March 28, 1990

Dockets Nos.: 50-413
and 50-414

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: ISSUANCE OF AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NPF-35 AND AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NPF-52 - CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TACS 71477/71478)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 73 to Facility Operating License NPF-35 and Amendment No. 67 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 19, 1989, as supplemented September 13, 1989.

The amendments revise TS 3/4.4.9, "Pressure/Temperature Limits," and its associated Figures 3.4-2 and 3.4-3, and Bases. The amendments also revise TS Table 4.4-5 regarding the withdrawal schedule for the reactor vessel material surveillance program.

A copy of the related Safety Evaluation supporting the amendments is also enclosed. Notice of issuance of the amendments will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Kahtan N. Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 73 to NPF-35
2. Amendment No. 67 to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

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DATED: March 28, 1990

AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NPF-35 - Catawba Nuclear Station, Unit 1
AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NPF-52 - Catawba Nuclear Station, Unit 2

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

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 73
License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc., (licensees) dated April 19, 1989, as supplemented September 13, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   D. The issuance of this 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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 73 , are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: March 28, 1990

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

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency, (licensees) dated April 19, 1989, as supplemented September 13, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   D. The issuance of this 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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 67 , are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: March 28, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 73

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 67

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
XIV	XIII
3/4 4-32	3/4 4-31
3/4 4-33	
3/4 4-34	
3/4 4-35	
B3/4 4-8	B3/4 4-7
B3/4 4-12	B3/4 4-11

SECTION PAGE

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3/4.4.10 STRUCTURAL INTEGRITY..... B 3/4 4-16
3/4.4.11 REACTOR COOLANT SYSTEM VENTS..... B 3/4 4-17

3/4.5 EMERGENCY CORE COOLING SYSTEMS

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3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS..... B 3/4 5-1
3/4.5.4 REFUELING WATER STORAGE TANK..... B 3/4 5-2

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3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE..... B 3/4 7-1
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3/4.7.7 AUXILIARY BUILDING FILTERED EXHAUST SYSTEM..... B 3/4 7-4
3/4.7.8 SNUBBERS..... B 3/4 7-4
3/4.7.9 SEALED SOURCE CONTAMINATION..... B 3/4 7-6
3/4.7.10 FIRE SUPPRESSION SYSTEMS..... B 3/4 7-6
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3/4.7.12 GROUNDWATER LEVEL..... B 3/4 7-7
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3/4.1.1	BORATION CONTROL	B 3/4 1-1
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3/4.1.3	MOVABLE CONTROL ASSEMBLIES	B 3/4 1-3
3/4.2	POWER DISTRIBUTION LIMITS	B 3/4 2-1
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3/4.2.2	and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR	B 3/4 2-2
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3/4.4.9	PRESSURE/TEMPERATURE LIMITS	B 3/4 4-7

TABLE 4.4-4 (Continued)

TABLE NOTATIONS

#Until the specific activity of the Reactor Coolant System is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.

