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Docket Nos. 50-269
50-270
and 50-287

November 4, 1977

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President
Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

The Commission has issued the enclosed Amendment Nos. 51, 51 and 48 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2 and 3. These amendments consist of changes to the Stations common Technical Specifications and are in response to your request dated June 6, 1977.

These amendments revise the common Oconee Technical Specifications to incorporate changes to the Oconee Unit 2 pressurization heatup and cooldown limitations.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original signed by
A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 51 to DPR-38
2. Amendment No. 51 to DPR-47
3. Amendment No. 48 to DPR-55
4. Safety Evaluation
5. Notice of Issuance

cc w/enclosures:
See next page

x27433:tsb

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SURNAME➤	DNeighbors		ASchwencer			
DATE➤	10/31/77	11/3/77	11/4/77			

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE 1976 – 626-624

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Micheal McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806-15th Street, NW.,
Washington, D.C. 20005

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Coutland Street, N. E.
Atlanta, Georgia 30308

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated June 6, 1977, 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 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 3.B of Facility License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 51 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 4, 1977

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. DPR-47

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

   A. The application for amendment by Duke Power Company (the licensee) dated June 6, 1977, 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 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 3.B of Facility License No. DPR-47 is hereby amended to read as follows:

   3.B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 51, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 4, 1977

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated June 6, 1977, 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 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 3.B of Facility License No. DPR-55 is hereby amended to read as follows:

   3.B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: November 4, 1977

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 51 TO DPR-38

AMENDMENT NO. 51 TO DPR-47

AMENDMENT NO. 48 TO DPR-55

DOCKET NOS. 50-269, 50-270 and 50-287

Revise Appendix A as follows:

Remove the following pages and replace with identically numbered pages.

3.1-3
3.1-3a
3.1-4
3.1-5
3.1-6a
3.1-7a
3.1-7b
3.1-8
3.1-9

Add the following new pages:

3.1-6b
3.1-7c
3.1-7d

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited as follows:

Heatup:

Heatup rates and allowable combinations of pressure and temperatures shall be limited in accordance with Figure 3.1.2-1A Unit 1
3.1.2-1B Unit 2
3.1.2-1C Unit 3.

Cooldown:

Cooldown rates and allowable combinations of pressure and temperature shall be limited in accordance with Figure 3.1.2-2A Unit 1
3.1.2-2B Unit 2
3.1.2-2C Unit 3.

3.1.2.2 Leak Tests

Leak tests required by Specification 4.3 shall be conducted under the provisions of 3.1.2.1.

3.1.2.3 Hydro Tests

For thermal steady state system hydro test 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 3.1.2.1 and to ASME Code Section III limits when no fuel assemblies are present provided the reactor coolant system is to the right of and below the limit line in Figure 3.1.2-3A Unit 1
3.1.2-3B Unit 2.

3.1.2.4 The secondary side of the steam generator shall not be pressurized above 237 psig if the temperature of the vessel shell is below 110°F.

3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 410°F.

3.1.2.6 Pressurization heatup and cooldown and hydro test limits shall be updated based on the results of the reactor vessel materials surveillance program. These revised limits shall be submitted to the NRC at least 90 days prior to exceeding four (Units 1 & 2) effective full power years of operation or an integrated exposure of 1.7×10^{18} n/cm^2 or DTT 144°F for Unit 3.

Bases - Units 1 and 2

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, startup and shutdown operations, and inservice leak and hydrostatic tests. The various categories of load cycles used for design purposes are provided in Table 4.8 of the FSAR.

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-1421(7) and BAW-1437(8).

The figures specified in 3.1.2.1, 3.1.2.2 and 3.1.2.3 present the pressure-temperature limit curves for normal heatup, normal cooldown and hydrostatic test respectively. The limit curves are applicable up to the indicated effective full power years of operation. These curves are adjusted by 25 psi and 10°F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also 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 the figure specified in 3.1.2.1 for reactor criticality and on the figure specified in 3.1.2.3 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, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core areas, or in test reactors.

The limitation on steam generator pressure and temperature provide protection against nonductile failure of the secondary side of the steam generator. At metal temperatures lower than the RT_{NDT} of +60°F, the protection against nonductile failure if achieved by limiting the secondary coolant pressure to 20 percent of the preoperational system hydrostatic test pressure. The limitations of 110°F and 237 psig are based on the highest estimated RT_{NDT} of +40°F and the preoperational system hydrostatic test pressure of 1312 psig. The average metal temperature is assumed to be equal to or greater than the coolant temperature. The limitations include margins of 25 psi and 10°F for possible instrument error.

