March 14, 1997

Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ISSUANCE OF AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE NO. DPR-51 - ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. M97529)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment No. 188 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications in response to your application dated November 26, 1996 as supplemented by letters dated December 17, 1996, March 4, 1997, and March 10, 1997.

The amendment changes the reactor coolant system pressure/temperature limits to incorporate updated parameters and requirements.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
ORIGINAL SIGNED BY:
George Kalman, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 188 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

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

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

ENTERGY OPERATIONS INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated November 26, 1996 as supplemented on December 17, 1996, March 4, 1997 and March 10, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 188 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Kalman

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

Attachment: Changes to the Technical Specifications

Date of Issuance: March 14, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

| REMOVE PAGES | INSERT PAGES |
|---|---|
| 18a | 18a |
| 19 | 19 |
| 20 | 20 |
| 20a | 20a |
| 20b | 20b |
| 20c | 20c |

3.1.2.7 Prior to reaching thirty one effective full power years of operation, Figures 3.1.2-1, 3.1.2-2 and 3.1.2-3 shall be updated for the next service period in accordance with 10CFR50, Appendix G. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543(5). The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.8. The provisions of Specification 3.0.3 are not applicable.

3.1.2.8 The updated proposed technical specifications referred to in 3.1.2.7 shall be submitted for NRC review at least 90 days prior to the end of the service period.

3.1.2.9 With the exception of ASME Section XI testing and when the core flood tank is depressurized, during a plant cooldown the core flood tank discharge valves shall be closed and the circuit breakers for the motor operators opened before depressurizing the reactor coolant system below 600 psig.

3.1.2.10 With the exception of ASME Section XI testing, fill and vent of the reactor coolant system, emergency RCS makeup and to allow maintenance of the valves, when the reactor coolant temperature is less than 262°F, the High Pressure Injection motor operated valves shall be closed with their opening control circuits for the motor operators disabled.

3.1.2.11 The plant shall not be operated in a water solid condition when the RCS pressure boundary is intact except as allowed by Emergency Operating Procedures and during System Hydrotest.

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.[1] These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation.[2] The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the DTT.[3]

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 [4]. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543. [5] The chemical composition of the limiting weld material is reported in the B&W Report, BAW-2121P [6]. The effect of neutron irradiation on the $RT_{NDT}$ of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 [7].

Figures 3.1.2-1, 3.1.2-2, and 3.1.2-3 present the pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown respectively. The limit curves are applicable through the thirty first effective full power year of operation. The service period was reduced by one effective full power year from that assumed in FTI Document 77-1258569-01 to be conservative with respect to independent calculations performed by the NRC staff. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all allowed operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-2 for reactor criticality and on Figure 3.1.2-1 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in $RT_{NDT}$ of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.

Specification 3.1.2.9 is to ensure that the core flood tanks are not the source for pressurizing the reactor coolant system when in cold shutdown.

Specification 3.1.2.10 is to ensure that high pressure injection is not the source of pressurizing the reactor coolant system when in cold shutdown. The LTOP enable temperature has been calculated in accordance with Code Case N-514. Instrument error is not included in the reactor coolant temperature of 262°F.

Specification 3.1.2.11 is to ensure that the reactor coolant system is not operated in a manner which would allow overpressurization due to a temperature transient.

