DOCKET FILE

Docket No. 50-313

November 14, 1991

Mr. Neil S. Carns
Vice President, Operations ANO
Entergy Operations, Inc.
Route 3 Box 137G
Russellville, Arkansas 72801

Dear Mr. Carns:

SUBJECT: ISSUANCE OF AMENDMENT NO. 154 TO FACILITY OPERATING LICENSE NO. DPR-51 - ARKANSAS NUCLEAR ONE, UNIT NO. 1 (TAC NO. M77665)

The Commission has issued the enclosed Amendment No.154to 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 (TSs) in response to your application dated September 20, 1990, as supplemented by letters dated February 28, and August 14, 1991.

The amendment revises the reactor coolant system TS pressure/temperature operating limits for the first 15 effective full power years, using the methodology of Regulatory Guide 1.99, Revision 2. The proposed amendment also revises the low-temperature overprotection (LTOP) enable temperature.

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.

Also note that the staff requests a response, within 60 days of your receipt of this letter, regarding your long-term plans for satisfying the requirements of Appendix G of 10 CFR Part 50. This request is discussed in Section 2.2 of the enclosed Safety Evaluation.

This requirement affects fewer than 10 respondents and is, therefore, not subject to Office of Management and Budget review under Public Law 96-511.

Sincerely,

/s/

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 154 to DPR-51
2. Safety Evaluation

cc w/enclosures:
See next page

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

November 14, 1991

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Entergy Operations, Inc.
Route 3 Box 137G
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Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 154 to DPR-51
2. Safety Evaluation

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See next page

Mr. Neil S. Carns
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Arkansas Nuclear One, Unit 1

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

ENTERGY OPERATIONS INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
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 September 20, 1990, as supplemented by letters dated February 28, and August 14, 1991, 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. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 154 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul W. O'Connor

for John T. Larkins, Director
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 14, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 154

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Revise 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
18	18
18a	18a
19	19
20	20
20a	20a
20b	20b
20c	20c

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.1.2.1 Hydro Tests

For thermal steady state system hydro tests, the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core, under the provisions of 3.1.2.3, and to ASME Code limits when no fuel assemblies are present provided the reactor coolant system limits are to the right of and below the limit line in Figure 3.1.2-1. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.1.2.2 Leak Tests

Leak tests required by Specification 4.3 shall be conducted under the provision of 3.1.2.3. The provisions of Specification 3.0.3 are not applicable.

3.1.2.3 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-2 and Figure 3.1.2-3, and are as follows:

Heatup:

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-2. The heatup rates shall not exceed those shown in Figure 3.1.2-2.

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the right of and below the limit line in Figure 3.1.2-3. Cooldown rates shall not exceed those shown in Figure 3.1.2-3.

3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100F.

3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430F.

3.1.2.6 With the limits of Specifications 3.1.2.3 or 3.1.2.4 or 3.1.2.5 exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg to less than 200F, while maintaining RCS temperature and pressure below the curve, within the following 30 hours.

3.1.2.7 Prior to reaching fifteen 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, Section V.B. 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 Specifications 3.0.3 and 3.0.4 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. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR Part 50, Appendix G. Section V.C.

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, and to allow maintenance of the valves, when the reactor coolant temperature is less than 300°F the four 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.

BASES

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 100F 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 BAW-2106[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-1511P.[6] The effect of neutron irradiation on the RT_{NDT} of the limiting weld material is reported in the B&W Report, BAW 2075[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 fifteenth effective full power year of operation. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all 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.

