NRC BULLETIN

2001-01

WATERFORD STEAM ELECTRIC STATION, UNIT 3

LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

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;

Waterford 3 Response

Waterford Steam Electric Station, Unit 3 (Waterford 3) 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 [Ref. 1]. By letter dated, August 21, 2001, the Nuclear Energy Institute (NEI) submitted MRP-48 [Ref. 1] on behalf of the industry to the NRC staff. This report provided an industry response to information requested in Item 1.a of Bulletin 2001-01.

Using the criteria stated in NRC Bulletin 2001-01, Waterford 3 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 Waterford 3 was developed from CE-NSPD-1074. [Ref. 2]

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;

Waterford 3 Response

Waterford 3 has 102 total RPV head penetrations of which 91 are control element drive mechanism (CEDM) nozzles, 10 incore instrumentation (ICI) nozzles, and 1 vent line nozzle. The requested nozzle information is provided in Table 2-3 of MRP-48 [Ref. 1].

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

Waterford 3 Response

As reported in Table 2-1 of MRP-48 [Ref. 1], Waterford 3 has reflective contoured RPV head insulation.

Waterford 3 has TRANSCO metal reflective insulation on the reactor vessel top head. The insulation panels are contoured and fit directly on top of the head with little or no gaps between the insulation and the surface of the head. The insulation panels consist of 24 gauge austenitic stainless steel sheet metal enclosing internal layers of waffled stainless steel foil 0.002 inches thick. A flexible collar made of woven fiberglass cloth (donut shaped) is placed around each CEDM and ICI nozzle. The head vent pipe is encapsulated by the metal reflective insulation and does not have a collar.

The insulation was installed in three different arrays around the head. The top center and the middle layers were not designed to be removable. The outer layer was designed to be removable to allow access to the RPV head circumferential weld. The top center panel is one piece with a radius of 28.5 inches and covers 23 CEDM nozzles (1 – 23). This round panel was lowered from the top of the control rod drives down to the RPV head surface. The middle layer is composed of nine panels out to a radius of 58 inches and covering CEDM nozzles 24 – 67. The flexible collars were installed first and then the metal reflective panels were fitted over the collars. The panels were fastened to the middle panel and to each other with stainless steel screws. They are not removable. The outer and final layer is composed of nine panels of removable insulation fitted around the flexible collars and fastened with buckles. The outer layer covers CEDM nozzles 68 – 91 and the 10 ICI nozzles. An additional panel of insulation covers a portion of the head vent pipe.

After installation of the insulation panels, the shroud was installed, surrounding the CEDM control rod housings and enclosing the insulation. The shroud has a lower support plate below the CEDM nozzle to housing weld. This plate is about nine inches from the top of the RPV head, which limits how far the panel can be raised. The presence of the steel shroud, the support plate above the CEDM nozzles and the very limited space on top of the RPV head make it difficult to remove the insulation panels. Since the majority of the panels are not removable, they would have to be cut by remote operated tooling or extension rods to remove them and get them out through the shroud windows. This is expected to be a time consuming and dose intensive process.

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;

Waterford 3 Response

As reported in Table 2-1 of MRP-48 [Ref. 1], Waterford 3 has not performed bare metal RPV head and nozzle inspections within the past four years. Only the inspections required by Generic Letter (GL) 88-05 have been performed in the last four years. The GL 88-05 inspections have not identified any signs of VHP leakage; however, they do not require the removal of RPV head insulation. Just over four years ago, in April 1997, the insulation was removed around the perimeter of the RPV head and approximately twenty percent of the nozzles were visually inspected. No signs of leakage was detected during this partial bare-metal visual inspection. As described above, only the insulation around the outer perimeter of the RPV head is designed to be removable.

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.

Waterford 3 Response

General Description

Waterford 3 is a Combustion Engineering plant. The CEDM housings are attached to the CEDM nozzles at a constant elevation. Cables to the 91 CEDMs and to the 10 ICIs travel down and connect to the top of the CEDM housings (or ICI nozzles) at the maintenance structure. The missile shield is located above the CEDMs and is supported by the secondary shield walls (also called bio shield walls). The component cooling water system (CCW) supplies water to the CEDM coolers on top of the missile shield. The CEDM cooling system air ducts run from the coolers on top of the missile shield down to the cooling shroud just above the reactor vessel head. See Figure 1.