REFERENCES

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.11.5

(4) FTI Document Number 77-1258569-01

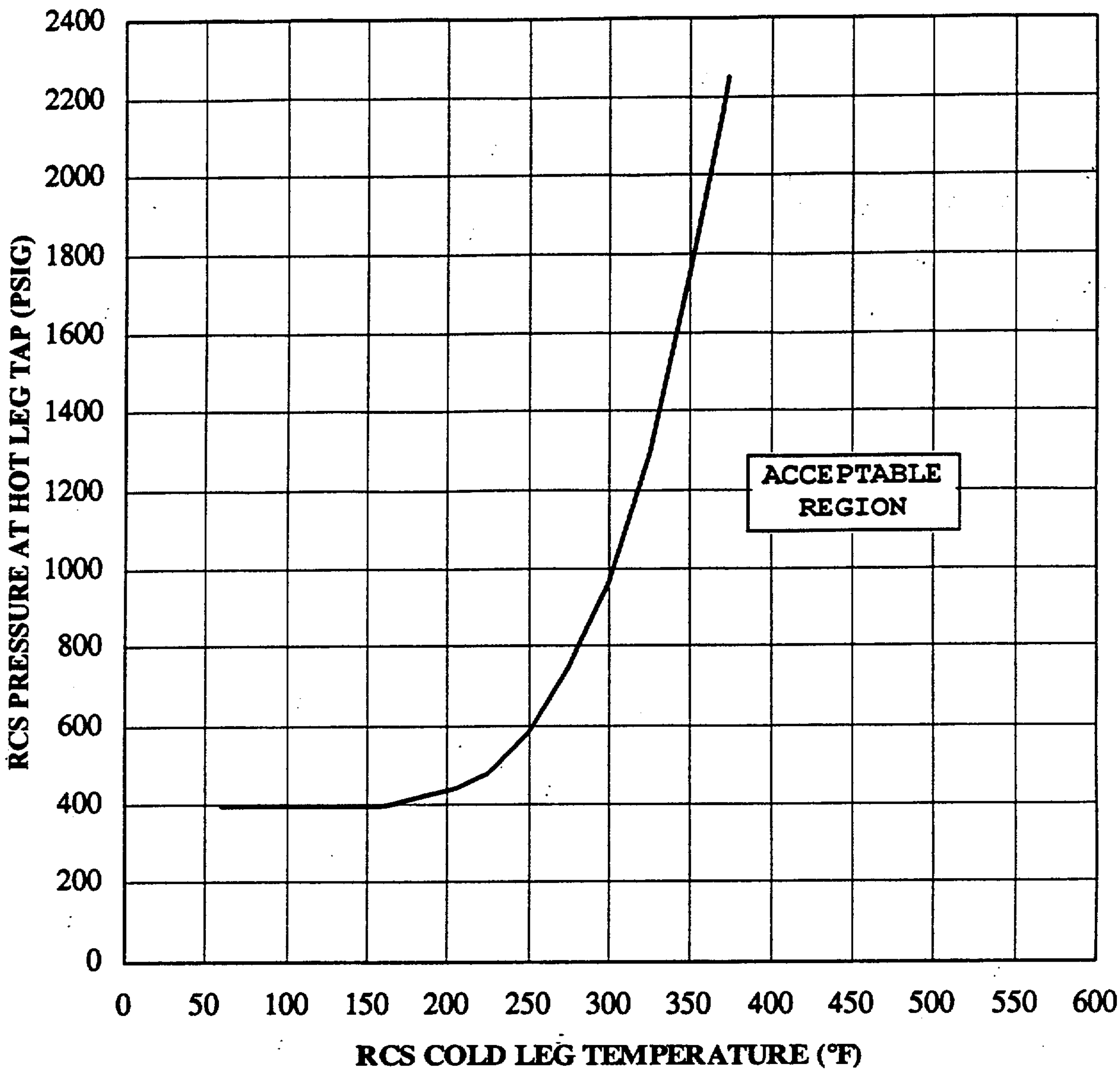
(5) BAW-1543, latest revision

(6) BAW-2121P

(7) FTI Calculation Numbers 32-1245917-00 and 32-1257716-00

**FIGURE 3.1.2-1**

**RCS INSERVICE HYDROSTATIC TEST H/U & C/D LIMITS TO 31 EFPY**



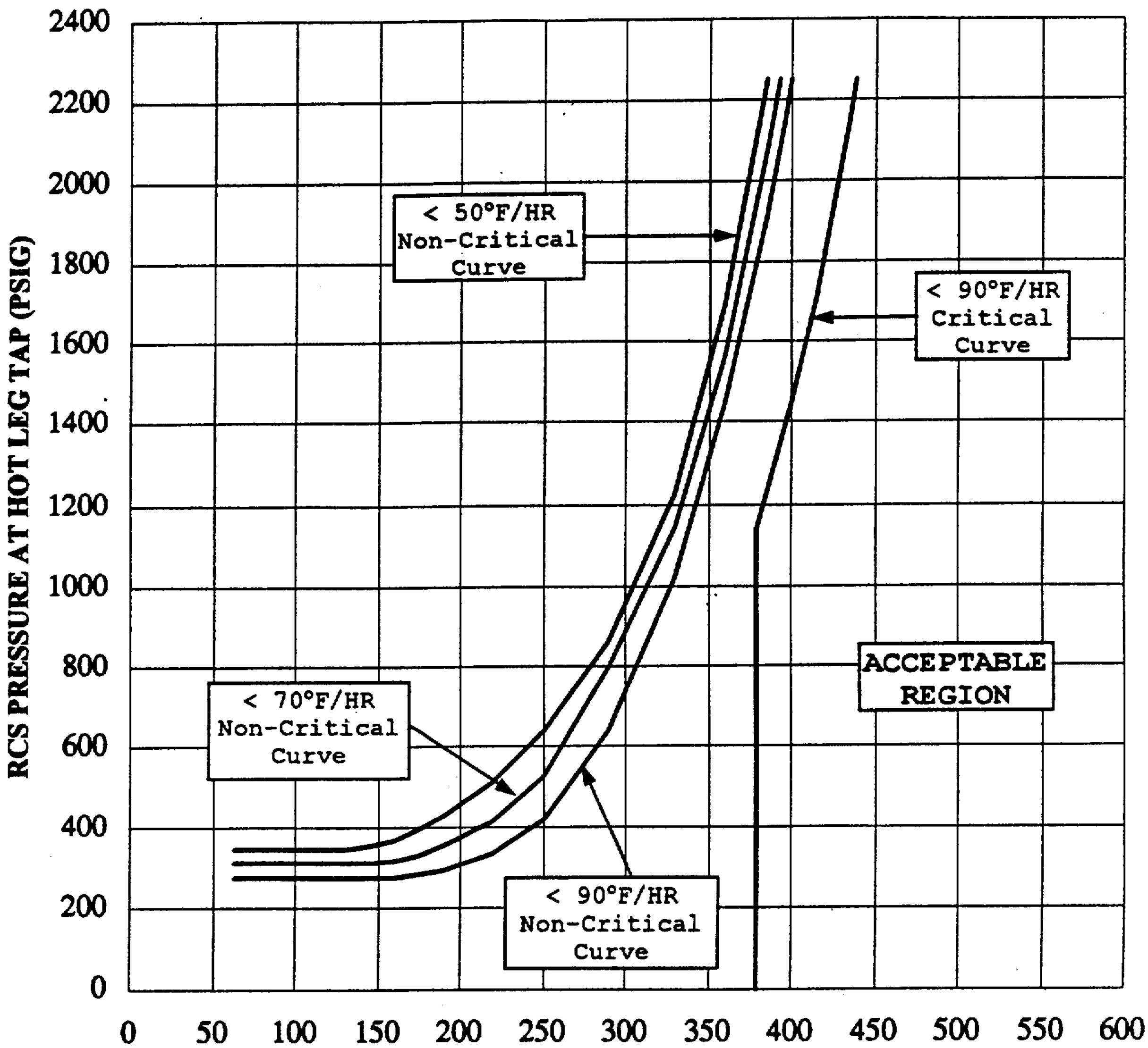

Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.

2. All Notes on Figure 3.1.2-2 are applicable for heatups. This curve is based on a heatup rate of < 90°F/HR.

3. All Notes on Figure 3.1.2-3 are applicable for cooldowns.

**FIGURE 3.1.2-2**
**RCS HEATUP LIMITATIONS TO 31 EFPY**




Notes:

