Entergy

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Bryan S. Ford
Senior Manager, Nuclear Safety & Licensing

CNRO-2009-00001
January 23, 2009

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Relief Requests for Third 120 Month Inservice Inspection Interval

| | |
|---|---|
| River Bend Station<br>Docket No. 50-458<br>License No. NPF-47 | Grand Gulf Nuclear Station<br>Docket No. 50-416<br>License No. NPF-29 |
| Arkansas Nuclear One<br>Unit 1<br>Docket Nos. 50-313<br>License Nos. DPR-51 | Waterford 3 Steam Electric<br>Station<br>Docket No. 50-382<br>License No. NPF-38 |

REFERENCES:

1. Letter CNRO-2008-00016 from Bryan S. Ford, Entergy Operations Inc. to Document Control Desk, USNRC, "Relief Requests for Third 120 Month Inservice Testing Interval" dated May 20, 2008
2. Letter dated January 15, 2009 from N. Kalyanam, Project Manger Plant Licensing Branch IV Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to Bryan S. Ford, "Arkansas Nuclear One, Unit 1, Grand Gulf Nuclear Station-Request For Additional Information RE: Relief Requests For Third 120 Month Inservice Testing Interval CEP-CISI-001 (TAC Nos. MD8817 and MD8818)

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(a)(3)(ii) or 10 CFR 50.55a(a)(3)(i) as indicated in the attachment, Entergy Operations, Inc. (Entergy) requests relief or alternatives for the Inservice Inspection Program. These requests supersede similar requests dated May 20, 2008 in their entirety. This submittal corrects certain errors in the earlier requests. This change reconciled differences in the header and the applicability section of two of the relief requests and updated to a later edition of a code case in a third request. This change also responds to the RAI from Reference 2 above. These requests are needed to support the 120 month update

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for the 3rd interval (fourth for ANO 1). The requests apply to Arkansas Nuclear One - Unit 1, Grand Gulf Nuclear Station, River Bend Station, and Waterford Steam Electric Station, Unit 3. Most of these requests are similar to the requests approved for use for the current interval. The details to the 10 CFR 50.55a requests are provided in the attachment.

Entergy requests approval as soon as practical.

If you have any questions or require additional information, please contact Bryan Ford at (601) 368- 5516.

This letter contains no new commitments.

Sincerely,

BSF/WBB
Attachment: ASME Section XI Relief Requests

cc: Mr. T. A. Burke (ECH)
Mr. E. E. Collins, Regional Administrator, RIV
Mr. C. F. Lyon, Project Manager, GGNS, RBS
Mr. J. R. Douet (GGNS)
Mr. J. S. Forbes (ECH)
Mr. J. T. Herron (ECH)
Mr. K. Kalyanam, Project Manager, W-3
Mr. T. G. Mitchell (ANO)
Mr. M. Perito (RBS)
Mr. N. S. Reynolds (W&S)
Mr. L. J. Smith (Wise, Carter)
Mr. K. T. Walsh (W-3)
Mr. A. B. Wang, Project Manager, ANO

**Attachment 1**

**CNRO-2009-00001**

**ASME Section XI Relief Requests**

**ENTERGY OPERATIONS, INC.**
**ARKANSAS NUCLEAR ONE, UNIT 1**
**GRAND GULF NUCLEAR STATION**
**RIVER BEND STATION**
**WATERFORD STEAM ELECTRIC STATION UNIT 3**
**REQUEST FOR ALTERNATIVE**
**CEP-ISI-011**

## I. COMPONENTS

| | |
|---|---|
| Code Class: | 1 |
| References: | ASME Section XI, 2001 Edition with 2003 Addenda<br>Code Case N-747<br>Technical Basis for Reactor Vessel Head-to-Flange Weld Examinations as Prescribed in ASME Case N-747, dated November 8, 2005 |
| Examination Category: | B-A |
| Item Number: | B1.40 |
| Description: | Alternative Requirements for Examination of the Reactor Vessel Head-to-Flange Weld |
| Unit/Inspection Interval Applicability: | ANO-1, Unit 1 – Fourth (4th) 10 -Year Interval<br>Grand Gulf Nuclear Station – Third (3rd) 10 - Year Interval<br>River Bend Station – Third (3rd) 10 - Year Interval<br>Waterford 3 – Third (3rd) 10 - Year Interval |

## II. CODE REQUIREMENTS

Table IWB-2500-1, Category B-A, Item Number B1.40 requires a volumetric and a surface examination to be performed once per interval on the reactor vessel head-to-flange weld. The examination includes essentially 100% of the weld length.

## III. PROPOSED ALTERNATIVE

Pursuant to 10CFR50.55a(a)(3)(i), Entergy requests authorization to utilize the alternative requirements in ASME Code Case N-747 in lieu of the requirements of Table IWB-2500-1, Examination Category B-A, Item Number B1.40.

## IV. BASIS FOR PROPOSED ALTERNATIVE

The alternative examination requirements in Code Case N-747 provide an option to reduce undue burden and worker radiation exposure, while maintaining plant safety. Specifically, it provides alternative requirements for the reactor vessel head-to-flange weld to be inspected by surface examination once each 10-year inspection interval, using the current surface examination area shown in Figure IWB 2500-5. This alternative requirement may only be implemented after the weld has received at least one inservice volumetric examination, which may be performed as part of the preservice inspection, with no service-induced flaws having been identified.

