Response to NRC Generic Letter 92-01 for

Arkansas Nuclear One - Unit 2

A-MECH-ER-005, Rev. 00

June 23, 1992

Question 1

**Certain addressees are requested to provide the following information regarding Appendix H to 10 CFR Part 50:**

> **Addressees who do not have a surveillance program meeting ASTM E185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2 of GL 92-01), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).**

Response to Question 1

The Arkansas Nuclear One - Unit 2 (ANO-2) surveillance program was designed to meet the requirements of ASTM E185-73[1] and 10 CFR 50 Appendix H[2]. However, the vessel was fabricated to Section III of the ASME Code 1968 Edition through the Summer 1970 Addenda. Consequently, deviations to ASTM E185-73 and Appendix H were noted and identified to the NRC through licensing correspondence (Reference 3). The two specific exceptions were: (1) the plate selection for the surveillance program was based on longitudinal data, and (2) the surveillance program requires the attachment of the surveillance capsule to the cladding on the inside of the vessel in the beltline region. Following receipt and review, the NRC provided a favorable Safety Evaluation and a specific exemption to Appendix H was authorized (Reference 4) pursuant to 10 CFR 50.12 for these exceptions.

Question 2.a

**Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:**

> **Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50.**

Response to Question 2.a

The initial upper shelf energy, copper weight percent and fluence for each beltline location were evaluated and utilized to assess the change in upper shelf energy and predict end-of-life values.

The Charpy upper shelf energy for the ANO-2 reactor vessel beltline materials, as predicted in accordance with Regulatory Guide 1.99 Revision 2, does not fall below 50 ft-lbs at end of design life. Therefore, no further action is required.

Question 2.b

**Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the considerations given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:**

**(1) The results from all Charpy and drop weight tests for all unirradiated beltline materials, and the unirradiated reference temperature for each beltline material, and the method for determining the unirradiated reference temperature from the Charpy and drop weight test;**

**(2) The heat treatment received by all beltline and surveillance materials;**

**(3) The heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;**

**(4) the heat number for each surveillance plate or forging and heat number of wire and flux lot number used to fabricate the surveillance weld;**

**(5) the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material; and**

**(6) the heat number of the wire used for determining the weld and chemical composition if different than Item (3) above.**

Response to Question 2.b.1

Table 1 provides the unirradiated reference temperatures for the materials in the beltline region. As requested the basis for these values have been provided. In most instances this information summarizes previously docketed information.

Table 1

ANO-2 BELTLINE MATERIAL UNIRRADIATED REFERENCE TEMPERATURES

| Beltline Location | Drop Weight NDTT (°F) | Charpy Specimen Orientation | Initial $RT_{NDT}$ (°F) | Method |
|---|---|---|---|---|
| Plates | | | | |
| C-8009-1 | -30[3] | longitudinal[3] | -26[5] | MTEB 5.2 1.1(3)a[3,7] |
| C-8009-2 | 0[3] | longitudinal[3] | 0[5] | MTEB 5.2 1.1(3)a[3,7] |
| C-8009-3[1] | 0[6] | transverse[3] | 0[5][2] | NB-2331[3,8] |
| C-8010-1 | -20[3] | longitudinal[3] | +12[5] | MTEB 5.2 1.1(3)a[3,7] |
| C-8010-2 | -30[3] | longitudinal[3] | -28[5] | MTEB 5.2 1.1(3)a[3,7] |
| C-8010-3 | -30[3] | longitudinal[3] | -30[5] | MTEB 5.2 1.1(3)a[3,7] |
| Welds | | | | |
| 2-203 A,B,C | not determined | N/A | -56[5,9] | Generic[9] |
| 9-203 | -40[3] | N/A | -40[3] | NB-2331[8] |
| Surv. Weld (Same consumables as 9-203) | -10[6] | N/A | -10[6] | NB-2331[8] |
| 3-203 A,B,C | not determined | N/A | -56[5,9] | Generic[9] |

N/A - Not Applicable
1 - Surveillance Plate
2 - Based on Surveillance Plate Data

Response to Question 2.b.2

The heat treatment process received by test and surveillance program materials is summarized below:

1. 1600°F for 4 hours. Water Quenched. (Austenitizing)
2. 1225°F for 4 hours. (Tempering)
3. 1150°F for 40 hours. Furnace Cooled to 500°F. (Post-weld stress relief)

In addition, this heat treatment process which was applied to the test materials is representative of the heat treated condition of the reactor vessel.