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

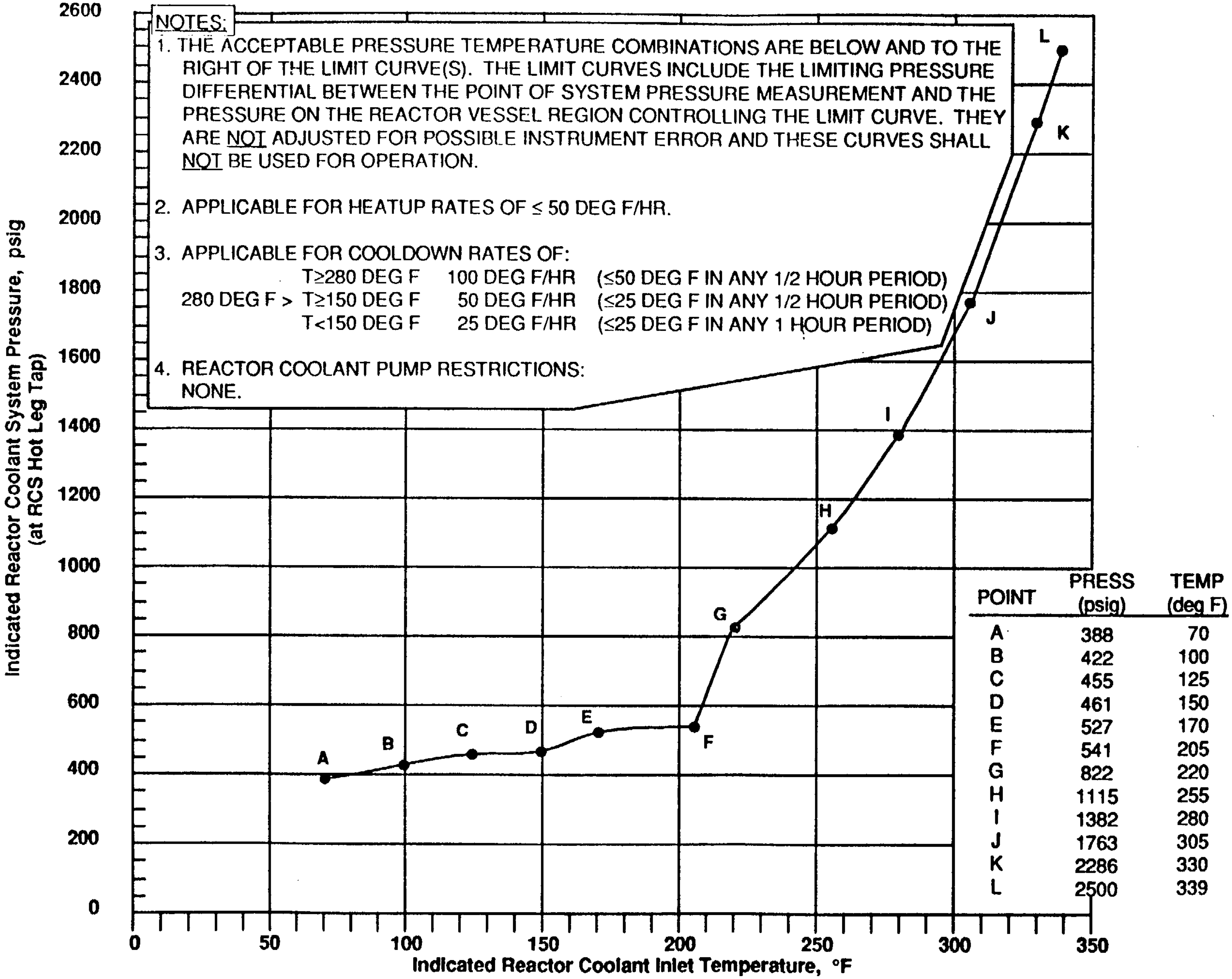
(4) BAW-2106

(5) BAW-1543, latest revision

(6) BAW-1511P

(7) BAW-2075, Revision 1

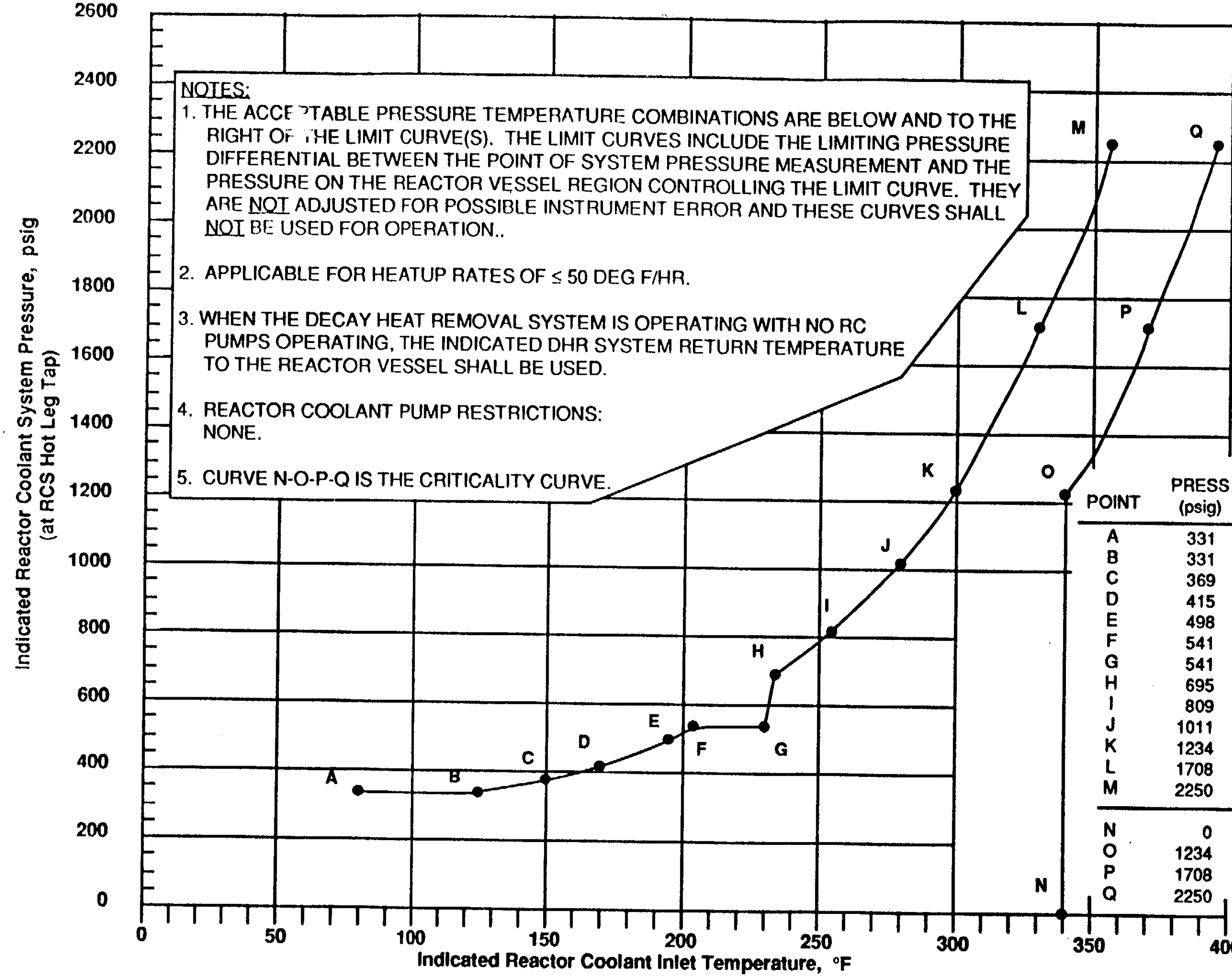
Amendment No 27, 82, 154



POINT	PRESS (psig)	TEMP (deg F)
A	388	70
B	422	100
C	455	125
D	461	150
E	527	170
F	541	205
G	822	220
H	1115	255
I	1382	280
J	1763	305
K	2286	330
L	2500	339

