UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 5, 1994

Docket No. 50-313

Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

Dear Mr. Yelverton:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," ENTERGY OPERATIONS, INC., ARKANSAS NUCLEAR ONE, UNIT 1 (ANO-1) (TAC NO. M83730)

By letters dated July 1, 1992, and October 29, 1993, Entergy Operations, Inc., provided its response to GL 92-01, Revision 1. The NRC staff has completed its review of your responses. Based on its review, the staff has determined that Entergy Operations, Inc., has provided the information requested in GL 92-01.

The GL is part of the staff's program to evaluate reactor vessel integrity for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs). The information provided in response to GL 92-01, including previously docketed information, is being used to confirm that licensees satisfy the requirements and commitments necessary to ensure reactor vessel integrity for their facilities.

A substantial amount of information was provided in response to GL 92-01, Revision 1. These data have been entered into a computerized data base designated Reactor Vessel Integrity Database (RVID). The RVID contains the following tables: A pressurized thermal shock (PTS) table for PWRs, a pressure-temperature limit table for BWRs, and an upper-shelf energy (USE) table for PWRs and BWRs. Enclosure 1 provides the PTS table, Enclosure 2 provides the USE table for your facility, and Enclosure 3 provides a key for the nomenclature used in the tables. The tables include the data necessary to perform USE and RT_{pts} evaluations. These data were taken from your responses to GL 92-01 and previously docketed information. References to the specific source of the data are provided in the tables.

We request that, within 30 days, you provide confirmation of the plant-specific applicability of the Topical Reports BAW-2178P and BAW-2192P and submit a request for approval of the topical reports as the basis for demonstrating compliance with 10 CFR Part 50, Appendix G, Paragraph IV.A.1. To demonstrate that the topical reports are applicable to ANO-1, you must compare the limiting material properties of the ANO-1 reactor vessel to the values reported in the topical reports. This review will be a plant-specific licensing action. We further request that you verify that the information you have provided for your facility has been accurately entered in the summary

DF01

data file. If no comments are made in your response to the last request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel. Once your confirmation of the applicability of the topical reports and a request for approval are received, the staff will consider your actions related to GL 92-01, Revision 1, to be complete.

The information requested by this letter is within the scope of the overall burden estimated in GL 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)." The estimated average number of burden hours is 200 person-hours for each addressee's response. This estimate pertains only to the identified response-related matters and does not include the time required to implement actions required by the regulations. This action is covered by the Office of Management and Budget Clearance Number 3150-0011, which expires June 30, 1994.

Sincerely,

ORIGINAL SIGNED BY:

George Kalman, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Pressurized Thermal Shock Table
2. Upper-Shelf Energy Table
3. Nomenclature Key

cc w/enclosures:
See next page

DISTRIBUTION:
Docket File
NRC & Local PDRs
PD4-1 Reading
J. Roe
E. Adensam
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OGC
ACRS (10) (P-315)
A. B. Beach
D. McDonald
S. Sheng

OFC	LA:PD4-1	PM:PD4-1	D:PD4-1
NAME	PNoonan	GKalman:pk	WBeckner
DATE	5/4/94	5/5/94	5/15/94
COPY	YES/NO	YES/NO	YES/NO

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Project Directorate IV-1
Division of Reactor Projects - III/IV
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Arkansas Nuclear One, Unit 1

cc:

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Admiral Kinnaird R. McKee, USN (Ret)
214 South Morris Street
Oxford, Maryland 21654