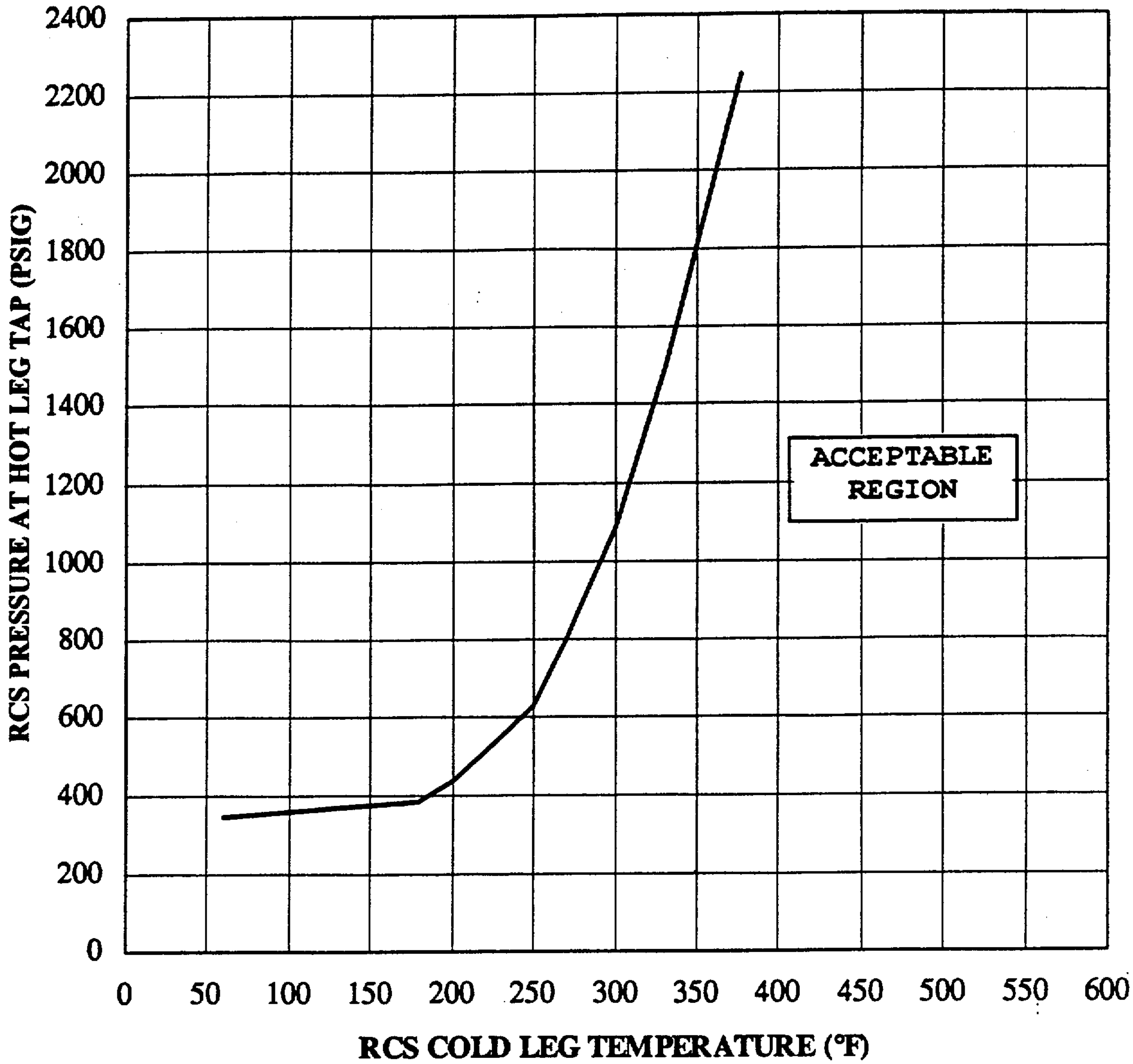
1. These curves are not adjusted for instrument error and shall not be used for operation.
2. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

| RCS TEMP | RCP RESTRICTIONS |
|---|---|
| T > 300°F | None |
| 300°F ≥ T ≥ 225°F | ≤ 3 |
| 225°F > T ≥ 84°F | ≤ 2 |
| T < 84°F | No RCPs operating |

4. Allowable Heatup Rates:

| RCS TEMP | H/U RATE |
|---|---|
| 60°F < T ≤ 84°F | ≤ 15°F/HR |
| T > 84°F | As allowed by applicable curve |

**FIGURE 3.1.2-3**
**RCS COOLDOWN LIMITS TO 31 EFPY**




Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. A maximum step temperature change of 25°F is allowable when securing all RCPs with the DHR system in operation. This change is defined as the RCS temperature prior to securing all the RCPs minus the DHR return temperature after the RCPs are secured. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

| RCS TEMP | RCP RESTRICTIONS |
|---|---|
| T > 255°F | None |
| 150°F ≤ T ≤ 255°F | ≤ 2 (See Note 5) |
| T < 150°F | No RCPs operating |