REACTOR COOLANT SYSTEM INSERVICE HYDROSTATIC TEST HEATUP AND COOLDOWN LIMITATIONS APPLICABLE FOR FIRST 15.0 EFFECTIVE FULL POWER YEARS

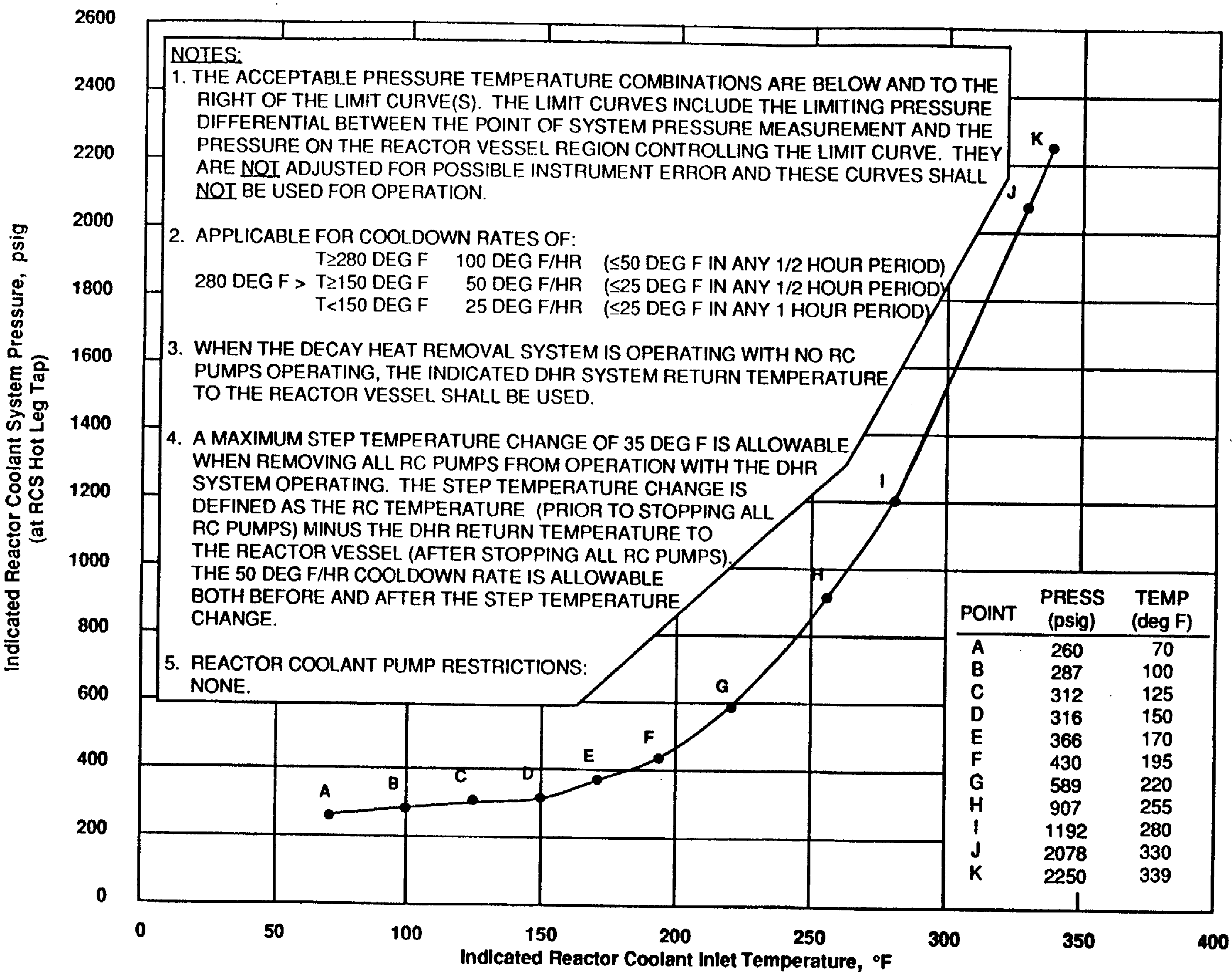
Figure 3.1.2-1



POINT	PRESS (psig)	TEMP (deg F)
A	331	70
B	331	125
C	369	150
D	415	170
E	498	195
F	541	204
G	541	230
H	695	234
I	809	255
J	1011	280
K	1234	300
L	1708	330
M	2250	355
N	0	340
O	1234	340
P	1708	370
Q	2250	395

