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Acting Director
Nuclear Safety & Licensing

CNRO-2006-00024

April 24, 2006

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Request for Alternative ANO-ISI-005
Proposed Alternative to Extend the Third Inservice Inspection Interval for Reactor Vessel Inservice Examinations

Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) proposes an alternative to the requirements of ASME Section XI, paragraph IWB-2412, *Inspection Program B*, for the Arkansas Nuclear One, Unit 1 (ANO-1). ANO-1 is currently in its third inservice inspection (ISI) interval, which began June 1, 1997 and ends May 31, 2007. ASME Section XI IWA-2430(d) allows a one-year extension of an interval, which would extend the interval to May 31, 2008. (Use of this one-year extension does not require approval from the NRC.) In order to comply with Code requirements, third interval examination of the reactor vessel welds (Examination Category B-A), the nozzle-to-vessel welds and inner radius sections (Examination Category B-D), and reactor vessel nozzle-to-piping welds (Examination Category B-J), must be performed during ANO-1's spring 2007 refueling outage (1R20). Entergy proposes to perform these examinations during the fall 2008 refueling outage (1R21). Because 1R21 is beyond May 31, 2008, Entergy is submitting Request for Alternative ANO1-ISI-005 (see Enclosure 1), which proposes an additional extension to the third ISI interval. Entergy believes extending the inspection interval to the end of 1R21 for these examinations continues to provide an acceptable level of quality and safety. Entergy is submitting this request as a result of an ongoing initiative with the Westinghouse Owners Group for extension of the Inservice Inspection requirements.

During preparation of this request, Entergy identified a condition where we had not sought NRC relief regarding inadequate coverage of certain ANO-1 reactor vessel ISI welds. This condition is being investigated in accordance with the Entergy corrective action process.

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Entergy requests NRC approval by October 1, 2006 in order to support planning activities for ANO-1's upcoming spring 2007 refueling outage (1R20). Should you have any questions regarding this submittal, please contact Guy Davant at (601) 368-5756.

This letter contains one commitment as identified in Enclosure 2.

Very truly yours,

FGB/GHD/baa

Enclosures: 1. Request for Alternative ANO1-ISI-005
2. Licensee-Identified Commitments

cc: Mr. W. A. Eaton (ECH)
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ENCLOSURE 1

CNRO-2006-00024

REQUEST FOR ALTERNATIVE
ANO-ISI-005

ENTERGY OPERATIONS, INC.
ARKANSAS NUCLEAR ONE, UNIT 1
REQUEST FOR ALTERNATIVE
ANO-ISI-005

I. COMPONENTS

The affected component is the Arkansas Nuclear One, Unit 1 (ANO-1) reactor vessel; specifically, the following ASME Section XI Examination Categories and Item Numbers covering examinations of the reactor vessel. These examination categories and item numbers are from Table IWB-2500-1 of the 1992 Edition of ASME Section XI.

Examination Category	Item Number	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Shell Welds
B-A	B1.30	Shell-to-Flange Weld
B-A	B1.51	Beltline Region Repair Welds
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Areas
B-J	B9.11	Circumferential Welds in Piping (only for the reactor vessel inlet and outlet nozzle to piping welds)

Code Class: 1

References:

1. R. Gramm of the NRC to G. Bischoff of the WOG, "Summary of teleconference with the Westinghouse Owners Group regarding potential one cycle relief of reactor pressure vessel shell weld inspections at pressurized water reactors related to WCAP-16168-NP, *Risk-Informed Extension of Reactor Vessel In-Service Inspection Intervals*," January 27, 2005

2. ASME Code Case N-691, "Application of Risk-Informed Insights to Increase the Inspection Interval for Pressurized Water Reactor Vessels," Section XI, Division 1, November 2003

3. NRC Memorandum, Thadani to Collins, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule (10 CFR 50.61)," December 31, 2002