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL/EFPY	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Arkansas 1 EOL: 5/20/2014	Nozzle Belt Forging	AYN 131	8.62E18	3°F	Generic	20	Table	0.03	0.70
	Upper Shell	C-5114-2	9.79E18	-10°F	Plant Specific	50.872	Calculated	0.15	0.52
	Upper Shell	C-5120-2	9.79E18	-10°F	Plant Specific	122.75	Table	0.17	0.55
	Lower Shell	C-5114-1	9.4E18	0°F	Plant Specific	50.872	Calculated	0.15	0.52
	Lower Shell	C-5120-1	9.4E18	-10°F	Plant Specific	122.75	Table	0.17	0.55
	Nozzle Belt/Upper Shell Circ. Weld WF-182-1	821T44	8.62E18	-5°F	Generic	162.09	Calculated	0.24	0.63
	Upper/Lower Shell Circ. Weld WF-112	406L44	9.4E18	-5°F	Generic	173.62	Calculated	0.31	0.59
	Upper Shell Axial Welds WF-18	8T1762	7.05E18	-5°F	Generic	152.25	Table	0.20	0.55
	Lower Shell Axial Welds WF-18	8T1762	6.95E18	-5°F	Generic	152.25	Table	0.20	0.55

Reference

Chemistry Factor for WF-182-1 weld was calculated from Davis-Besse surveillance data that was reported in BAW-1803, Rev. 1. The Davis-Besse surveillance weld was fabricated with the same heat number as WF-182-1.

Chemistry Factor for WF-112 weld was calculated from Oconee 1, Davis-Besse, ANO-1, and Rancho Seco surveillance data that was reported in BAW-1803, Rev. 1. These surveillance welds were fabricated with the same heat number as WF-112.

IRT_{ndt} for Nozzle Belt Forging is a mean value from 24 forgings similar to AYN 131. The data is reported in BAW-10046P and has a standard deviation of 31°F.

Fluence, IRT_{ndt}, and chemical composition data are from July 1, 1992, letter from J. J. Fisicaro (EO) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity"

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
Arkansas 1 EOL: 5/20/2014	Nozzle Belt Forging	AYN 131	A 508-2	55	5.18E18	66	Generic
	Upper Shell	C-5114-2	A 533B-1	68	5.88E18	86	65%
	Upper Shell	C-5120-2	A 533B-1	63	5.88E18	86	65%
	Bottom Shell	C-5114-1	A 533B-1	66	5.64E18	86	65%
	Bottom Shell	C-5120-1	A 533B-1	58	5.64E18	84	65%
	Nozzle Belt/Upper Shell Circ. Weld WF-182-1	821T44	Linde 80, SAW	EMA[2]	5.18E18	80	
	Upper/ Lower Shell Circ. Weld WF-112	406L44	Linde 80, SAW	EMA[2]	5.64E18	EMA[2]	
	Upper Shell Axial Welds WF-18	8T1762	Linde 80, SAW	EMA[2]	4.23E18	EMA[2]	
	Lower Shell Axial Welds WF-18	8T1762	Linde 80, SAW	EMA[2]	4.23E18	EMA[2]	

[2]Licensee must confirm applicability of Topical Reports BAW-2178P and BAW-2192P

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE

References

Fluence and chemical composition data are from July 1, 1992, letter from J. J. Fisicaro (EO) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity"

UUSE data for plates C5120-2, C5114-1, and C5114-2; and forging AYN 131 are from BAW-2075, which analyzed the results from capsule C. These values were confirmed to be from longitudinal specimens by letter dated October 29, 1993 (Response to GL 92-01 RAI).

Unirradiated USE value for Nozzle Belt Forging, AYN 131, was determined from data from similar forgings reported in BAW-1820 and an October 29, 1993 letter from Entergy to USNRC. The value is a lower tolerance limit with 95% confidence that at least 95% of the population is less than the tolerance limit

($TL = \overline{X} - K\sigma$, where $\overline{X} = 127$, $\sigma = 17.87$, $K = 3.399$)

The 1/4T USE at EOL for Nozzle Belt Forging, AYN 131, was calculated using the lower limit line (0.10% copper) in Figure 2 of RG 1.99, Rev. 2.