4. Allowable Cooldown Rates:

| RCS TEMP | C/D RATE | STEP CHANGE |
|---|---|---|
| T ≥ 280°F | 100°F/HR | ≤ 50°F in any 1/2 HR |
| 280°F > T ≥ 150°F | 50°F/HR (See Note 5) | ≤ 25°F in any 1/2 HR |
| T < 150°F | 25°F/HR | ≤ 25°F in any 1 HR |

5. If RCPs are operated < 200°F, then the RCS cooldown rate from 150°F ≤ T ≤ 180°F is reduced to 30°F in 15 hours.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 188 TO

FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

## 1.0 INTRODUCTION

By letter dated November 26, 1996, Entergy Operations, Inc. (the licensee) submitted a request to amend the pressure-temperature (P-T) limit curves in the Technical Specifications (TSs) for Arkansas Nuclear One, Unit 1 (ANO-1). Additional information was submitted by letters dated December 17, 1996 and March 4, 1997. The current P-T limit curves are valid for a service period of 15 effective full power years (EFPY). The current service period will end in March 1997. The amendment was intended to extend the validity of the ANO-1 P-T limit curves to 32 EFPY.

However, based on an initial NRC staff review of the submittals, a discrepancy was identified in the methodology used by the licensee for determining the standard deviation when calculating the reference temperature. To incorporate the more conservative conclusions that resulted from using the NRC staff proposed standard deviation, the licensee revised the TS amendment request to decrease the validity of the P-T curves from 32 EFPY to 31 EFPY. The revised amendment request, changing the curve validity to 31 EFPY, was transmitted by letter dated March 10, 1997.

The licensee's letters sent subsequent to the November 26, 1996, request to amend the P-T curves were letters transmitting clarifying information and details related to the methodologies for performing calculations. They did not change the initial proposed no significant hazards determination.

In conjunction with the requested amendment, the licensee requested an exemption from certain requirements of 10 CFR 50.60. The requested exemption was granted on March 12, 1997. The exemption permits the licensee to use the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Case N-514 to determine the safety margins associated with low temperature overpressure protection in lieu of the safety margins required by 10 CFR Part 50, Appendix G.

The staff evaluates the P-T limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1 (Rev. 1); GL 92-01, Rev. 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and Standard Review Plan (SRP), Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2 to review P-T limit curves. RG 1.99, Rev. 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves, and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME) Code.

SRP 5.3.2 provides an acceptable method of calculating the P-T limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor $K_I$, which is a function of the stress state and flaw configuration. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or $RT_{NDT}$). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial $RT_{NDT}$), the mean value of the adjustment in reference temperature caused by irradiation ($\Delta RT_{NDT}$), and a margin (M) term. The $\Delta RT_{NDT}$ is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2 or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial $RT_{NDT}$ is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2 or surveillance data. The margin term is used to account for uncertainties in the values of initial $RT_{NDT}$, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2 describes the methodology to be used in calculating the margin term.

## 2.0 EVALUATION

### 2.1 LICENSEE EVALUATION OF P-T CURVES

The licensee's P-T limit curves were calculated using an ART of 212°F for the 1/4T location for the limiting RPV beltline material, the upper to lower shell circumferential weld WF-112. Weld WF-112 was fabricated by Babcock & Wilcox (B&W) using a submerged arc process, Linde 80 flux and with heat number 406L44 weld wire. The ART was the sum of an initial $RT_{NDT}$ of -5°F, a margin value of 67°F and a $\Delta RT_{NDT}$ of 150°F. The initial $RT_{NDT}$ is a generic value for B&W fabricated submerged arc welds with Linde 80 flux. (This value has a standard deviation of 19.7°F). The $\Delta RT_{NDT}$ was calculated using surveillance data from the B&W Owners Group (B&WOG) Integrated Surveillance Program. The margin term was calculated using a standard deviation for the initial $RT_{NDT}$ of 19.7°F and a standard deviation for the $\Delta RT_{NDT}$ of 27°F.