4. Nuclear Regulatory Commission Reactor Vessel Integrity Database, Version 2.0.1, July 6, 2000

5. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis," November 2002

6. BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," B&WOG Generic License Renewal Program, June 1996

7. BAW-2308, Revision 1-A, "Initial RT_{NDT} of Linde 80 Weld Materials," August 2005

8. Nuclear Regulatory Commission, "Safety Evaluation for Topical Report BAW-2308, Revision 1, 'Initial RT_{NDT} of Linde 80 Weld Materials' (TAC No. MB6636)," August 2005

9. NRC SER dated June 19, 1996, "Arkansas Nuclear One, Unit 1 Second 10-Year Inspection Interval Relief Request No. 95-001," (1CNA069601)

10. Westinghouse Owners Group letter WOG-05-100, *Cover Letter and Template for WOG Members' Use to Request a One Operating Fuel Cycle RV ISI Relief Request (MUHP-5097/5098/5099, Tasks 2008/2059)*, dated March 3, 2005

Unit / Inspection Interval: ANO-1 / Third (3rd) 10-Year Interval

II. CODE REQUIREMENTS

ASME Section XI IWB-2412, *Inspection Program B*, requires volumetric examination of essentially 100% of reactor vessel and piping pressure-retaining welds identified in Table IWB-2500-1 once each 10-year interval. IWA-2430(d) allows inspection intervals to be extended by as much as one year if this adjustment does not cause successive intervals to be altered by more than one year.

III. PROPOSED ALTERNATIVE

Pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) proposes an alternative from the requirement of IWB-2412 that volumetric examination of essentially 100% of reactor vessel and piping pressure-retaining welds, Examination Categories B-A, B-D, and B-J welds associated with the reactor vessel (identified in Table 1), be performed once each 10-year interval. Specifically, Entergy proposes to extend the inservice inspection (ISI) interval for the identified Examination Categories B-A, B-D, and B-J welds to the end of 1R21 (approximately 180 days beyond the currently scheduled interval and the Code-allowed one-year extension).

The intent of the requested extension is to defer the subject examinations to 1R21. This effort uses ASME Section XI Code Case N-691 (Reference 2) as a basis for using risk-informed insights to show that extending the inspection interval from 10 to 20 years results in a change in reactor vessel failure frequency that satisfies the requirements of NRC Regulatory Guide 1.174 (Reference 5).

IV. BASIS FOR PROPOSED ALTERNATIVE

A. Background

ANO-1 is currently in its third inservice inspection (ISI) interval, which began June 1, 1997, and ends May 31, 2007. ASME Section XI IWA-2430(d) allows a one-year extension of an interval, which would extend the interval to May 31, 2008. (Use of this one-year extension does not require approval from the NRC.) In order to comply with Code requirements, third interval examination of the reactor vessel welds (Examination Category B-A), the nozzle-to-vessel welds and inner radius sections (Examination Category B-D), and reactor vessel nozzle-to-piping welds (Examination Category B-J), must be performed during ANO-1's spring 2007 refueling outage (1R20). Entergy proposes to perform these examinations during the fall 2008 refueling outage (1R21).

B. Basis for Proposed Alternative

The requirements for a technical basis to extend the 10-year reactor vessel ISI interval to the end of 1R21 are contained in a letter from R. Gramm of the NRC to G. Bischoff of the Westinghouse Owners Group (WOG), dated January 27, 2005 (Reference 1). The technical justification consists of five areas as identified in Reference 1. These are:

A. Plant-specific reactor vessel ISI history

B. PWR reactor vessel ISI history

C. Degradation mechanisms in the reactor vessel

D. Material condition of the reactor vessel relative to embrittlement

E. Operational experience relative to reactor vessel structural integrity challenging events

Each area is discussed below.

1. Plant-Specific Reactor Vessel ISI History

ANO-1 is in its third ISI interval; therefore, the preservice and two inservice inspections have been performed on the Examination Category B-A, B-D, and B-J welds associated with the reactor vessel (except as noted in Table 1). These examinations were performed in accordance with the edition of ASME Section XI in affect at the time and achieved coverage as shown in Table 1; no reportable indications were found. Based on the examination method and coverage obtained, it is reasonable to conclude that the

examinations were of sufficient quality to detect any significant flaws that could challenge reactor vessel integrity.