***A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

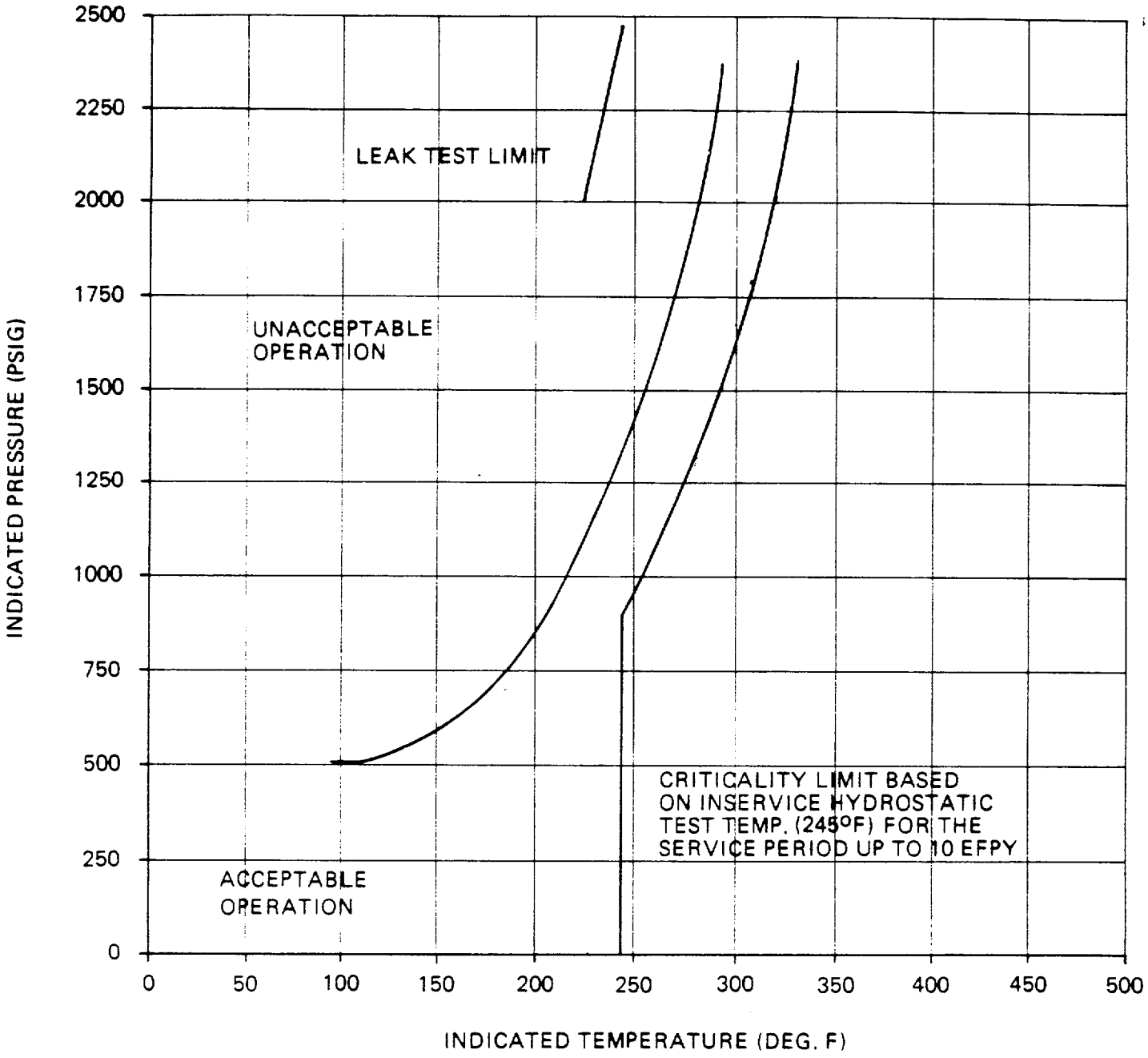
ACTION:

With any of the above limits 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 structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