The spray temperature difference is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

Bases Unit 3

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 rate of 100°F per hour satisfies stress limits for cyclic operation. (2) The 237 psig pressure limit for the secondary side of the steam generator at a temperature less than 110°F satisfies stress levels for temperatures below the DTT. (3) The reactor vessel plate material and welds have been tested to verify conformity to specified requirements and a maximum NDTT value of 20°F has been determined based on Charpy V-Notch tests. The maximum NDTT value obtained for the steam generator shell material and welds was 40°F.

Figures 3.1.2-1C and 3.1.2-2C contain the limiting reactor coolant system pressure-temperature relationship for operation at DTT(4) and below to assure that stress levels are low enough to preclude brittle fracture. These stress levels and their bases are defined in Section 4.3.3 of the FSAR.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT with accumulated nuclear operation. The predicted maximum NDTT increase for the 40-year exposure is shown on Figure 4.10.(4) The actual shift in NDTT will be determined periodically during plant operation by testing of irradiated vessel material samples located in this reactor vessel.(5) The results of the irradiated sample testing will be evaluated and compared to the design curve (Figure 4-11 of FSAR) being used to predict the increase in transition temperature.

The design value for fast neutron (E > 1 MeV) exposure of the reactor vessel is 3.0 x 10^{10} n/cm^2 -- s at 2,568 MWt rated power and an integrated exposure of 3.0 x 10^{19} n/cm^2 for 40 years operation. (6) The calculated maximum values are 2.2 x 10^{10} n/cm^2 -- s and 2.2 x 10^{19} n/cm^2 integrated exposure for 40 years operation at 80 percent load. (4) Figure 3.1.2-1C is based on the design value which is considerably higher than the calculated value. The DTT value for Figure 3.1.2-1C is based on the projected NDTT at the end of the first two years of operation. During these two years, the energy output has been conservatively estimated to be 1.7 x 10^6 thermal megawatt days, which is equivalent to 655 days at 2,568 MWt core power. The projected fast neutron exposure of the reactor vessel for the two years is 1.7 x 10^{18} n/cm^2 which is based on the 1.7 x 10^6 thermal megawatt days and the design value for fast neutron exposure.

The actual shift in NDTT will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of this or a similar vessel in the core area or in test reactors. To compensate for the increases in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the established stress limits during heatup and cooldown.

The NDTT shift and the magnitudes of the thermal and pressure stresses are sensitive to integrated reactor power and not to instantaneous power level. Figure 3.1.2-1C and 3.1.2-2C are applicable to reactor core thermal ratings up to 2,568 MWt.

The pressure limit line on Figure 3.1.3-1C has been selected such that the reactor vessel stress resulting from internal pressure will not exceed 15 percent yield strength considering the following:

1. A 25 psi error is measured pressure.
2. System pressure is measured in either loop.
3. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations.

For adequate conservatism in fracture toughness including size (thickness) affect, a maximum pressure of 550 psig below 275°F with a maximum heatup and cooldown rate of 50°F/hr has been imposed for the initial two year period as shown on Figure 3.1.2-1C. During this two year period, a fracture toughness criterion applicable to Oconee Unit 3 beyond this period will be developed by the AEC. It will be based on the evaluation of the fracture toughness properties of heavy section (thickness) steels, both irradiated and unirradiated, for the AEC-HSST program and the PVRC program, and with considerations of test results of the Oconee Units 2 and 3 reactor surveillance programs.

The spray temperature difference restriction 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.

REFERENCES

(1) FSAR Section 4.1.2.4.

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

(3) FSAR Section 4.3.10.5.