The basis for elimination of the concurrent surface and volumetric examination requirement for the head-to-flange weld is rooted in nearly 40 years of service experience for this weld. The technical bases for the alternative criteria of Code Case N-747 are provided in the associated White Paper for the action entitled, "Technical Basis for Reactor Vessel Head-to-Flange Weld Examinations as Prescribed in ASME Case N-747," dated November 8, 2005. This White Paper evaluated a number of factors including component geometry, associated stresses, fracture toughness, fatigue considerations, corrosion and industry experience with examinations. Based on this evaluation, the White Paper concluded that only a surface examination should be required for the reactor vessel head-to-flange weld provided no defects had been detected during any preservice or inservice examinations. In addition, the Examination Category B-P pressure tests and visual examinations normally conducted in conjunction with refueling outages will also continue.

This weld is not a dissimilar metal or Alloy 600 weld, and is a full penetration design. In addition, there have been no defects detected during preservice or inservice examinations. It is therefore concluded that the concurrent volumetric and surface examination requirement may be eliminated for the reactor vessel head-to-flange weld, and that the outer surface examination discussed above provides an acceptable level of quality and safety.

## V. CONCLUSION

10CFR50.55a(a)(3) states:

"Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

As discussed in Section IV above, the proposed alternative requirements in Code Case N-747 provide an acceptable level of quality and safety to the requirements in ASME Section XI, 2001 Edition with 2003 Addenda, Table IWB-2500-1, Category B-A, Item Number B1.40. Therefore, Entergy requests authorization to perform the requested alternative to the Code requirement pursuant to 10CFR50.55a(a)(3)(i).

**ENTERGY OPERATIONS, INC.**
**ARKANSAS NUCLEAR ONE, UNIT 1**
**GRAND GULF NUCLEAR STATION**
**RIVER BEND STATION**
**WATERFORD STEAM ELECTRIC STATION UNIT 3**
**REQUEST FOR ALTERNATIVE**
**CEP-ISI-012**

## I. COMPONENTS

| | |
|---|---|
| Code Class: | Not Applicable |
| References: | ASME Section XI, 2001 Edition with 2003 Addenda<br>Code Case N-753 |
| Examination Category: | Not Applicable |
| Item Numbers: | Not Applicable |
| Description: | Alternative Requirements to the Visual Acuity Demonstration Requirements of IWA-2321(a) |
| Unit/Inspection Interval Applicability: | ANO-1, Unit 1 – Fourth ($4^{th}$) 10-Year Interval<br>Grand Gulf Nuclear Station – Third ($3^{rd}$) 10-Year Interval<br>River Bend Station – Third ($3^{rd}$) 10-Year Interval<br>Waterford 3 – Third ($3^{rd}$) 10-Year Interval |

## II. CODE REQUIREMENTS

IWA-2321(a) requires that NDE personnel be administered the following vision tests annually: "Personnel shall demonstrate natural or corrected near-distance acuity of 20/25 or greater Snellen fraction, with at least one eye, by reading words or identifying characters on a near-distance test chart, such as a Jaeger chart, that meets the requirements of IWA-2322. Equivalent measures of near-distance acuity may be used. In addition, personnel performing VT-2 or VT-3 visual examinations shall demonstrate natural or corrected far-distance acuity of 20/30 or greater Snellen fraction or equivalent with at least one eye."

## III. PROPOSED ALTERNATIVE

Pursuant to 10CFR50.55a(a)(3)(i), Entergy requests authorization to utilize the alternative requirements in ASME Code Case N-753 in lieu of the requirements of IWA-2321(a).

## IV. BASIS FOR PROPOSED ALTERNATIVE

Code Case N-753 provides an alternative to the visual acuity demonstration requirements of IWA-2321(a) that will allow the testing to be administered and documented by an Optometrist, Ophthalmologist, or other health care professional who administers vision tests.

The visual acuity testing for NDE personnel performing ASME Section XI examinations is required to be administered annually. In addition to this vision testing, which is typically administered by utility personnel, many NDE personnel also have annual visual acuity testing in conjunction with routine eye examinations administered by an Optometrist, an Ophthalmologist, or other health care professional who administers vision tests.

Optometrists, Ophthalmologists, and other health care professionals who administer vision tests are typically educated and experienced in the proper techniques for vision testing, such as the Snellen fraction or Jaeger chart methods required by ASME Section XI. This training and expertise provides a sound level of confidence that the visual acuity testing administered will be a reliable indicator that the tested NDE personnel can satisfactorily perform Section XI non-destructive examinations.

The testing performed by Optometrists, Ophthalmologists, and other health care professionals who administer vision tests will satisfy IWA-2321(a) requirements, including documentation which details the tests performed, compliance with IWA-2321(a) criteria and the date the testing was administered.

The use of Code Case N-753 alternative requirements allows the flexibility for utilities to accept visual acuity testing performed by outside health care professionals in lieu of the visual acuity testing performed by in-house personnel. In many instances this flexibility will eliminate duplicate testing and thus provide a reduction in the costs and manpower associated with qualifying NDE personnel.

Because Code Case N-753 does not change the qualification criteria in IWA-2321(a), the implementation of the included alternative requirements does not affect the level of quality or safety provided by NDE personnel.