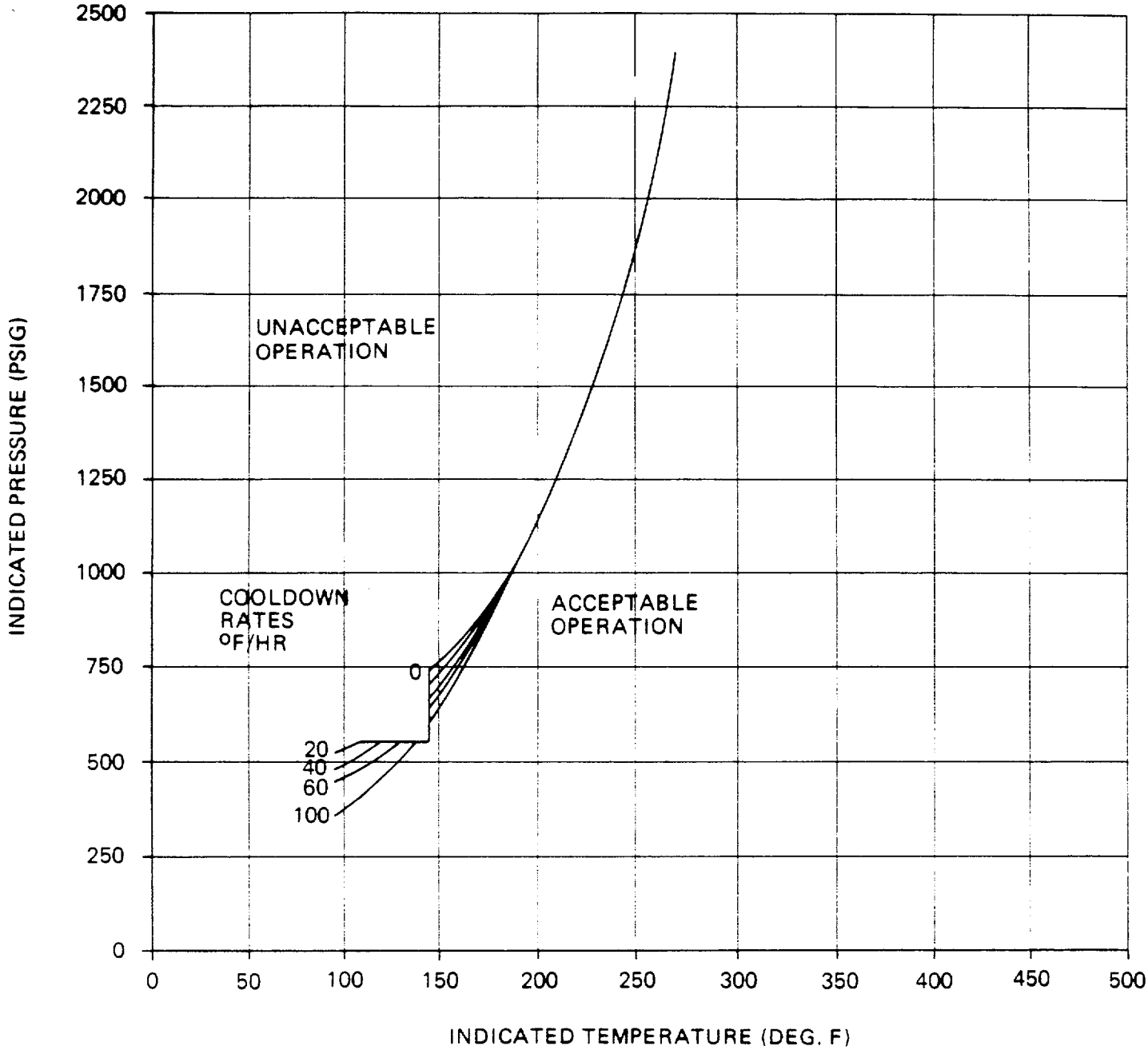
4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.



CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

MATERIAL BASIS
CONTROLLING MATERIAL– UNIT 2 INTERMEDIATE SHELL PLATE B8605–2
COPPER CONTENT–0.07wt%
NICKEL CONTENT– 0.61 wt %
RT_{NDT} INITIAL–33°F
RT_{NDT} AFTER 10 EFPY 1/4T, 104°F
3/4T, 92°F

FIGURE 3.4–2 CATAWBA NUCLEAR STATION, REACTOR COOLANT SYSTEM, HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 10 EFPY



CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

MATERIAL BASIS
CONTROLLING MATERIAL– UNIT 2 INTERMEDIATE SHELL PLATE B8605-2
COPPER CONTENT–0.07wt%
NICKEL CONTENT–0.61 wt%
RT_{NDT} INITIAL–33°F
RT_{NDT} AFTER 10 EFPY 1/4T, 104°F
3/4T, 92°F