Due to improvements in inspection technology, the most recent inspection is considered to be the best quality of the inspections performed. Therefore, the inspection data provided in Table 1 is from the most recent inspection.

TABLE 1
ANO-1 ISI RESULTS

Weld ID	ASME Weld Category	Date Last Inspected	% Coverage obtained	# of reportable indications	# of indications currently being monitored	Growth of indications currently being monitored (in)
01-001	B-A	3/95	72.21	0	0	N/A
01-002	B-A	3/95	74.6	0	0	N/A
01-003	B-A	3/95	100	0	0	N/A
01-004	B-A	3/95	100	0	0	N/A
01-005	B-A	3/95	32.99	0	0	N/A
01-006 Note 1	B-A	N/A	0	0	0	N/A
01-007	B-A	3/95	100	0	0	N/A
01-008	B-A	3/95	100	0	0	N/A
01-009	B-A	3/95	88.09	0	0	N/A
01-010	B-A	3/95	88.09	0	0	N/A
01-010A	B-A	3/95	100	0	0	N/A
01-011	B-D	3/95	45.46	0	0	N/A
01-011IR	B-D	3/95	73.78	0	0	N/A
01-012	B-D	3/95	68.81	0	0	N/A
01-012IR	B-D	3/95	66.84	0	0	N/A
01-013	B-D	3/95	82.88	0	0	N/A
01-013IR	B-D	3/95	50	0	0	N/A
01-014	B-D	3/95	68.81	0	0	N/A
01-014IR	B-D	3/95	66.84	0	0	N/A
01-015	B-D	3/95	45.46	0	0	N/A
01-015IR	B-D	3/95	73.78	0	0	N/A
01-016	B-D	3/95	68.58	0	0	N/A
01-016IR	B-D	3/05	66.84	0	0	N/A
01-017	B-D	3/95	82.88	0	0	N/A
01-017IR	B-D	3/95	50	0	0	N/A
01-018	B-D	3/95	68.81	0	0	N/A

Weld ID	ASME Weld Category	Date Last Inspected	% Coverage obtained	# of reportable indications	# of indications currently being monitored	Growth of indications currently being monitored (in)
01-018IR	B-D	3/95	66.84	0	0	N/A
19-019	B-J	3/95	98.24	0	0	N/A
19-022	B-J	3/95	98.24	0	0	N/A
13-015	B-J	3/95	100	0	0	N/A
09-015	B-J	3/95	100	0	0	N/A
14-028	B-J	3/95	100	0	0	N/A

Note 1: The NRC granted relief for the inability to perform examination of this weld for the second 10-year inspection interval (Reference 9).

2. PWR Reactor Vessel ISI History

As part of the technical basis for ASME Code Case N-691 (Reference 2), a survey of reactor vessel ISI history for 14 PWRs was performed. At the time of the survey, these 14 plants represented 301 total years of service and included reactor vessels fabricated by various vendors. The plants surveyed reported that no reportable findings had been discovered during examinations of Examination Category B-A, B-D, and B-J welds associated with the reactor vessels.

All PWR plants except one have performed their first 10-year inspections of the subject welds. No surface-breaking or unacceptable near-surface flaws have been found in any of these inspections performed per the requirements of Regulatory Guide 1.150 or ASME Section XI Appendix VIII.

3. Degradation Mechanisms in the Reactor Vessel

The welds for which the subject examinations are conducted are similar metal low alloy steel welds. The only currently known degradation mechanism for this type of weld is fatigue due to thermal and mechanical cycling from operational transients. Studies have shown that while flaw growth of simulated flaws in a reactor vessel would be small, the operational transient that has the greatest contribution to flaw growth is the cooldown transient. The cooldown transient is a low frequency transient and is not expected to occur more than a few instances during the requested inspection extension period. Therefore, any flaw growth during the requested deferral period is expected to be inherently small.

According to the WOG (Reference 10), the fatigue usage factors for the welds in the subject examinations are much less than the ASME Code design limit of 1.0 after 40 years of operation. These usage factors are calculated using a very conservative design duty cycle. It is very unlikely that more than a few of these events (e.g., heatup or cooldown) would actually occur during the extension period of this proposed alternative.