The surveillance data used to calculate the $\Delta RT_{NDT}$ were from surveillance weld materials irradiated in surveillance capsules of ANO-1, Oconee-1, Rancho Seco, B&WOG and Point Beach-2. All these surveillance welds were fabricated by B&W using the submerged arc process, Linde 80 flux, and using the same heat number of weld wire as used in the limiting weld in the ANO-1 RPV beltline. The reported copper content for the surveillance welds is 0.28 wt% for the ANO-1 weld, 0.32 wt% for the Oconee-1 and B&WOG welds, 0.31 wt% for the Rancho Seco weld, and 0.25 wt% for the Point Beach-2 weld. All the surveillance welds contained 0.59 wt% Ni. The licensee indicates that weld WF-112 in the ANO-1 reactor vessel beltline has a best estimate chemistry of 0.31 wt% Cu and 0.59 wt% Ni. The licensee calculated the chemistry factor from the surveillance weld data using the ratio procedure specified in RG 1.99, Revision 2, Position 2.1. This procedure specifies that the measured values of $\Delta RT_{NDT}$ be adjusted by multiplying the values by the ratio of the chemistry factor for the RPV weld to that for the surveillance welds. Then, using the adjusted $\Delta RT_{NDT}$ values and their corresponding fluence, the chemistry factor was calculated by multiplying each adjusted $\Delta RT_{NDT}$ by the corresponding fluence factor, summing the products, and dividing by the sum of the squares of the fluence factors. Using these calculated/normalized data, the chemistry factor for the limiting ANO-1 RPV weld was calculated to be 185.6°F. The licensee determined that the standard deviation of the difference between the adjusted $\Delta RT_{NDT}$ data and the curve representing the best fit of the adjusted data was 26.9°F (27°F rounded to the nearest °F). The licensee used this value of standard deviation in its calculation of the margin term. The margin term is 67°F when it is calculated using the methodology in RG 1.99, Rev. 2, a standard deviation for the initial $RT_{NDT}$ of 19.7°F, and a standard deviation for the $\Delta RT_{NDT}$ of 27°F.

## 2.2 STAFF EVALUATION OF P-T CURVES

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the reactor vessel of ANO-1. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff confirmed that the material with the highest ART at the 1/4T location at 31 EFPY for ANO-1 is the upper shell to lower shell circumferential weld WF-112. To calculate the ART at the 1/4T location, the staff used the same values for initial $RT_{NDT}$ and $\Delta RT_{NDT}$ as the licensee; but used a margin value of 68.5°F. This value of margin is calculated using the methodology in RG 1.99, Rev. 2, a standard deviation for the initial $RT_{NDT}$ of 19.7°F and a standard deviation for the $\Delta RT_{NDT}$ of 28°F.

RG 1.99, Rev. 2 indicates that the standard deviation for the $\Delta RT_{NDT}$ is 28°F when surveillance data are not credible. RG 1.99, Rev. 2 permits the standard deviation for the $\Delta RT_{NDT}$ to be reduced to 14°F when data are credible. If data are to be considered credible, RG 1.99, Rev. 2 indicates that the scatter of the $\Delta RT_{NDT}$ values about the best-fit line, as described in Regulatory Position 2.1, should normally be less than 28°F for welds. There are 14 data points from the B&WOG Integrated Surveillance Program for welds that were fabricated using heat number 406L44 weld wire. The difference between the adjusted measured data and the curve representing the best fit of the adjusted data exceeded 28°F for four of the adjusted data points. Hence, in accordance with RG 1.99, Rev. 2, the data is not credible and the standard deviation for the $\Delta RT_{NDT}$ should be 28°F.

The staff does not believe the standard deviation for the $\Delta RT_{NDT}$ that is recommended in RG 1.99, Rev. 2 should be reduced based on the small number of data points (14) in the licensee's evaluation. The standard deviation in RG 1.99, Rev. 2 is a more meaningful value, since it was based on analysis of all surveillance weld data available at the time of the development of the RG.