FIGURE 3.4–3 CATAWBA NUCLEAR STATION, REACTOR COOLANT SYSTEM, COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 10 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	UNIT 1 LEAD FACTOR	UNIT 2 LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	58.5°	3.85	4.01	Standby
V	61°	3.65	3.74	9
W	121.5°	3.85	4.01	Standby
X	238.5°	3.85	4.01	Standby
Y	241°	3.65	3.74	5
Z	301.5°	3.85	4.05	First Refueling

Amendment No. 73 (Unit 1)
Amendment No. 67 (Unit 2)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively, and

5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the vessel are determined in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the NRC Branch Technical Position MTEB 5-2, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

PRESSURE/TEMPERATURE LIMITS (Continued)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of the effective full power years (EFPY) of service life as indicated on the applicable heatup or cooldown curves. The service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT}.

For Unit 2 the adjusted reference temperature has been computed using the guidance of Regulatory Guide 1.99, Revision 2. For Unit 1, the analysis documented in WCAP 11527 and reanalyzed using the guidance of Regulatory Guide 1.99, Revision 2, indicates the heatup and cooldown limit curves in Figures 3.4-2 and 3.4-3 are applicable to both units to predict the shift in RT_{NDT} at the end of the identified service life.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the pressure vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the pressure vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in the following paragraphs.