Missile Shield

The Waterford 3 missile shield is centered above the reactor vessel and is supported by the secondary shield walls. It rolls on crane rails anchored on the top of the secondary shield walls. The missile shield is a 28' by 19.5' by 4.5' thick reinforced concrete slab and has a steel beam girder frame, enabling it to span the refueling canal. During normal operations, the missile shield is bolted to the secondary shield walls. The CEDM cooling system chillers, blowers, and ductwork are anchored on top of the missile shield. The top of the missile shield is at the 67.5 feet elevation. The top of the secondary shield walls are at the 62.25 feet elevation. The ducts come down the north and south ends of the missile shield and connect to the cooling shroud above the vessel head.

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 at a constant elevation (approx. 28 feet). Slightly below this elevation there is a steel support plate (called plenum plate) 2 inches thick, which is attached to the shroud. This plate restricts any lateral movement of the CEDM nozzles. The shroud is a cylindrical structure composed of a rolled 0.75 inch thick steel plate bolted to the top of the vessel head and surrounding the CEDM and ICI nozzles. The shroud is 10 feet high and serves the purpose of a cooling plenum for the CEDM cooling system as well as being part of the vessel head lifting rig. The CEDM housings extend up through the top of the RPV head lifting structure to an approximate elevation of 47 feet.

Other Components

Other components in the area are the supply and return lines from the component cooling water system to the CEDM coolers on top of the missile shield and the reactor head vent

piping which is attached to the top of the head and comes out of the shroud at an elevation between 24 and 28 feet. The vent line eventually connects with the pressurizer vent line.

Other Structures

Other structures within the refueling canal above the reactor vessel include the cable trays carrying the cables to the reactor vessel and cooling ducts. The cable trays run above the refueling canal on both sides and are supported by the secondary shield walls.

The CEDM cooling ducts come off the missile shield to the CEDM cooling shroud and are supported by the missile shield and the refueling canal walls. These ducts connect to the 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. The ducts run down the north and south sides of the missile shield and are removed and stored out of the way for refueling operations.

There is a maintenance structure above the top of the CEDM housings. Steel beams are supported from the secondary shield walls and cantilever over the refueling canal to support the maintenance structure. The maintenance structure is constructed of steel members. It provides supports for the cables to the reactor vessel, access to the connections at the cable trays at the top of the structure and the top of the CEDMs, and a platform for maintenance of the CEDMs and cables.

Other Cabling

The CEDM control and power cables come into the refueling canal area in the cable trays running along the secondary shield walls. Also contained in these cable trays are the ICI cables and other instrumentation cabling going to the reactor vessel head area. All of these cables come onto the maintenance structure above the CEDM housings.