REACTOR COOLANT SYSTEM NORMAL OPERATION - HEATUP LIMITATIONS APPLICABLE FOR FIRST 15.0 EFFECTIVE FULL POWER YEARS

Figure 3.1.2-2



POINT	PRESS (psig)	TEMP (deg F)
A	260	70
B	287	100
C	312	125
D	316	150
E	366	170
F	430	195
G	589	220
H	907	255
I	1192	280
J	2078	330
K	2250	339

REACTOR COOLANT SYSTEM NORMAL OPERATION - COOLDOWN LIMITATIONS APPLICABLE FOR FIRST 15.0 EFFECTIVE FULL POWER YEARS

Figure 3.1.2-3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 154 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 September 20, 1990, as supplemented by letters dated February 28, and August 14, 1991, Entergy Operations, Inc. (the licensee) submitted a request for changes to the Arkansas Nuclear One, Unit No. 1 (ANO-1) Technical Specification (TS). The requested changes would revise the reactor coolant system TS pressure/temperature (P/T) operating limits for the first 15 effective full power years (EFPYs), using the methodology of Regulatory Guide 1.99, Revision 2. The proposed amendment would also revise the low-temperature overpressure protection (LTOP) enable temperature.

The February 28, and August 14, 1991, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Proposed Fast Neutron Fluence

The determination of the reactor coolant pressure boundary material strength is required to comply with the provisions of Appendix G to 10 CFR Part 50. Analyses for the P/T limits for 15 EFPYs for ANO-1 are described in BAW-2106 by B&W Nuclear Service Company. The maximum inside pressure vessel exposure for 15 EFPYs is 0.488 E19 n/cm^2 and was estimated in BAW-2075, Revision 1. The methodology in BAW-2075 is based on BAW-1485, which is under NRC review. However, the NRC staff has separately reviewed BAW-2075, Revision 1, for the specific AN1-C capsule analysis and results. The transport calculations were carried out with the DOT-4.3 computer code using an S_8 angular quadrature and a P_3 scattering approximation. The calculation is based on the CASK cross section set with which DOT 4.3 has been benchmarked; thus, the calculation is acceptable. The dosimeters used ENDF/B-V based cross sections. A set of reasonable uncertainties has been used for the estimation of the fluence. Therefore, the staff finds the information in BAW-2075, Revision 1, adequate to accept the fluence estimate for 15 EFPYs.

2.2 Proposed PT Limits

The proposed P/T limits are valid for 15 EFPYs. The proposed P/T limits were developed using Regulatory Guide (RG) 1.99, Rev. 2. Generic Letter 88-11 recommends that RG 1.99, Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TSs for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TSs. The P/T limits are among the limiting conditions of operation in the TSs for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, specifies that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards, which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the ANO-1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 15 EFPYs was weld WF-18 with 0.29% copper (Cu), 0.55% nickel (Ni), and an initial reference temperature (RT_{ndt}) of -6°F.

The licensee has removed four surveillance capsules from ANO-1. The results from capsules E, B, A, and C were published in Babcock and Wilcox Reports BAW-1440, BAW-1698, BAW-1836, and BAW-2075, respectively. Surveillance capsules E, A, and C contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal. Surveillance capsule B contained only base metal and HAZ metal samples. These surveillance data have been used wherever applicable in calculating ART for beltline materials.