## V. CONCLUSION

10CFR 50.55a(a)(3) states:

"Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

As discussed in Section IV above, the proposed alternative requirements in Code Case N-753 provide an acceptable level of quality and safety to the requirements in ASME Section XI, 2001 Edition with 2003 Addenda, IWA-2321(a). Therefore, Entergy requests authorization to perform the requested alternative to the Code requirement pursuant to 10CFR50.55a(a)(3)(i).

**ENTERGY OPERATIONS, INC.**
**ARKANSAS NUCLEAR ONE, UNIT 1**
**GRAND GULF NUCLEAR STATION**
**REQUEST FOR ALTERNATIVE**
**CEP-CISI-001**

## I. COMPONENTS

| | |
|---|---|
| Code Class: | CC |
| References: | ASME Section XI, 2001 Edition with 2003 Addenda, Article IWA-2300 and Subsubarticle IWL-2310<br>Code Case N-739 -1 |
| Examination Categories: | L-A, and L-B |
| Item Numbers: | L1.11, L1.12, L2.30 |
| Description: | Request to Utilize The Alternative Requirements of Code Case N-739-1: Alternative Qualification Requirements for Personnel Performing Class CC Concrete and Post-tensioning System Visual Examinations |
| Unit/Inspection Interval Applicability: | ANO-1, Unit 1 – Fourth (4th) 10-Year Interval<br>Grand Gulf Nuclear Station – Third (3rd) 10 - Year Interval |

## II. CODE AND REGULATORY REQUIREMENTS

IWA-2300 provides requirements for qualification of nondestructive examination personnel.

IWL-2310 provides requirements for the qualification of personnel performing Class CC Concrete and Post-tensioning System Visual Examinations

Per 10CFR50.55a(b)(2)(viii)(F), personnel that examine containment concrete surfaces and tendon hardware, wires, or strands must meet the qualification provisions in IWA-2300. The "owner-defined" personnel qualification provisions in IWL-2310(d) are not approved for use.

## III. PROPOSED ALTERNATIVE

Pursuant to 10CFR50.55a(a)(3)(i), Entergy requests authorization to utilize the alternative requirements in ASME Code Case N-739-1 in lieu of the requirements of IWA-2300.

## IV. BASIS FOR PROPOSED ALTERNATIVE

The reason the NRC has included the modification in 10CFR5.55a(b)(2)(viii)(F), that prohibits licensees to utilize the "owner-defined" personnel qualification provisions in IWL-2310 is that they believe it is inappropriate to approve Code provisions that do not contain specific containment inspection guidance when prior experience demonstrates that specific containment inspection guidance is necessary. In lieu of the IWL-2310 personnel qualification requirements, the 10CFR5.55a(b)(2)(viii)(F) modification requires that licensees implement the IWA-2300 personnel qualification provisions.

In response to this NRC concern, ASME has developed an alternative set of qualification requirements in Code Case N-739-1. This Code Case provides detailed criteria to qualify personnel who examine containment concrete surfaces and tendon hardware, wires, or strands.

Similar to the requirements in IWA-2300, Code Case N-739-1 includes plant and Section XI Subsection IWL experience requirements as well as detailed training proficiency requirements for personnel performing Class CC concrete and post-tensioning system visual examinations.

Entergy believes that the alternative requirements in Code Case N-739-1 satisfy the NRC's concern of licensees utilizing Code requirements that do not contain specific containment inspection guidance. In addition, Entergy believes that the alternative requirements in Code Case N-739-1 provide an acceptable level of quality and safety to the requirements in IWA-2300.

## V. CONCLUSION

10CFR50.55a(a)(3) states:

"Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

As discussed in Section IV above, the proposed alternative requirements in Code Case N-739-1 provide an acceptable level of quality and safety to the requirements in ASME Section XI, 2001 Edition with 2003 Addenda, Article IWA-2300. Therefore, Entergy requests authorization to perform the requested alternative to the Code requirement pursuant to 10CFR50.55a(a)(3)(i).

**ENTERGY OPERATIONS, INC.**
**ARKANSAS NUCLEAR ONE, UNIT 1**
**GRAND GULF NUCLEAR STATION**
**RIVER BEND STATION**
**WATERFORD STEAM ELECTRIC STATION UNIT 3**
**REQUEST FOR ALTERNATIVE**
**CEP-PT-002**

## I. COMPONENTS

| | |
|---|---|
| Code Class: | 1 |
| Reference: | ASME Section XI, 2001 Edition with 2003 Addenda, Article IWB-5222(b) |
| Examination Category: | B-P |
| Item Number: | B15.10 |
| Description: | The Pressure Retaining Boundary During the System Leakage Test Performed at or Near the End of Each Inspection Interval |
| Unit/Inspection Interval Applicability: | ANO-1, Unit 1 - Fourth ($4^{th}$) 10-Year Interval<br>GGNS - Third ($3^{rd}$) 10-Year Interval<br>RBS - Third ($3^{rd}$) 10-Year Interval<br>WF3 - Third ($3^{rd}$) 10-Year Interval |

## II. CODE REQUIREMENTS

ASME Section XI IWB-5222(b) states, "The pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary."

## III. PROPOSED ALTERNATIVE

Pursuant to 10CFR50.55a(a)(3)(ii), Entergy requests authorization to utilize the following alternative requirements in lieu of the requirements of Article IWB-5222(b):

The pressure retaining boundary shall be visually examined during the system leakage test conducted at or near the end of the interval shall extend to all Class 1 components.