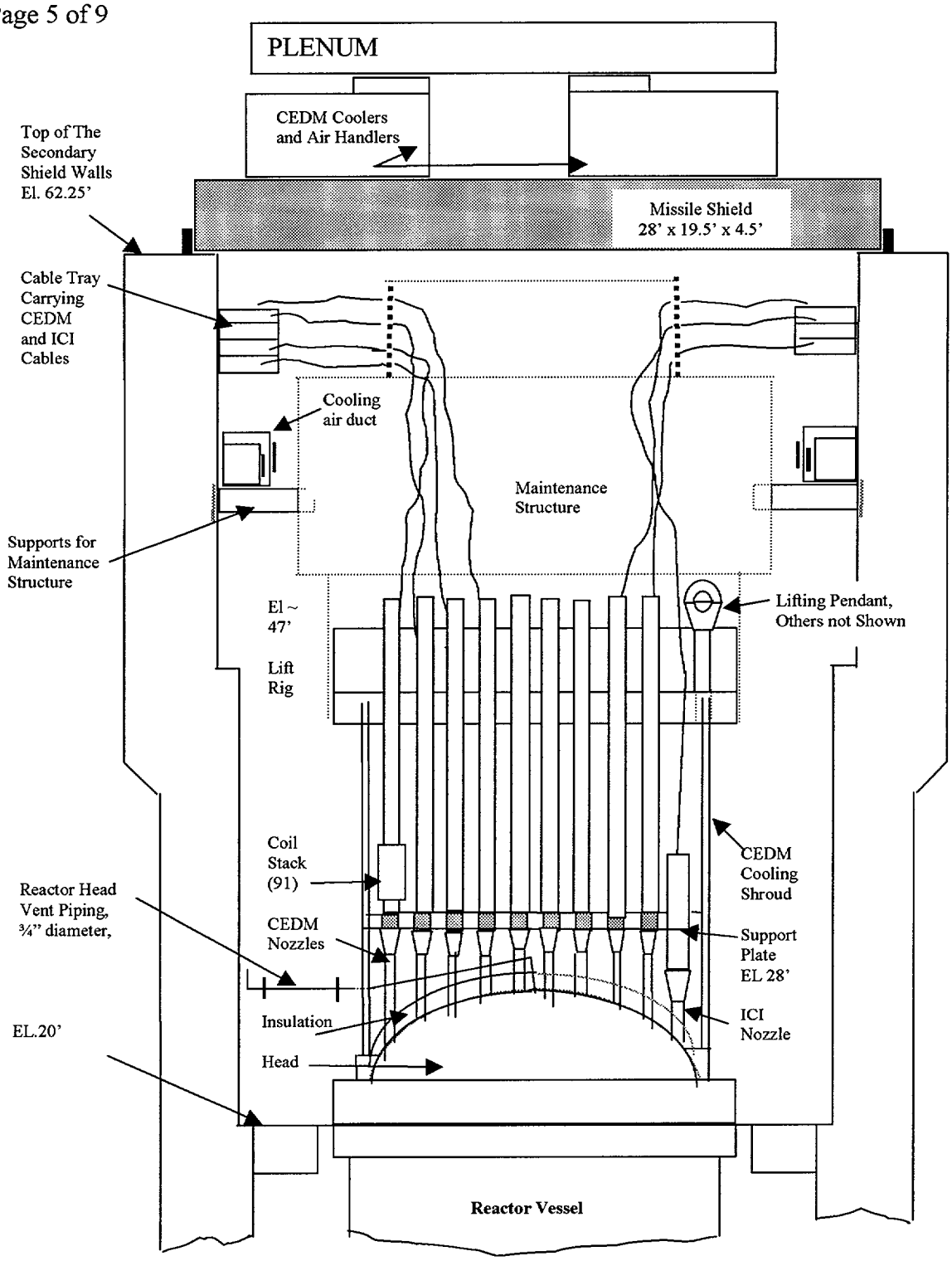

FIGURE 1
General View of Waterford 3 Structures and Components Above the Reactor Vessel

(Approximate Dimensions. 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:***

Waterford 3 Response

Not Applicable to Waterford 3

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

Waterford 3 Response

Not Applicable to Waterford 3

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.***

Waterford 3 Response

Waterford 3 plans to perform an *effective visual* examination of essentially 100% of the outer bare metal surface of the vessel head penetrations for evidence of leakage during the next scheduled refueling outage. 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. As recommended by the bulletin, an *effective visual* examination is the appropriate inspection method for Waterford 3, based on its design and effective time-at-temperature.

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 NDE equipment and trained and qualified workforce, and the impact to the refueling outage schedule.

4.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[1]] 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.

Waterford 3 Response

Waterford 3 plans to perform an *effective visual* examination of essentially 100% of the outer bare metal surface of the vessel head penetrations during the next scheduled refueling outage therefore NRC Bulletin item 4.b(1) is not applicable to Waterford 3.

In addition to the information provided in Attachment 2 specific to each of the applicable regulatory requirements cited in NRC Bulletin 2001-01, the following information is provided to support the inspections and other actions described by Entergy. This information is generally applicable to all of the cited regulatory requirements and is provided without specific reference to which it applies.

- 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 MRP that will provide an estimate of the likelihood of a pipe rupture in the CEDM penetrations. It should be noted that the NRC staff has accepted an approach to determine failure probability for SCC using the PRAISE and SARA codes for risk informed ISI programs.

- The assumption that an initiating event frequency of 1 for a rupture of a CEDM penetration made by the NRC staff 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.

[1] During an August 15, 2001 public meeting the NRC staff stated that 4.b(1) should have read "effective visual examination" instead of "qualified visual examination."

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. The estimate for the pipe failure frequency and the impact of inspections in reducing a rupture are products of the analytical work under way and will be compared against the above stated estimate of pipe failure frequency when the work is complete.

- 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 underway 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 control rod drive mechanisim (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 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 through wall 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 similar trend would be expected for the CE nozzles (except that 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 Waterford 3 together with a lower normalized EFPY for Waterford 3 compared to ANO-1 [Ref. 1] suggests that extensive degradation of the CEDM nozzles is unlikely at this time. This comparison shows that, even though the present inspection plan is to perform an effective visual examination of essentially 100 percent of the CEDM penetrations, a reasonable sample size would be adequate to ensure the detection of degraded condition. The structural analysis information could be used to select the penetration locations for the inspection campaign.

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.

Waterford 3 Response:

Entergy will provide the requested information for Waterford 3 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. "Evaluation of Reduction in Fluid Temperature in the Upper Plenum Due to Increased Bypass Flow;" CE-NSPD-1074, CEOG Task 953, March 1997.