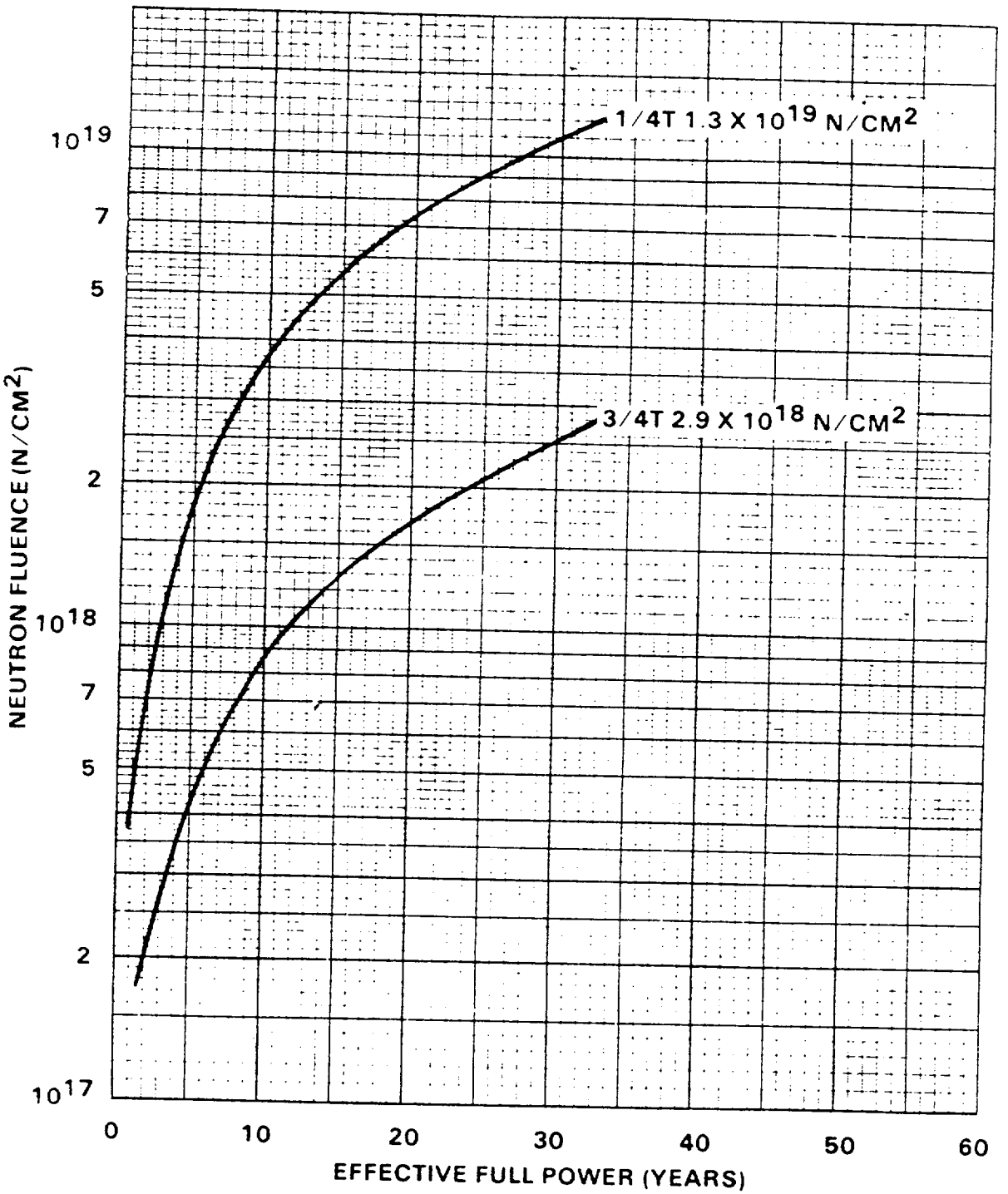


FIGURE B 3/4.4-1
FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKETS NOS. 50-413 AND 50-414

1.0 INTRODUCTION

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," the Duke Power Company, et al. (the licensee), requested NRC approval to revise the pressure/temperature (P/T) limits in Catawba Nuclear Station, Units 1 and 2, Technical Specification (TS) 3.4. The request was documented in a letter from the licensee dated April 19, 1989, and a September 13, 1989, supplement. The purpose of the revision is to change the effectiveness of the P/T limits from 16 to 10 effective full power years (EFPY). The licensee proposed to use one set of P/T limits for both units. The proposed P/T limits were developed from Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the NRC staff used 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, Revision 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, 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, Revision 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.

2.0 EVALUATION

The NRC staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Catawba 1 and 2 reactor vessels. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The material with the highest ART (most embrittled) at 10 EFPY for both units was the intermediate shell plate material (B8605-2) in Unit 2 which contains 0.07% copper and 0.61% nickel, and has an initial RT of 33^{o}F.

The licensee has removed one surveillance capsule from Unit 1 and one capsule from Unit 2. The results from capsule Z in Unit 1 were published in Westinghouse Report WCAP-11527. The results from capsule Z in Unit 2 were published in Westinghouse Report WCAP-11941. Both surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, intermediate shell plate material B8605-2 in Unit 2, the NRC staff calculated the ART at 10 EFPY at 1/4T (T = reactor vessel beltline thickness) to be 104^{o}F. The staff used a neutron fluence of 9.5E18 n/cm^{2} at the inner surface at 10 EFPY. The ART was determined by Section 1 of RG 1.99, Revision 2, because only one surveillance capsule was removed from the Unit 2 reactor vessel.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 104^{o}F at 10 EFPY at 1/4T for the same limiting plate material. Substituting the ART of 104^{o}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 also 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, the NRC staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The measured initial Charpy USE is 82 ft-lb for intermediate shell plate metal (B8605-2). Using the method in RG 1.99, Revision 2, the predicted Charpy USE of the intermediate shell plate metal at the end of life will be about 60 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

Based on its review, the NRC staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 10 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittals also satisfy Generic Letter 88-11 because the licensee used the method in RG 1.99, Revision 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Catawba Units 1 and 2 TSs.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet 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 these amendments.

4.0 CONCLUSION

The Commission's proposed determination that the amendments involve no significant hazards consideration was published in the Federal Register (55 FR 4264) on February 7, 1990. The Commission consulted with the State of South Carolina. No public comments were received, and the State of South Carolina did not have any comments.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health an safety of the public.

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