This request does not apply to any dissimilar metal welds.

4. Material Condition of the Reactor Vessel Relative to Embrittlement

The reactor vessel beltline is the limiting area in terms of embrittlement for the subject examinations. The composition of each material in the reactor vessel beltline, along with embrittlement data, can be found in the NRC Reactor Vessel Integrity Database (RVID) (Reference 4). This information for ANO-1 is provided in Table 2.

TABLE 2
ANO-1 REACTOR VESSEL INTEGRITY DATABASE INFORMATION

ANO Unit 1 Material Values contained in the RVID (Reference 4)									
Major Material Region Description				Cu	Ni	P	Un-Irradiated RT_{NDT}		RT_{PTS}
#	Type	Heat	Location	[wt%]	[wt%]	[wt%]	[°F]	Method	@32 EFPY
1	Forging	528360 (AYN)	Lower Nozzle Belt Forging	0.030	0.700	0.009	3.0	B&W Generic	86.5
2	Plate	C-5114-1	Lower Shell Course	0.150	0.520	0.010	0.0	Plant Specific	135.1
3	Plate	C-5114-2	Upper Shell Course	0.150	0.520	0.010	-10.0	Plant Specific	125.5
4	Plate	C-5120-1	Lower Shell Course	0.170	0.550	0.014	-10.0	Plant Specific	141.5
5	Plate	C-5120-2	Upper Shell Course	0.170	0.550	0.014	-10.0	Plant Specific	142.0
6	Weld	406L44	Upper/Lower Shell Circ Weld WF-112	0.270	0.590	0.016	-5.0	B&W Generic	236.7
7	Weld	821T44	Nozzle Belt/Upper Shell Circ Weld WF-182-1	0.240	0.630	0.014	-5.0	B&W Generic	191.2
8	Weld	8T1762	Lower Shell Axial Welds WF-18	0.190	0.570	0.004	-5.0	B&W Generic	194.5
9	Weld	8T1762	Upper Shell Axial Welds WF-18	0.190	0.570	0.004	-5.0	B&W Generic	194.5

It is widely recognized that the greatest possible challenge to reactor vessel integrity for a PWR is pressurized thermal shock (PTS). A PTS event can be generally described as a rapid cooling of the reactor vessel followed by a late repressurization. 10 CFR 50.61 currently provides PTS screening criteria of RT_{PTS} equal to 270°F for plates and axial welds and RT_{PTS} equal to 300°F for circumferential welds. Based on the beltline material information contained in RVID for ANO-1, the upper shell to lower shell circumferential weld, WF-112, is the limiting material with regards to the PTS screening criteria. The RT_{PTS} value at 32 effective full power years (EFPY) for this weld is 236.7°F, which is well below the current PTS screening criteria. Furthermore, as reported in BAW-2251A (Reference 6), the RT_{PTS} value for this weld at the end of license renewal (48 EFPY) is projected to be 278°F, which remains below the PTS screening criteria of 300°F.

An alternative initial reference temperature for the Linde 80 beltline welds in Babcock & Wilcox (B&W) fabricated reactor vessels is presented in BAW-2308 (Reference 7). This report was reviewed and approved for use by the NRC (Reference 8). Using the lowered Linde 80 alternative initial RT_{NDT} values, the limiting beltline material for the ANO-1 reactor vessel are the upper shell and lower shell axial welds (both), WF-18, with an RT_{PTS} value of 186.1°F at 48 EFPY. This value is also below the PTS screening criteria of 270°F for axial welds.

Finally, the NRC and industry recognize that a large amount of conservatism exists in the current PTS screening criteria. In the NRC PTS risk re-evaluation (Reference 3), results show that it may be possible to remove an amount of conservatism equivalent to reducing a plant's RT_{PTS} value by at least 70°F. While the exact amount of conservatism that will be removed has not been determined, it is clear that ANO-1 will remain below the current PTS screening criteria during the extension period.