Vent, drain and branch connections off the reactor coolant pressure boundary that have double manual isolation valves that perform no other safety function other than maintaining the Class 1 pressure boundary shall be visually examined for leakage with the inboard isolation valve in the normally closed position.

Branch connections off the reactor coolant pressure boundary that have double isolation valves that are also necessary to perform Class 2 safety functions shall be tested in conjunction with the Class 2 system pressure test each inspection period.

## IV. BASIS FOR PROPOSED ALTERNATIVE

The many vent and drain connections off the RCPB have double manual isolation valves. The requirement to extend the system leakage test boundary to the outboard valve on these vent and drain connections results in a hardship without a compensating increase in the level of quality and safety. Repositioning the inboard manual valves before and after the test will take considerable time and will result in an unnecessary increase in radiological dose to plant personnel. These off-normal configurations may also contribute to the risk of delaying normal plant start-up because of the critical path time and effort required to ensure system configuration is restored.

Based on previous pressure test dose rates, Entergy estimates that complying with the current IWB-5222(b) requirement would result in an additional accumulated dose of approximately 1 man-rem at ANO-1 and Waterford 3; and approximately 0.25 man-rem at GGNS and RBS.

The vent and drain connections are normally closed during plant operation. The outboard valves only see pressure if the inboard valve is open or leaks by the seat. Seat leakage, although undesirable, is not indicative of a flaw in the pressure boundary. Furthermore, these valves are generally located close to the main runs of pipe. The non-isolable portion of these vent and drain connections is pressurized and VT-2 examined during the test conducted at the end of the inspection interval and during the test conducted at each refueling outage. The portion that is normally isolated is VT-2 examined during each refueling outage with the inboard isolation valve closed. In the event that leakage past the inboard valve is occurring, the VT-2 exam would be performed on the pipe while pressurized.

As stated in Section III, Proposed Alternative (above), branch connections off the reactor coolant pressure boundary that have double isolation valves that are also necessary to perform Class 2 safety functions shall be tested in conjunction with the Class 2 system pressure test each inspection period. Testing these connections each inspection period in conjunction with the Class 2 system pressure tests ensures that any leakage would be identified in a timely manner and that appropriate actions, either maintenance or repair/replacement would be taken if leakage was identified from Class 1 portions of branch connections beyond the first reactor coolant pressure boundary isolation valve.

Additionally, the plant technical specifications for RCPB leakage monitoring provide reasonable assurance that appropriate actions, including plant shutdown, would be taken if leakage exceeded specified limits.

During the previous ten-year interval for the referenced Entergy plants, the NRC granted relief from the criteria of IWB-5222(b) as described herein in a letter to Entergy Operations, inc., dated February 2, 2007 (Reference TAC NOs. MD1399, MD1400, MD1401, MD1402, and MD1403).

## V. CONCLUSION

10CFR50.55a(a)(3) states:

"Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

As discussed in Section IV above, the proposed alternative requirements provide an acceptable level of quality and safety to the requirements in ASME Section XI, 2001 Edition with 2003 Addenda, Article IWB-5222(b). Therefore, Entergy requests authorization to perform the requested alternative to the Code requirement pursuant to 10CFR50.55a(a)(3)(ii).

**ENTERGY OPERATIONS, INC.**
**ARKANSAS NUCLEAR ONE, UNIT 1**
**REQUEST FOR ALTERNATIVE**
**ANO1-PT-002**

## I. COMPONENTS

| | |
|---|---|
| Components/Numbers: | Reactor Coolant Pressure Boundary<br>1. Decay Heat Removal System Loop "A", between check valves DH-14A, CF-1A, DH-13A and DH-18<br>2. Decay Heat Removal System Loop "B", between check valves DH-14B, CF-1B, DH-13B and DH-17<br>3. Pressurizer Auxiliary Spray piping between check valves DH-12 and DH-16 |
| Code Class: | 1 |
| Reference: | ASME Section XI, 2001 Edition with 2003 Addenda, IWB-5222(b) |
| Examination Category: | B-P |
| Item Number: | B15.50 |
| Description: | System Pressure Test Boundary |
| Unit / Inspection Interval Applicability: | ANO-1 Fourth (4th) 10-Year Interval |

## II. CODE REQUIREMENT(S)

ASME Section XI IWB-5222(b) requires, "The pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system."

## III. PROPOSED ALTERNATIVE

Pursuant to 10CFR50.55a(a)(3)(ii), Entergy requests authorization to visually examine the extended reactor coolant pressure boundary (RCPB) between the first and second normally closed isolation valves during the Class 2 system leakage test to be conducted at or near the end of the current inspection interval for the components identified in Section I, above. Section IV, below, provides the basis for applying this proposed alternative to each identified line.

## IV. BASIS FOR PROPOSED ALTERNATIVE

Performing leakage test of the Class 1 boundary beyond the inboard isolation valves at or near the end of each inspection interval requires conditions that place the plant in abnormal configurations or requires off-normal activities in order to pressurize the subject piping. These challenges include abnormal line-ups, installing jumpers around valve operation interlocks, installing and removing piping jumpers around valves, removing valve internals, and installing plugs. Associated with each challenge come additional burdens prior to plant restart, such as:

- High radiation exposure
- Erecting and removing scaffolding
- Welding
- Multiple disassembly and reassembly of valves and control circuitry

These off-normal configurations and challenges may also contribute to the risk of delaying normal plant start-up because of the critical path time and effort required to ensure system configuration is restored.