Therefore, the staff has determined that by summing the initial $RT_{NDT}$ of -5°F, the margin value of 68.5°F and the $\Delta RT_{NDT}$ of 150°F, the ART should be 213.5°F. Substituting the ART of 213.5°F for ANO-1 into equations in SRP 5.3.2, the staff determined that the proposed P-T limits for heatup, cooldown, hydrotest, and criticality meet the beltline material requirements in Appendix G of 10 CFR Part 50 for 31 EFPY.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P-T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 60°F for ANO-1, the staff has determined that the proposed P-T limits satisfy the requirements in Section IV.A.2 of Appendix G.

## 2.3 EVALUATION OF FLUENCE

The proposed values of the fluence at the inside surface of the pressure vessel were developed by Framatome Technologies Incorporated (FTI). The methodology is based on radiation transport calculations for which the uncertainty range was estimated experimentally. Plant specific measurements were used only to evaluate the range of uncertainty rather than to adjust the results of plant specific calculations. However, a benchmarking experiment was used to adjust the energy group values in the benchmarking calculation. These adjustments relate to the methodology and are: plant independent, dosimeter location independent and dosimeter type independent.

The calculational methodology included the following features:
**Neutron Sources:** Power distribution was included pin by pin and the effect of burnup on the neutron sources was calculated.
**Geometrical Model:** Neutron transport was performed using two dimensional discrete ordinates transport in $(r,\theta)$ and $(r,z)$ geometry.
**Macroscopic Cross Sections:** The BUGLE-93 cross section library was used which is based on the ENDF/B-VI data.
**Two Dimensional Transport:** The DORT code was used with a $P_3$ Legendre polynomial expansion for scattering and an $S_8$ for the quadrature expansion. A total of 48 directions were used in a 1/8 core configuration.
**C/M Ratios:** Comparison of calculated to measured (C/M) quantities was made on dosimeter activities. The fluence was judged to be correct based on statistical comparisons of the measured dosimeter activities to the corresponding calculated activities.
**Three Dimensional Results:** The $(r,\theta)$ and $(r,z)$ results were synthesized to produce an $(r,\theta,z)$ distribution.
**Best Estimate Fluence:** A calculational bias was determined using a statistical combination of the calculated dosimeter activities and the corresponding measured activities. The bias was given in terms of energy group adjustment constants, but the overall effect is about 5%. For ANO-1 the results were compared to the benchmark bias and it was found that there was no significant bias associated with this analysis beyond that identified in the cavity dosimetry program.

We find the methodology described above, the results of the benchmarking and the ANO-1 results to be acceptable, because they conform to the staff's recommendations for fluence calculation which are included in the draft Regulatory Guide on pressure vessel fluence.

## 3.0 TECHNICAL CONCLUSION

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality satisfy the requirements in Appendix G to Section XI of the ASME Code and Appendix G of 10 CFR Part 50 for 31 EFPY. The proposed P-T limits also satisfy GL 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. Hence, the proposed P-T limits may be incorporated into the ANO-1 TSs.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 4346). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 7.0 REFERENCES

(1) Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.

(2) NUREG-0800, Standard Review Plan, Section 5.3.2: "Pressure-Temperature Limits."

(3) Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements."

(4) Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," July 12, 1988.

(5) ASME Boiler and Pressure Vessel Code, Section XI, Appendix G for Nuclear Power Plant Components, Division 1, "Protection Against Non-ductile Failure."

(6) November 26, 1996, letter from C. Randy Hutchinson, (Entergy Operations, Inc.) to USNRC Document Control Desk, subject: "Arkansas Nuclear One, Unit 1 -Proposed Technical Specification Change To The Reactor Coolant System Pressure And Temperature Curves."

(7) December 17, 1996, letter from D. C. Mims, (Entergy Operations, Inc.) to USNRC Document Control Desk, subject: "Arkansas Nuclear One, Unit 1 - Additional Calculations Supporting Unit 1 Pressure and Temperature Limit TS Change Request."

(8) March 10, 1997, letter from D. C. Mims, (Entergy Operations, Inc.) to USNRC Document Control Desk, subject: "Arkansas Nuclear One, Unit 1 - Technical Specification Change To The Reactor Coolant System Pressure And Temperature Curves."

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