For the limiting beltline material, weld WF-18, the staff calculated the ART to be 175.2°F at 1/4T (T = reactor vessel beltline thickness) and 133.9°F for 3/4T at 15 EFPYs. The staff used a neutron fluence of 4.88E18 n/cm^2, which reduced to 2.94E18 n/cm^2 at 1/4T and 1.06E18 n/cm^2 at 3/4T. The ART was determined by using Section 1 of RG 1.99, Rev. 2, because weld WF-18 was not in the surveillance capsules.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 183°F at 15 EFPYs at 1/4T for the limiting beltline weld. This value reduced to 173°F when the extra margin of +10°F (5%) was removed. (This extra conservatism was mentioned in the licensee's response to the staff's request for additional information.) The staff judges that a difference of 2.2°F between the licensee's ART of 173°F (183°F-10°F) and the staff's ART of 175.2°F is acceptable. Substituting the ART of 175.2°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice 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 10°F and the preservice system hydrostatic test pressure of 3125 psi (1.25x2500 psi), the staff has determined that the corresponding temperature of 220°F from the proposed P/T limits satisfies Section IV.2 of Appendix G.

The staff finds that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 15 EFPYs because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the ANO-1 TSs.

Although the subject was not discussed by the licensee in the submittal, the staff notes that Section IV.A of Appendix G requires that the predicted Charpy USE at end-of-life (EOL) be above 50 ft-lb, unless it is demonstrated in a manner approved by the director, Office of Nuclear Reactor Regulation, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. The material with the lowest unirradiated USE and the highest Cu content is weld WF-18 with 0.29% Cu. Since surveillance data is not available for this type of weld, the staff used Figure 2 in RG 1.99, Rev. 2 directly and calculated the EOL USE at 1/4T (fluence of 4.58E18 n/cm^2) to be 44.9 ft-lb, which is below the required 50 ft-lb.

To satisfy the requirements of Section V.C of Appendix G, the licensee has to propose a program at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of Section V.B of this Appendix. Since further action is required, the staff requests that, within 60 days of receipt of this safety evaluation, the licensee respond regarding its plans to address this issue.

2.3 Proposed LTOP Limits

Standard Review Plan (SRP) 5.2.2 recommends that LTOP P/T limits meet the fracture mechanics criteria in Appendix G to the ASME Code Section III. However, the proposed LTOP limits were calculated using the non-Appendix G criteria that produced limits less restrictive than the limits calculated by the Appendix G criteria.

Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," states that "If changes can be implemented to show that the frequency of an LTOP event that would exceed Appendix G limits is expected to be much less than one per reactor lifetime, then the staff would consider alternatives to Appendix G LTOP setpoints." The alternatives allow licensees to establish LTOP P/T limits that are higher (less restrictive) than the Appendix G P/T limits.

In the proposed LTOP limits, the licensee considered that 1) an LTOP event has not occurred in over 100 years of the B&W nuclear plants operating experience and is therefore not an anticipated operational occurence; and 2) that the reactor coolant system in ANO-1, a B&W reactor, has the benefit of the nitrogen/steam bubble in the pressurizer. Such provision reduces the occurence of an LTOP by enabling the operator to reduce the reactor coolant system pressure within an allowable time.

Considering the operating experience at ANO-1 and other B&W plants, the staff judges that the ANO-1 LTOP limits may be calculated by using non-Appendix G criteria. The LTOP limits at Rancho Seco were calculated by using non-Appendix G criteria, and the staff approved those limits on a plant-specific basis. The staff has evaluated the proposed ANO-1 LTOP limits using the same criteria used in evaluating the Rancho Seco LTOP limits. The staff concludes that the proposed ANO-1 LTOP limits will provide adequate protection for the reactor vessel up to the requested EFPYs. Hence, the proposed LTOP limits may be incorporated in the ANO-1 TSs.

3.0 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.

4.0 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 in 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 (56 FR 890). 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.

5.0 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.

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