The piping subject to this request is outboard of the first isolation valve and is designed to RCPB conditions. However, its operation during normal conditions is typically not subject to RCPB operating conditions but to Class 2 system conditions of decay heat removal, auxiliary spray, or high pressure injection. While this piping is extremely difficult to test with the Class 1 leakage test, it is easily tested with the Class 2 system at Class 2 test conditions because of the check valve boundaries. Although Class 2 system pressure is lower than Class 1, it is representative of conditions for which the subject piping is exposed during both normal and accident conditions. Additionally, if the inboard valve leaked (thereby pressurizing the subject piping) and a through-wall flaw did exist that could only be detected at the higher pressure, the flaw would be discovered during the Class 1 leakage test, which is performed during each refueling outage with the inboard valve closed.

A description of each piping segment subject to this request and the burdens associated with performing the Class 1 leakage test currently required by ASME Section XI is provided below.

A. Decay Heat Removal Loop "A" Piping between Check Valves DH-14A, CF-1A, DH-13A and DH-18

The Decay Heat Removal Loop "A" piping and valves associated with this proposed alternative are shown in Figure 1.

The Class 2 function of the piping upstream of valve DH-14A is to provide a pathway to inject borated water from pressurized Core Flood Tank T-2A directly into the reactor vessel in the event of a loss of coolant accident (LOCA). This portion of piping between valves DH-14A, CF-1A, DH-13A and DH-18 is pressurized between 580 psig and 620 psig during normal plant operation.

Performing a ten-year Class 1 system leakage test on the extended RCPB piping of Decay Heat Removal Loop "A" between check valves DH-14A, CF-1A, DH-13A, and DH-18 involves the following actions:

1. Erect scaffolding to access the valve nearest the reactor vessel;

2. Disassemble the valve and install a hydro plug;

3. Temporarily reassemble the valve;

4. Perform the system leakage test;

5. Disassemble the valve and remove the hydro plug;

6. Reassemble the valve; and

7. Remove the scaffolding.

The radiological dose rate in the general area of the associated piping and components is approximately 60 mrem/hr. Entergy estimates the identified actions require approximately 60 man-hours to complete resulting in a radiological exposure of approximately 3.6 man-Rem.

B. Decay Heat Removal Loop "B" Piping between Check Valves DH-14B, CF-1B, DH-13B and DH-17

The Decay Heat Removal Loop "B" piping and valves associated with this proposed alternative are shown in Figure 2.

The Class 2 function of the piping upstream of valve DH-14B is to provide a pathway to inject borated water from pressurized Core Flood Tank T-2B directly into the reactor vessel in the event of a LOCA. The portion of piping between valve DH-14B and valves CF-1B, DH-13B, and DH-17 is pressurized between 580 psig and 620 psig during normal plant operation.

Performing a ten-year Class 1 system leakage test on the extended RCPB piping of Decay Heat Removal Loop "B" between check valves DH-14B, CF-1B, DH-13B, and DH-17 involves the same actions identified in

Section IV.A, above, applied to Loop "B" piping and components. The radiological dose rate in the general area of the associated components is approximately 20 mrem/hr. Entergy estimates the identified actions require approximately 60 man-hours to complete resulting in a radiological exposure of approximately 1.2 man-Rem.

C. <u>Pressurizer Auxiliary Spray Piping between Check Valves DH-12 and DH-16</u>

The Pressurizer Auxiliary Spray System piping and valves associated with this proposed alternative are shown in Figure 3.

The Class 2 function of the Pressurizer Auxiliary Spray piping is to provide a boron dilution flow path to the reactor core via the pressurizer hot leg. A non-safety function provides a method to cool down and depressurize the Reactor Coolant System (RCS) using the Decay Heat Removal Auxiliary Spray System during plant shutdown. Depressurization is performed at approximately 280 psig every refueling outage. Pressurizer Auxiliary Spray is put into service to complete the RCS cooldown. While RCS is cooled down at pressures between 200 to 250 psi, the Pressurizer Auxiliary Spray line is put into service via the Decay Heat Removal System. With the Decay Heat Removal pump discharge pressure below 400 psi, Auxiliary Spray provides a continuous, small fluid volume to the reactor vessel for approximately 2 to 3 hours. Therefore, this 5-inch, non-insulated portion of the line would see approximately 280 psi during the remaining cool-down period of 2 - 3 hours.

Performing a ten-year Class 1 system leakage test on the extended RCPB piping of the Pressurizer Auxiliary Spray System piping between check valves DH-12 and DH-16 requires the same actions identified in Section IV.A, above, applied to the Pressurizer Auxiliary Spray piping and components. The section of piping between check valves DH-12 and DH-16 is 5 inches long and 1½ inches in diameter. The radiological dose rate in the general area of these components is approximately 10 mrem/hr. Entergy estimates the identified actions require approximately 40 man-hours to complete resulting in a radiological exposure of approximately 0.4 man-Rem.

During the third ten-year interval, the NRC granted relief to ANO-1 from the criteria of IWB-5222(b) for the piping discussed herein in a letter to Entergy Operations, Inc., dated January 31, 2007 (Reference TAC NO. MD1394).