Response to Questions 2.b.3, 2.b.4, 2.b.5

Table 2 provides the beltline material and surveillance material source information pertaining to the materials provided in response to question 2.b.1. This information has been extracted from previous submittals and provided in tabular form for convenience. A majority of the information was previously provided by Reference 3. Recently, Entergy Operations submitted a license amendment request (No. 124, Reference 5) to update the ANO-2 operating limits and also provided, as required by the NRC Safety Evaluation for the ANO-2 PTS Evaluation (dated July 20, 1987), projected end-of-life adjusted reference temperatures in accordance with 10 CFR §50.61 which utilized the best available information included below.

Response to Question 2.b.6

The wire heat used for determining the weld and chemical composition is the same as was identified in response to 2.b.3; therefore, this question is not applicable to ANO-2.

Question 3

**Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:**

**a. How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525°F were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy.**

**b. How their surveillance results on the predicted amount of embrittlement were considered.**

**c. If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.**

Table 2

ANO-2 BELTLINE MATERIAL SOURCE INFORMATION

| Beltline Location | Plate/Weld Wire Heat No. | Flux Type/Lot No. | Cu (w%) | Ni (w%) | P (w%) | S (w%) |
|---|---|---|---|---|---|---|
| Plates | | | | | | |
| C-8009-1 | C 8161-3 | N/A | 0.12 | 0.63 | 0.010 | 0.014 |
| C-8009-2 | C 8161-1 | N/A | 0.08 | 0.59 | 0.009 | 0.011 |
| C-8009-3 | C 8182-2 | N/A | 0.08 | 0.60 | 0.009 | 0.011 |
| C-8010-1 | C 8161-2 | N/A | 0.08 | 0.59 | 0.006 | 0.008 |
| C-8010-2 | B 2545-1 | N/A | 0.07 | 0.66 | 0.003 | 0.008 |
| C-8010-3 | B 2545-2 | N/A | 0.07 | 0.65 | 0.003 | 0.007 |
| Welds | | | | | | |
| 2-203 A,B,C | 10120 | Linde 0091/3999 | 0.05 | 0.18 | NA | NA |
| 9-203 | 83650 | Linde 0091/1122 | 0.05 | 0.08 | NA | NA |
| Surveillance Weld | 83650 | Linde 0091/1122 | 0.04 | 0.08 | 0.004 | 0.009 |
| 3-203 A,B,C | 10120 | Linde 0091/3458 | 0.05 | 0.18 | NA | NA |

N/A - Not Applicable
NA - Not Available (As deposited information was not taken. Previously reported information was based on weld wire heat chemistry.)

Response to Question 3.a

In response to Generic Letter 88-11, Arkansas Nuclear One - Unit 2 did not consider the temperature effects of operating at an irradiation temperature below 525°F. ANO-2 is precluded from operating below 525°F by the Technical Specifications and operational procedures.

ANO-2 Technical Specification 3.1.1.5, "Minimum Temperature for Criticality," dictates that the lowest operating loop temperature ($T_{avg}$) shall be $\geq$ 525°F when the reactor is critical. If this condition is not met and cannot be restored within 15 minutes, the reactor is placed in hot standby within the next 15 minutes. In addition, a corresponding surveillance requirement delineates that when the reactor is critical and $T_{avg}$ is less than 535°F, the reactor coolant system (RCS) temperature is to be determined at least once per thirty minutes. To date, the limiting condition Technical Specification 3.1.1.5 has only been exceeded once. This occurred when the unit was operating at approximately 8% power and a relief valve lifted causing a cooldown. The reactor was tripped at 524°F and brought to hot standby conditions as required.

Technical Specification, 3.2.6, "Reactor Coolant Cold Leg Temperature," further limits the cold leg temperature above 30% of rated thermal power. This specification delineates that the cold leg temperature be maintained between 542°F and 554.7°F. Should the temperature be outside these limits, restoration is required within 2 hours or a reduction in thermal power to less than 30% of rated thermal power is dictated in the following 4 hours. To date, no evidence has been found that indicates this specification has ever been exceeded such that a power reduction has been necessary.