5. **Operational Experience Relative to Reactor Vessel Structural Integrity Challenging Events**

As stated above, the greatest possible challenge to reactor vessel integrity for a PWR is PTS. Plants have taken steps such as implementing emergency operating procedures (EOPs) and operator training to reduce the likelihood of a PTS event occurring. Due to the implementation of such measures, the number of occurrences of PTS events fleet-wide is very small. When considered over the combined fleet-wide PWR operating history, the frequency of PTS events is very small. When considering the frequency of PTS events and the length of the requested extension, the probability of a PTS event occurring during the requested extension is also very low. Combining the low probability of a PTS event with the low probability of a flaw existing in the reactor vessel (given the previously discussed inspection history), the probability of reactor vessel failure due to PTS is also very small.

In the NRC PTS risk re-evaluation (Reference 3), it has been determined there are three types of accident sequences that cause the more severe PTS events and thereby dominate the risk. ANO-1 has implemented EOPs and operator training to provide assurance that the likelihood of a severe PTS

event due to these sequences over the next operating cycle that would challenge the integrity of the reactor vessel, provided a flaw was present, is very low. These sequences and details of the operating procedures pertinent to these sequences are identified below.

Sequence 1 - Any transient with reactor trip followed by one stuck-open pressurizer safety relief valve that re-closes after approximately 1 hour. Severe PTS events also require the failure to properly control high-head injection.

This event is characterized as a pressurizer steam space loss-of-coolant accident (LOCA) that results in an uncontrolled Reactor Coolant System (RCS) depressurization. Upon receipt of a reactor trip, operators would enter the reactor trip procedure (OP-1202.001). It is expected that as RCS pressure continues to decrease, Engineered Safeguard Actuation System (ESAS) will actuate on low RCS pressure, providing High Pressure Injection (HPI) flow to the system. Operators would transition to the ESAS procedure (OP-1202.010). Reactor Coolant Pumps (RCPs) would be secured due to the loss of sub-cooled margin (SCM), setting up the conditions that operators recognize as those which can lead to PTS (HPI running with RCPs secured).

RCS pressure/temperature would continue to drop until system inflow matches system outflow followed by system pressure stabilization.

The ESAS procedure provides guidance to control RCS pressure low within limits of the RCS PT curve (OP-1202.013 Figure 3), provided SCM is adequate. Assuming the pressurizer safety relief valve closes, operators would recognize rising RCS pressure, validate that HPI throttling criteria had been met, and throttle or secure HPI to prevent system re-pressurization.

If the pressurizer safety relief valve were to remain open, ANO-1 procedures are structured such that upon stabilization of system pressure, operators would restart the RCPs and transition to the forced-flow cooldown procedure (OP-1203.040). This strategy reduces the time period in which the RCS remains in a PTS vulnerable condition because of the mixing effect at the reactor vessel wall provided by the running RCPs. RCS pressure would be controlled by balancing inflow with outflow, while system cooling would be with a combination of forced-flow primary-to-secondary heat transfer combined with HPI cooling out the opening.

Sequence 2 - Large loss of secondary steam from steam line break or stuck-open atmospheric dump valves. Severe PTS events also require the failure to properly control auxiliary feedwater flow rate and destination (e.g., away from affected steam generators), and failure to properly control high-pressure injection.

This event is characterized as an overcooling event. Upon receipt of a reactor trip, operators would enter the reactor trip procedure (OP-1202.001). The affected steam generator pressure would decrease resulting in automatic actuation of Main Steam Line Isolation (MSLI) and Emergency Feedwater (EFW). These systems would function to close the affected main steam isolation valve, close the affected main feedwater isolation valve, and

provide emergency feedwater to the unaffected steam generator. Guidance is provided to operators for verifying proper MSLI and EFW response (OP-1202.012, RT-6). In the event of system or component malfunction, contingency actions are provided to operators to manually control the systems.

Operators would transition to the overcooling procedure (OP-1202.003), which directs initiation of HPI to make up for RCS inventory shrink. After the steam generator fully depressurizes which terminates overcooling, timely operator actions are necessary to "steam" the unaffected steam generator so that RCS temperature can be stabilized; otherwise, the transient could then proceed to an overheating condition, which could result in high RCS temperature and pressure. Procedure steps provide this direction to stabilize RCS temperature and to secure HPI, provided RCS inventory is satisfactory.