## V. CONCLUSION

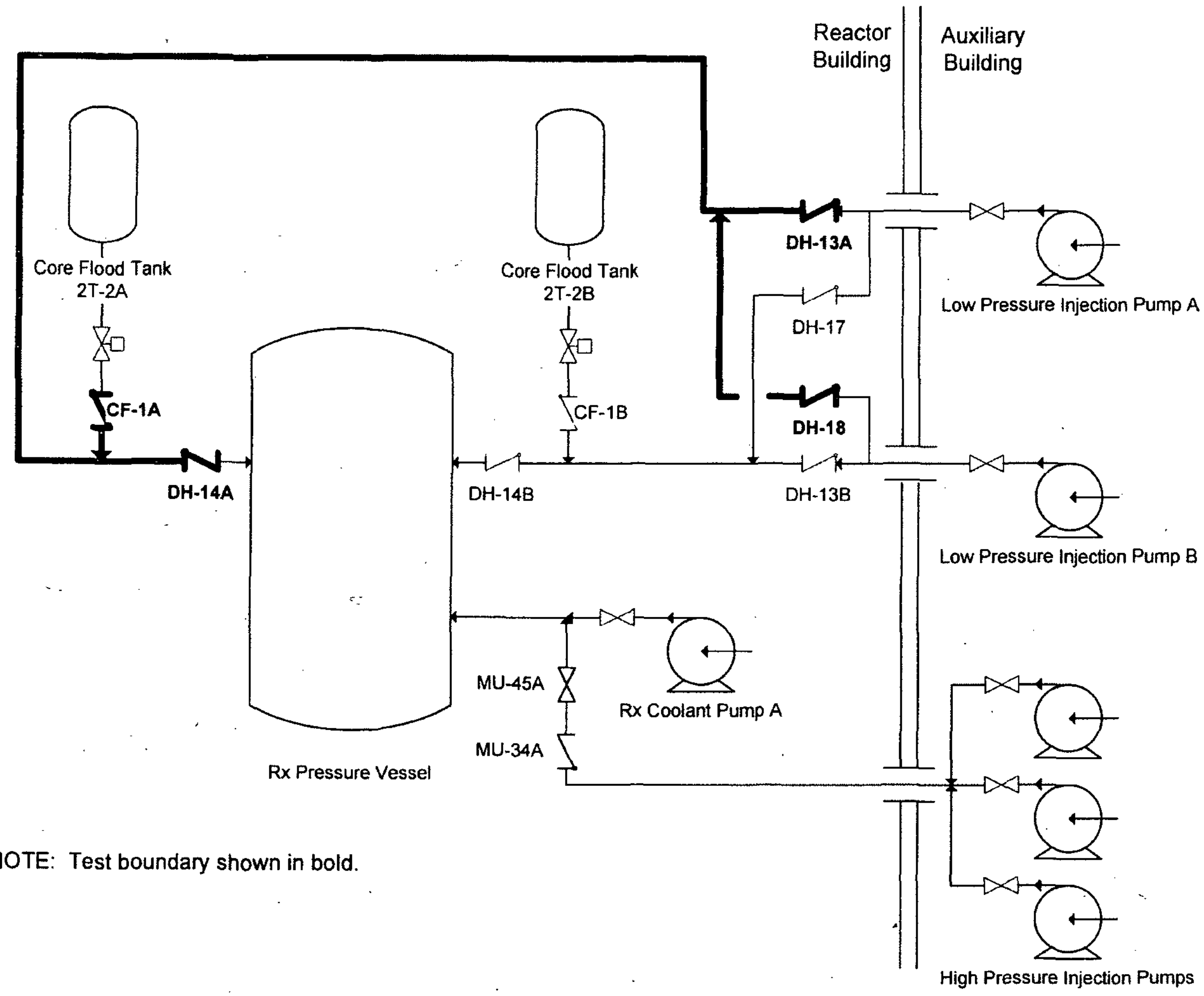
10CFR50.55a(a)(3) states:

"Proposed alternatives to the requirements of (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

As discussed in Section IV above, to perform a Class 1 system leakage test of the subject piping will result in undue burden without a compensating increase in quality and safety. The proposed alternative to visually examine the extended RCPB between the first and second normally closed isolation valves that experience Class 2 pressure during the Class 2 system leakage test conducted at or near the end of the current inspection interval provides adequate assurance of the pipe's leak tightness. Therefore, Entergy requests authorization to perform the requested alternative to the Code requirement pursuant to 10CFR50.55a(a)(3)(ii).




NOTE: Test boundary shown in bold.