In addition to the Technical Specification requirements, operating procedures dating back to initial startup direct the operating temperatures to be above 525°F. Initial conditions for criticality are established with a RCS temperature of 545° $\pm$ 5°F. Power escalation is then directed to proceed with a linear increase in $T_{cold}$ from 545°F to 553°F corresponding to reactor powers of 0 to 100% (Figure 1). Reactor power decreases are also conducted utilizing the same guidance. In the event of a reactor trip, the Steam Dump and Bypass Control System effects a smooth transition to hot zero-power conditions, and automatically controls the steam pressure and thus RCS temperature at the hot-zero power value (545°F).

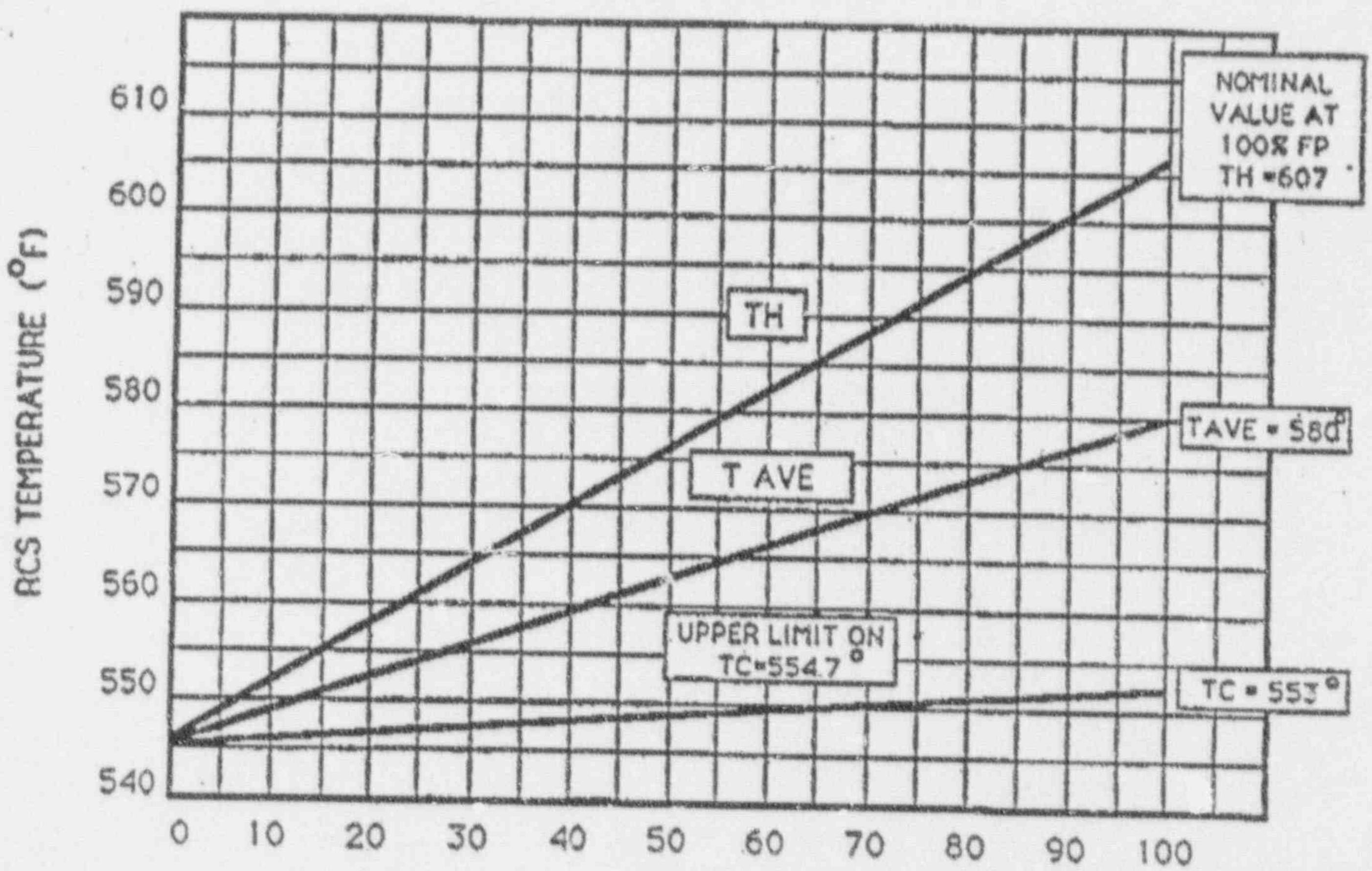
In conclusion, core critical operation of ANO-2 is not conducted at temperatures below 525°F and question 3.a is not applicable.