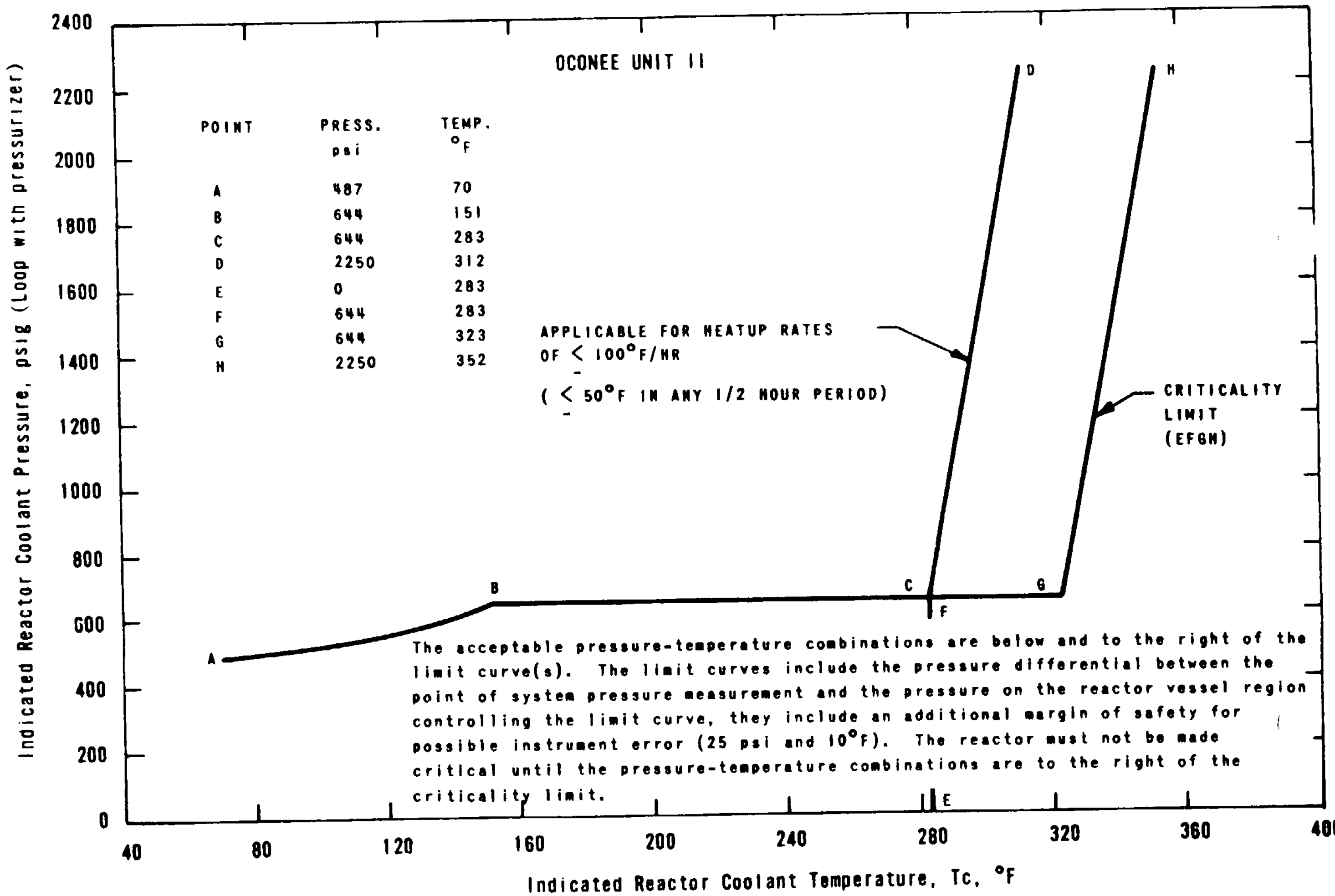
(4) FSAR Section 4.3.3.

(5) FSAR Section 4.4.6.

(6) FSAR Sections 4.1.2.8 and 4.3.3.

(7) Analysis of Capsule OC1-F from Duke Power Company Oconee Unit 1 Reactor Vessel Materials Surveillance Program, BAW-1421 Rev. 1, September 1975.

(8) Analysis of Capsule OC11-C from Duke Power Company Oconee Unit 2 Reactor Vessel Materials Surveillance Program, BAW-1437, May, 1977.