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

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

ATTACHMENT 2

TO

W3F1-2001-0081

BASIS FOR CONCLUDING THAT

APPLICABLE REGULATORY REQUIREMENTS

AS CITED IN NRC BULLETIN 2001-01 ARE MET

LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

NRC Bulletin 2001-01, item 4.b, references the Bulletin section entitled Applicable Regulatory Requirements. This section of the Bulletin lists several provisions of the NRC regulations and plant operating licenses that pertain to the issue of vessel head penetration (VHP) nozzle cracking. These provisions are:

- 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 attachment discusses how the regulatory requirements contained in these provisions are met by the inspections described in response to NRC Bulletin 2001-01, Request 4.a.

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."

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 meets these requirements. The following information demonstrates how Entergy complies with the design criteria for the cracking of reactor pressure vessel (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 standard review plans (SRPs) 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 Waterford 3'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."

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 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, Waterford 3 plant Technical Specifications include a limiting condition for operation and associated action statements addressing reactor coolant system leakage. Per Technical Specification 3.4.5.2 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 3.4.5.2 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 3.4.5.2 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 and leakage from reactor coolant system pressure isolation vlaves, 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 3.4.5.1 are as follows:

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

a. A containment atmosphere particulate radioactivity monitoring system,
b. The containment sump level and flow monitoring system, and
c. Either the containment air cooler condensate flow switches on at least three coolers or a containment atmosphere gaseous radioactivity monitoring system.

Applicability: Modes 1, 2, 3 and 4.

Action:
With only two of the above required leakage detection systems operable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours.

Technical Specification Bases 3.4.5.1 for the reactor coolant system leakage detection systems state in part:

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.

Criterion 30 - Quality Of Reactor Coolant Pressure Boundary Criterion:

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Entergy's response for Waterford 3 as contained in the Waterford 3 Final Safety Analysis Report (FSAR) states in part:

> *The RCPB components are designed, fabricated, erected, and tested in accordance with the ASME Code, Section III. All major components are classified safety class I as specified in Subsection 3.2.2. Accordingly, they receive all of the quality assurance measures appropriate to that classification. Detection and identification of reactor coolant leakage is discussed in Subsection 5.2.5. The system is designed to detect and, to the extent practical, identify the source of reactor coolant leakage.*

Subsection 5.2.5, "Detection Of Leakage Through Reactor Coolant Pressure Boundary" states in part:

> *The reactor coolant pressure boundary (RCPB) Leakage Detection System is designed to detect and identify abnormal leakage within the limits given in the Technical Specifications. The Leakage Detection System is capable of reliably:*
>
> *a) Detecting unidentified sources of abnormal leakage as low as 1.0 gpm.*
>
> *b) Identifying particular sources of abnormal leakage as low as 1.0 gpm.*
>
> *The RCPB Leakage Detection System is consistent with the recommendations of NRC Regulatory Guide 1.45 (May 1973). Leakage Detection System is capable of performing the functions following seismic events that do not require plant shutdown. In addition, the airborne particulate radioactivity monitoring system is designed to remain functional when subjected to the safe shutdown earthquake (SSE).*

Most leaks from reactor coolant system Alloy 600 control rod drive mechanism (CRDM) penetrations have been well below the sensitivity of on-line leakage detection systems. 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 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 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.

Entergy has met and will continue to meet Technical Specification requirements for reactor coolant system leakage at Waterford 3. 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."

Title 10 of the Code of Federal Regulations, Part 50.55a 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.

Waterford 3 is in its second 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

[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.

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.

In addition to the inspection requirements of ASME Section XI, Waterford 3 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 staff 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.

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 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."

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 is 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 requirement 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 staff's generic communication process for an Information Notice, which reports industry experience, but does not require a response to the NRC staff. 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 staff 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 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 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 all success paths, whether specifically stated or not, are first directed to ensuring public health and safety and second to restoring the systems, structures, or components (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 SSC 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 the criteria of Appendix B to 10 CFR Part 50, with the additional guidance of Generic Letter 91-18, for Waterford 3, the inspections described for NRC Bulletin item 4.a are adequate to address the potential for PWSCC of the CEDMs.

COMMITMENT IDENTIFICATION/VOLUNTARY ENHANCEMENT FORM

Attachment 3 to W3F1-2001-0081
30 Day Response To NRC Bulletin 2001-01 For Waterford 3
Circumferential Cracking of VHP Nozzles

COMMITMENT(S)	ONE-TIME ACTION	CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (IF REQUIRED)	ASSOCIATED CR OR ER
Waterford 3 plans to perform an *effective visual* examination of essentially 100% of the outer bare metal surface of the vessel head penetrations for evidence of leakage during the next scheduled refueling outage. 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.	X		RF11 (Spring 2002)	ER-W3-99-0198-04-00
Entergy will provide the requested information for Waterford 3 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 RF11 restart	