**FIGURE 1**
**DECAY HEAT REMOVAL SYSTEM LOOP "A"**

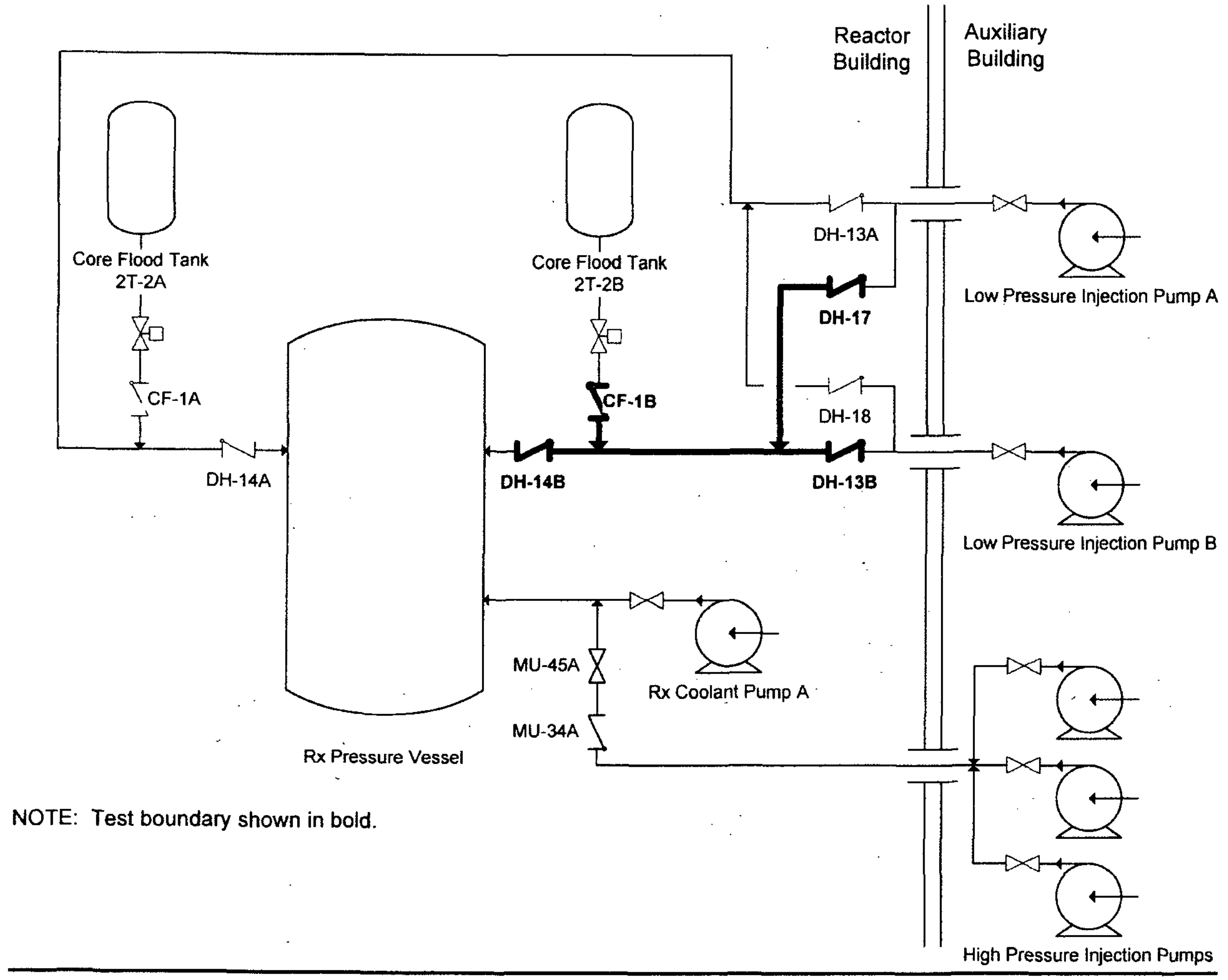



**FIGURE 2**
**DECAY HEAT REMOVAL SYSTEM LOOP "B"**

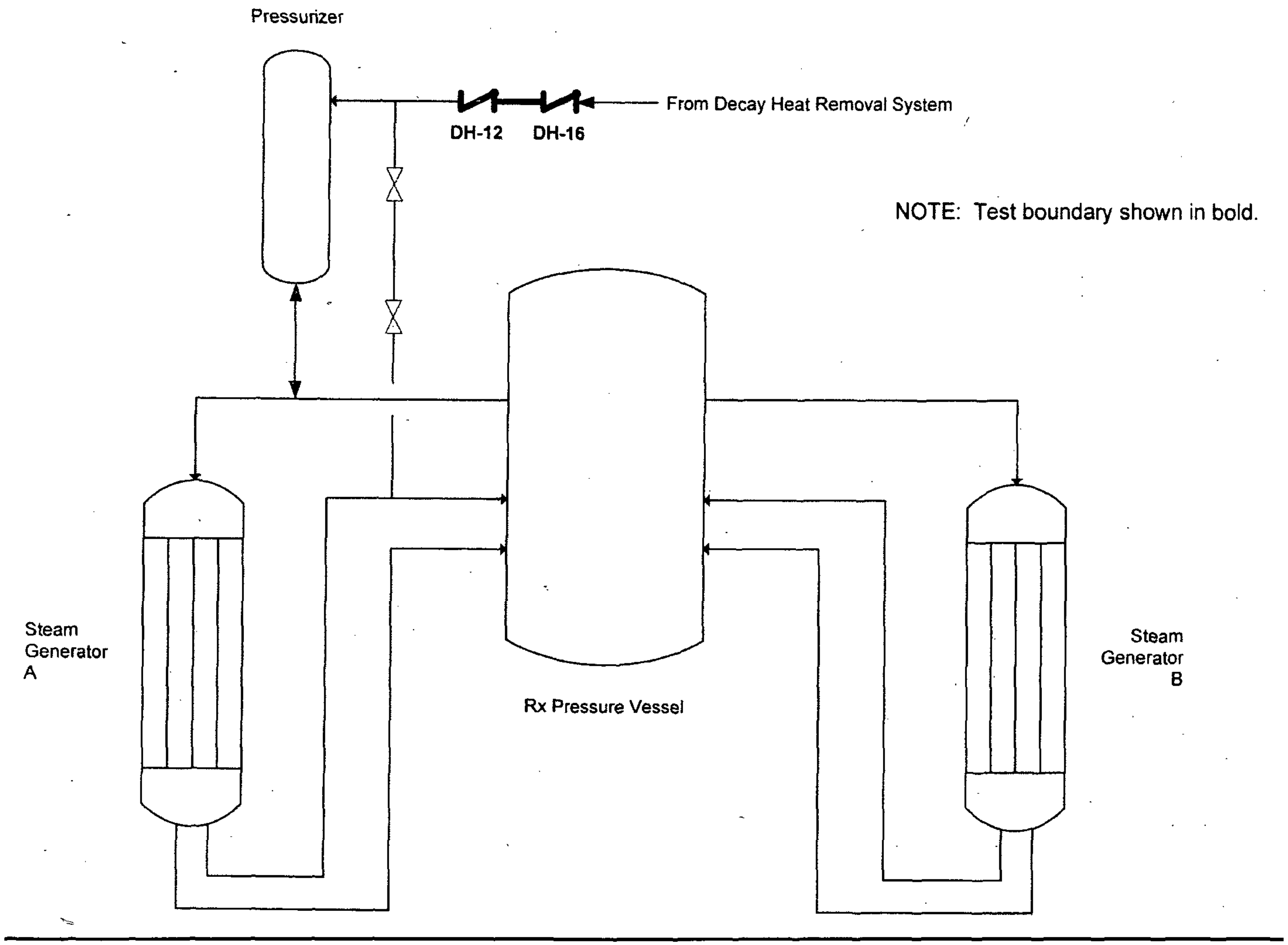



**FIGURE 3**
**PRESSURIZER AUXILIARY SPRAY SYSTEM**

**ENTERGY OPERATIONS, INC.**
**ARKANSAS NUCLEAR ONE, UNIT 1**
**WATERFORD STEAM ELECTRIC STATION UNIT 3**
**REQUEST FOR ALTERNATIVE**
**PWR-PT-001**

## I. COMPONENTS

| | |
|---|---|
| Code Class: | 1 |
| References: | ASME Section XI, 2001 Edition with 2003 Addenda<br>Code Case N-731 |
| Examination Category: | B-P |
| Item Number: | B15.10 |
| Description: | System Leakage Test (IWB-5220) of Class 1 Pressure Retaining Components. |
| Component Numbers: | N/A |
| Unit/Inspection Interval Applicability: | ANO-1, Unit 1 – Fourth ($4^{th}$) 10-Year Interval<br>Waterford 3 – Third ($3^{rd}$) 10-Year Interval |

## II. CODE REQUIREMENTS

Table IWB-2500-1, Category B-P, Item Number B15.10 requires a Visual, VT-2 examination to be performed each refueling outage on the Class 1 pressure retaining boundary in conjunction with a system leakage test per IWB-5220. Note (2) states, "The system leakage test (IWB-5220) shall be conducted prior to plant startup following a reactor refueling outage." IWB-5221(a) states, "The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power."

## III. PROPOSED ALTERNATIVE

Pursuant to 10CFR50.55a(a)(3)(i), Entergy requests authorization to utilize the alternative requirements in ASME Code Case N-731 in lieu of the requirements of IWB-5221(a). Code Case N-731 will be used for portions of Class 1 systems that are continuously pressurized during an operating cycle by statically-pressurized safety injection systems. Code Case N-731 states that for portions of Class 1 safety injection systems that are continuously pressurized during an operating cycle, the pressure associated with the statically-pressurized safety injection systems may be used in lieu of the pressure corresponding to 100% rated reactor power.