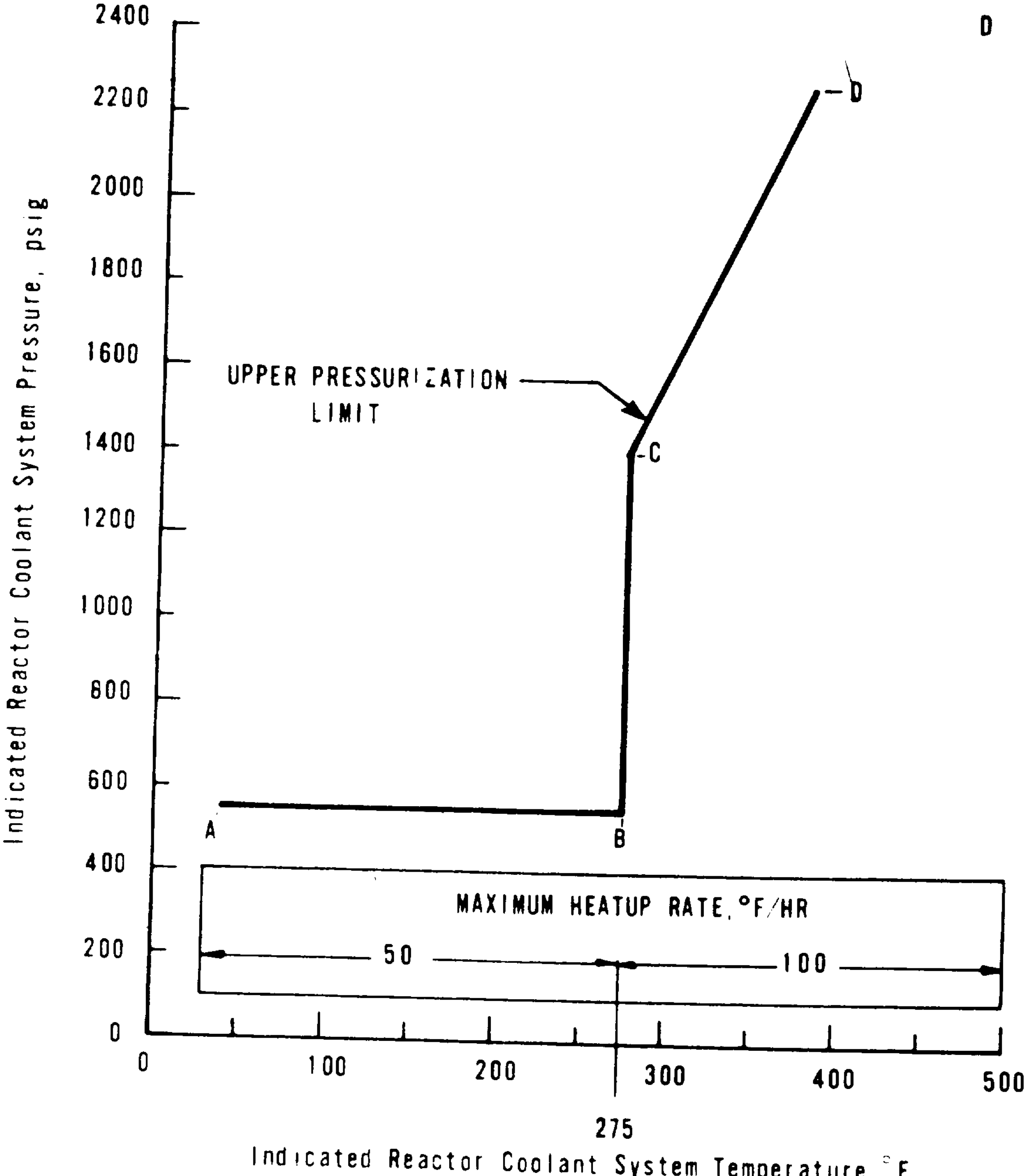
UNIT 2
REACTOR COOLANT SYSTEM
NORMAL OPERATION HEATUP LIMITATIONS
APPLICABLE FOR FIRST 4.0 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-1B