Response to Question 3.b

To date, one surveillance capsule has been removed and evaluated from the ANO-2 vessel. The results of this capsule evaluation were provided in Reference 10. The evaluation results have established the fluence and this information has been considered in a conservative fashion when developing normal operation limits and projected $RT_{PTS}$ values provided by Reference 5. Since only one capsule has been removed and analyzed, credible surveillance measurements are not yet available for application using Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2.

# FIGURE 1

RCS TEMPERATURE v.s. REACTOR POWER LEVEL



Response to Question 3.c

A review of the ANO-2 surveillance material test report (Reference 10) and comparison with the required shift predictions methods for $\Delta RT_{NDT}$ (shift in Nil-Ductility Transition Reference Temperature caused by irradiation) and decrease USE given by Regulatory Guide 1.99 Revision 2 was performed.

The $\Delta RT_{NDT}$ was determined using mean estimates of the material properties for both plate and weld and compared with the test measurements. The results are shown in the following table.

$\Delta RT_{NDT}$ for ANO-2 Surveillance Materials
Predicted vs. Measured

| | Plate C-8009-3 | Weld 9-203 |
|---|---|---|
| Predicted $\Delta RT_{NDT}$ (°F) | 36 | 25 |
| Measured $\Delta RT_{NDT}$ (°F) | 50 (Transverse)<br>21 (Longitudinal) | 10 |

Based on the preceding results, the general conclusion is that no anomalies exist and the predicted $\Delta RT_{NDT}$ is within one standard deviation for both plate and weld materials.

The decrease in USE was predicted in using the mean material properties for both plate and weld and compared with measured values. The results which follow show that the Regulatory Guide predictions are conservative for the ANO-2 materials.

Decrease in USE for ANO-2 Surveillance Materials
Predicted vs. Measured

| | Plate C-8009-3 | Weld 9-203 |
|---|---|---|
| Predicted $\Delta$USE (ft-lb) | 15% | 15% |
| Measured $\Delta$USE (ft-lb) | 1.2% (Transverse)<br>8.4% (Longitudinal) | 2.7% |

In conclusion, the data from the Arkansas Nuclear One - Unit 2 surveillance program have been found to be predictable or conservative when compared to the methodology of Regulatory Guide 1.99 Revision 2 and no shift in reference temperature has exceeded the mean-plus-two standard deviation method intrinsic in the Regulatory Guide. The Regulatory Guide conservatively predicts the decrease in USE for the ANO-2 beltline materials.

References


1. ASTM E185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels".

2. 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements.

3. D. H. Williams (AP&L) to J. F. Stolz (NRC), Pressure Vessel Fracture Toughness Properties, Letter No. 2-068-10 (2CAN067810), dated June 1[illegible], 1978.

4. R. S. Boyd (NRC) to W. Cavanaugh III (AP&L), Issuance of Amendment 1 to Facility Operating License No. NPF-6 (Arkansas Nuclear One, Unit 2) Letter No. 2N-78-121 (2CNA097801), dated September 1, 1978.

5. J. W. Yelverton (Entergy) to U.S. NRC, Proposed Change to Technical Specification Pressure/Temperature Limits, Letter No. 2CAN069109, dated June 18, 1991.

6. Report No. TR-MCD-002, Arkansas Nuclear One - Unit 2, Evaluation of Baseline Specimens, Combustion Engineering, Inc., dated March 1976.

7. Branch Technical Position - MTEB 5-2, Fracture Toughness Requirements (Part of NUREG-75/087), dated 1975.

8. ASME Boiler and Pressure Vessel Code, Section III, Summer 1972 Addenda.

9. Report CEN-189, Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCA's with Loss of Feedwater for the Combustion Engineering NSSS, Combustion Engineering, Inc., December 1981.

10. J. T. Enos (AP&L) to J. R. Miller (NRC), Reactor Vessel Surveillance Capsule Summary Report, Letter No. 2CAN028503, dated February 8, 1985.