## IV. BASIS FOR PROPOSED ALTERNATIVE

Some portions of the Class 1 pressure retaining boundary cannot be pressurized to the pressure corresponding to 100% rated reactor power without installation of jumpers or other extraordinary means which could result in entering system valve lineups not authorized by plant technical specifications for Mode 3. This Request for Alternative is for Class 1 portions of safety injection systems that are continuously pressurized during an operating cycle (i.e., the portion of the safety injection system between the first-off and second-off check valves from the RCS which are maintained pressurized during the operating cycle by the safety injection accumulators).

In order to obtain the pressure corresponding to 100% rated reactor pressure a jumper would have to be installed between the reactor coolant system (RCS) and the volume between the first-off and second-off check valves. This lineup is not allowed by Technical Specifications (all vents and drains are required to remain closed) in Mode 3, the mode the RCS would have to be in to be at the required pressure.

Alternatively, this volume of pipe could be pressurized using hydrostatic testing pumps, however this would result in excessive dose, unnecessary special test procedures and unnecessary expenditure of plant resources during the ascension to power phase following a refueling outage.

Because these sections of piping are continuously pressurized, adequate time exists for leakage to be identifiable at the lower pressure the safety injection accumulators provide. This was the same conclusion that was reached in the White Paper that supported Code Case N-731. Additionally, the level and pressure of the safety injection accumulators are continuously monitored and any leakage identified from the safety injection accumulators would be investigated and identified in accordance with station operating procedures.

## V. CONCLUSION

10CFR50.55a(a)(3) states:

"Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

As discussed in Section IV above, the proposed alternative requirements in Code Case N-731 provide an acceptable level of quality and safety to the requirements in ASME Section XI, 2001 Edition with 2003 Addenda, Article IWB-5221(a). Therefore, Entergy requests authorization to perform the requested alternative to the Code requirement pursuant to 10CFR50.55a(a)(3)(i).