Amendment Nos. 51, 51 & 48

POINT	TEMP.	PRESS.
A	40	550
B	275	550
C	275	1400
D	380	2275



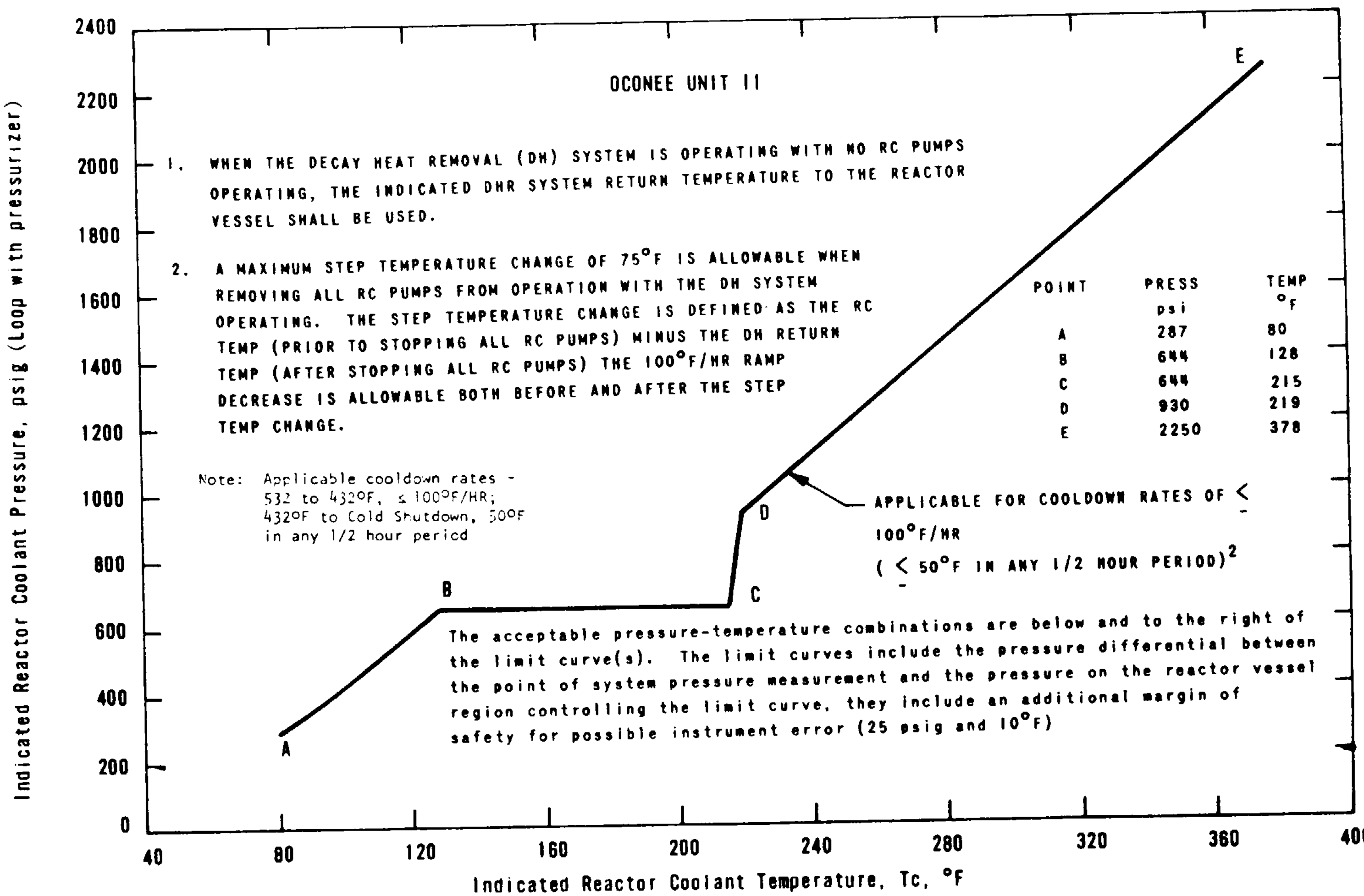
OCONEE NUCLEAR STATION

Unit 3

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS

(APPLICABLE UP TO AN INTEGRATED EXPOSURE OF 1.7 x 10^{18} n/cm^2 OR DTT 144 °F)

Figure 3.1.2-1C



UNIT 2
REACTOR COOLANT SYSTEM
NORMAL OPERATION COOLDOWN LIMITATIONS
APPLICABLE FOR FIRST 4.0 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-2B