PRESSURIZED THERMAL SHOCK TABLES AND USE TABLES FOR ALL PWR PLANTS

NOMENCLATURE

Pressurized Thermal Shock Table

Column 1: Plant name and date of expiration of license.
Column 2: Beltline material location identification.
Column 3: Beltline material heat number; for some welds that a single-wire or tandem-wire process has been reported, (S) indicates single wire was used in the SAW process, (T) indicates tandem wire was used in the SAW process.
Column 4: End-of-life (EOL) neutron fluence at vessel inner wall; cited directly from inner diameter (ID) value or calculated by using Regulatory Guide (RG) 1.99, Revision 2, neutron fluence attenuation methodology from the quarter thickness (T/4) value reported in the latest submittal (GL 92-01, PTS, or P/T limits submittals).
Column 5: Unirradiated reference temperature.
Column 6: Method of determining unirradiated reference temperature (IRT).

Plant-Specific
This indicates that the IRT was determined from tests on material removed from the same heat of the beltline material.

MTEB 5-2
This indicates that the unirradiated reference temperature was determined from following MTEB 5-2 guidelines for cases where the IRT was not determined using American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, NB-2331, methodology.

Generic
This indicates that the unirradiated reference temperature was determined from the mean value of tests on material of similar types.

Column 7: Chemistry factor for irradiated reference temperature evaluation.
Column 8: Method of determining chemistry factor.

Table
This indicates that the chemistry factor was determined from the chemistry factor tables in RG 1.99, Revision 2.

Calculated
This indicates that the chemistry factor was determined from surveillance data via procedures described in RG 1.99, Revision 2.

Column 9: Copper content; cited directly from licensee value except when more than one value was reported. (Staff used the average value in the latter case.)

No Data
This indicates that no copper data has been reported and the default value in RG 1.99, Revision 2, will be used by the staff.

Column 10: Nickel content; cited directly from licensee value except when more than one value was reported. (Staff used the average value in the latter case.)

No Data

This indicates that no nickel data has been reported and the default value in RG 1.99, Revision 2, will be used by the staff.

Upper Shelf Energy Table

Column 1: Plant name and date of expiration of license.
Column 2: Beltline material location identification.
Column 3: Beltline material heat number; for some welds that a single-wire or tandem-wire process has been reported, (S) indicates single wire was used in the SAW process. (T) indicates tandem wire was used in the SAW process.
Column 4: Material type; plate types include A 533B-1, A 302B, A 302B Mod., and forging A 508-2; weld types include SAW welds using Linde 80, 0091, 124, 1092, ARCOS-B5 flux, Rotterdam welds using Graw Lo, SMIT 89, LW 320, and SAF 89 flux, and SMAW welds using no flux.
Column 5: EOL upper-shelf energy (USE) at T/4; calculated by using the EOL fluence and either the cooper value or the surveillance data. (Both methods are described in RG 1.99, Revision 2.)

EMA
This indicates that the USE issue may be covered by the approved equivalent margins analysis in a topical report.

Column 6: EOL neutron fluence at T/4 from vessel inner wall; cited directly from T/4 value or calculated by using RG 1.99, Revision 2, neutron fluence attenuation methodology from the ID value reported in the latest submittal (GL 92-01, PTS, or P/T limits submittals).

Column 7: Unirradiated USE.

EMA

This indicates that the USE issue may be covered by the approved equivalent margins analysis in a topical report.

Column 8: Method of determining unirradiated USE.

Direct

For plates, this indicates that the unirradiated USE was from a transverse specimen. For welds, this indicates that the unirradiated USE was from test date.

65%

This indicates that the unirradiated USE was 65% of the USE from a longitudinal specimen.

Generic

This indicates that the unirradiated USE was reported by the licensee from other plants with similar materials to the beltline material.

NRC generic

This indicates that the unirradiated USE was derived by the staff from other plants with similar materials to the beltline material.

10, 30, 40, or 50 °F

This indicates that the unirradiated USE was derived from Charpy test conducted at 10, 30, 40, or 50 °F.

Surv. Weld

This indicates that the unirradiated USE was from the surveillance weld having the same weld wire heat number.

Equiv. to Surv. Weld

This indicates that the unirradiated USE was from the surveillance weld having different weld wire heat number.

Sister Plant

This indicates that the unirradiated USE was derived by using the reported value from other plants with the same weld wire heat number.

Blank

Indicates that there is insufficient data to determine the unirradiated USE.