Sequence 3 – A small-break LOCA that exceeds normal makeup capacity. Severity of PTS event depends on break location (worst location appears to be in the pressurizer line) and primary injection systems flowrate and water temperature.

A LOCA is an accident that is caused by a break in the RCS pressure boundary. The break can be as large as a double-ended guillotine break in the hot leg, or as small as a break which results in a loss of RCS fluid at a rate that is just in excess of the available makeup capacity of the plant.

Small- and large-break LOCAs differ in their effect on the post-LOCA RCS heat removal process. For a large-break LOCA, the only path necessary for RCS heat removal in both the short and long term is the break flow with core boil-off. For small breaks, heat removal out the break is not sufficient to provide cooling and, therefore, steam generator heat removal is required. The ANO-1 EOPs take this into account and provide the necessary guidance to supplement the cooling through the break by use of primary-to-secondary heat transfer with the steam generators. Since ANO-1 EOPs are "symptom based", rather than "event based", the operator is not required to diagnose the size of the break; however, guidance is provided to bound all break spectrums to maintain RCS pressure and temperature within the limits of the applicable PT curve (OP-1202.013, Figure 3).

The RCS pressure-sensing elements in the B&W design utilized by ANO-1 are located in the hot legs, not in the pressurizer as other designs employ. The B&W design allows operators the capability to monitor RCS pressure even during the unlikely event of a LOCA at the pressurizer surge line.

As mentioned above, ANO-1 EOPs are symptom-based documents written to the standards outlined in B&W EOP Technical Basis Document. Industry standards maintaining these documents are monitored through the WOG Operations Support Committee with a target of maintaining a high degree of technical accuracy and operating experience in their basis.

ANO-1 operator training stresses fundamental EOP coping strategies in both the classroom and simulator forum. Included in the curriculum are procedure entry conditions, floating steps, fundamental rules, mitigation strategies, time

critical actions, and background information from the basis documents. Simulator evaluation scenarios utilize critical tasks as the basis for pass/fail for crew performance. These critical tasks are those chosen to be of the utmost importance to ensure the health and safety of the public are preserved and typically include the preservation and protection of fission product barriers.

V. CONCLUSION

10 CFR 50.55a(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."

The current requirements for inspection of reactor vessel pressure-retaining welds have been in effect since the 1989 Edition of ASME Section XI. The industry has expended significant cost and radiological exposure that have shown no service-induced flaws in the reactor vessel for ASME Section XI Examination Category B-A, B-D, or B-J welds associated with the reactor vessel. ASME Section XI Code Case N-691 and industry efforts have shown that risk insights can be used to extend the reactor vessel ISI interval from 10 to 20 years. This extension satisfies the change in risk requirements of Regulatory Guide 1.174 and, in accordance with 10 CFR 50.55a(a)(3)(i), maintains an acceptable level of quality and safety.

Based on these efforts having shown that the risk of reactor vessel failure with a 10-year inspection interval extension is low and achieves an acceptable level of quality and safety, it is reasonable to conclude that an extension of one refueling cycle will also achieve an acceptable level of quality and safety. Furthermore, items A through E of Section IV, above, provide a qualitative basis that the risk associated with extending the inspection interval to the end of 1R21, the ANO-1 fall 2008 refueling outage, is small. Based on this, Entergy considers the proposed alternative for the subject examinations at ANO-1 to provide an acceptable level of quality and safety. Therefore, Entergy requests that the NRC staff approve the proposed alternative pursuant to 10 CFR 50.55a(a)(3)(i).

ENCLOSURE 2

CNRO-2006-00024

LICENSEE-IDENITIFIED COMMITMENTS

LICENSEE-IDENTIFIED COMMITMENTS

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy will perform the inservice inspection of the Examination Category B-A, B-D, and B-J welds associated with the reactor vessel during the subsequent ANO-1 fall 2008 refueling outage.	X		Fall 2008 refueling outage