POINT	TEMP	PRESS
A	380	2275
B	275	1400
C	275	550
D	250	550
E	250	450
F	175	450
G	175	200
H	120	200

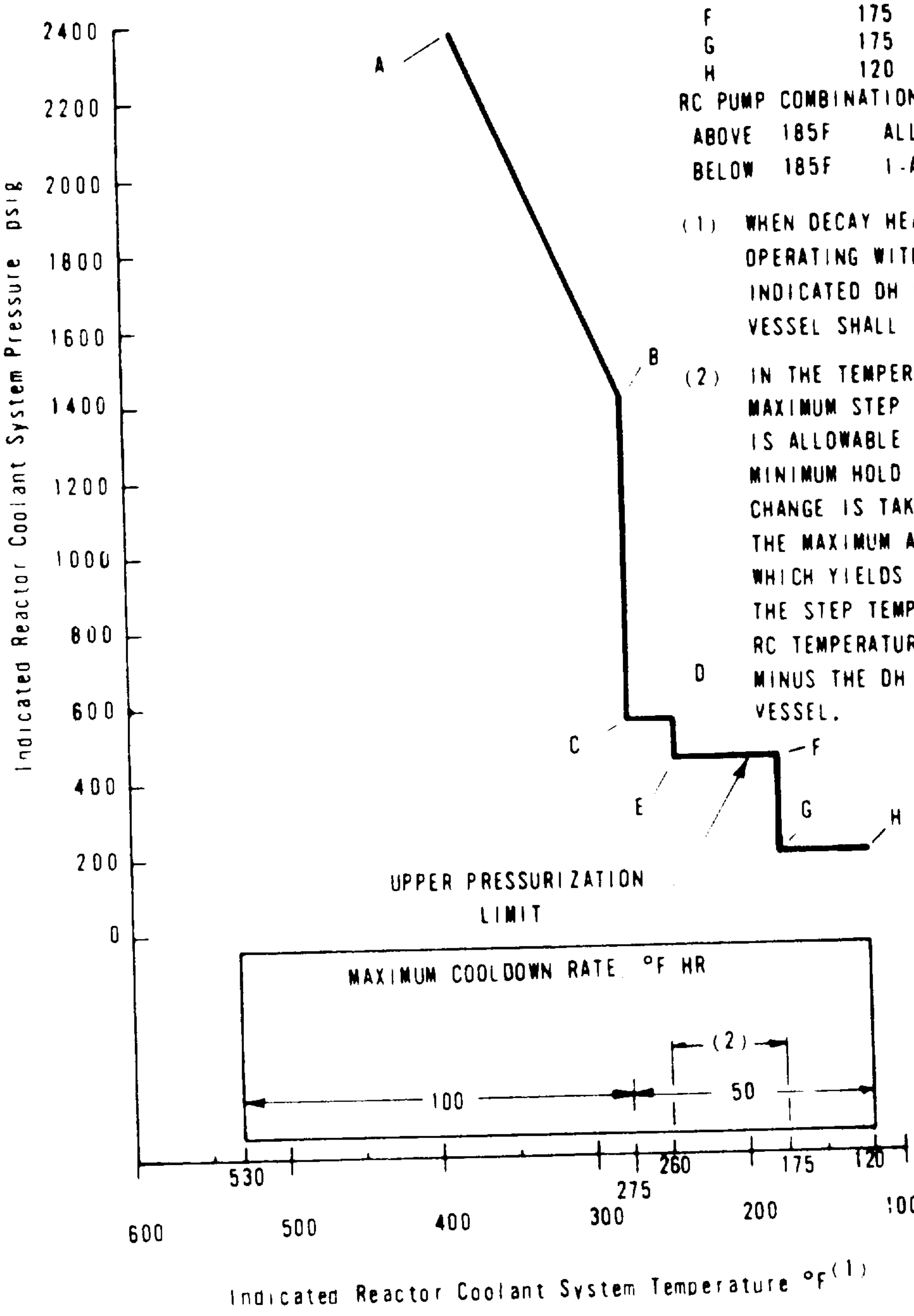
RC PUMP COMBINATIONS ALLOWABLE

ABOVE 185F ALL

BELOW 185F 1-A 1-B;0-A 2-B;1-A 0-B;0-A 1-B

(1) WHEN DECAY HEAT REMOVAL SYSTEM (DH) IS OPERATING WITHOUT ANY RC PUMPS OPERATING, INDICATED DH RETURN TEMP TO THE REACTOR VESSEL SHALL BE USED

(2) IN THE TEMPERATURE RANGE 260F TO 175F, A MAXIMUM STEP TEMPERATURE CHANGE OF 75F IS ALLOWABLE FOLLOWED BY A ONE HOUR MINIMUM HOLD ON TEMPERATURE. IF THE STEP CHANGE IS TAKEN BELOW 250F RC TEMPERATURE, THE MAXIMUM ALLOWABLE STEP SHALL BE THAT WHICH YIELDS A FINAL TEMPERATURE OF 175F. THE STEP TEMPERATURE CHANGE IS DEFINED AS RC TEMPERATURE(BEFORE STOPPING ALL RC PUMPS) MINUS THE DH RETURN TEMPERATURE TO THE REACTOR VESSEL.

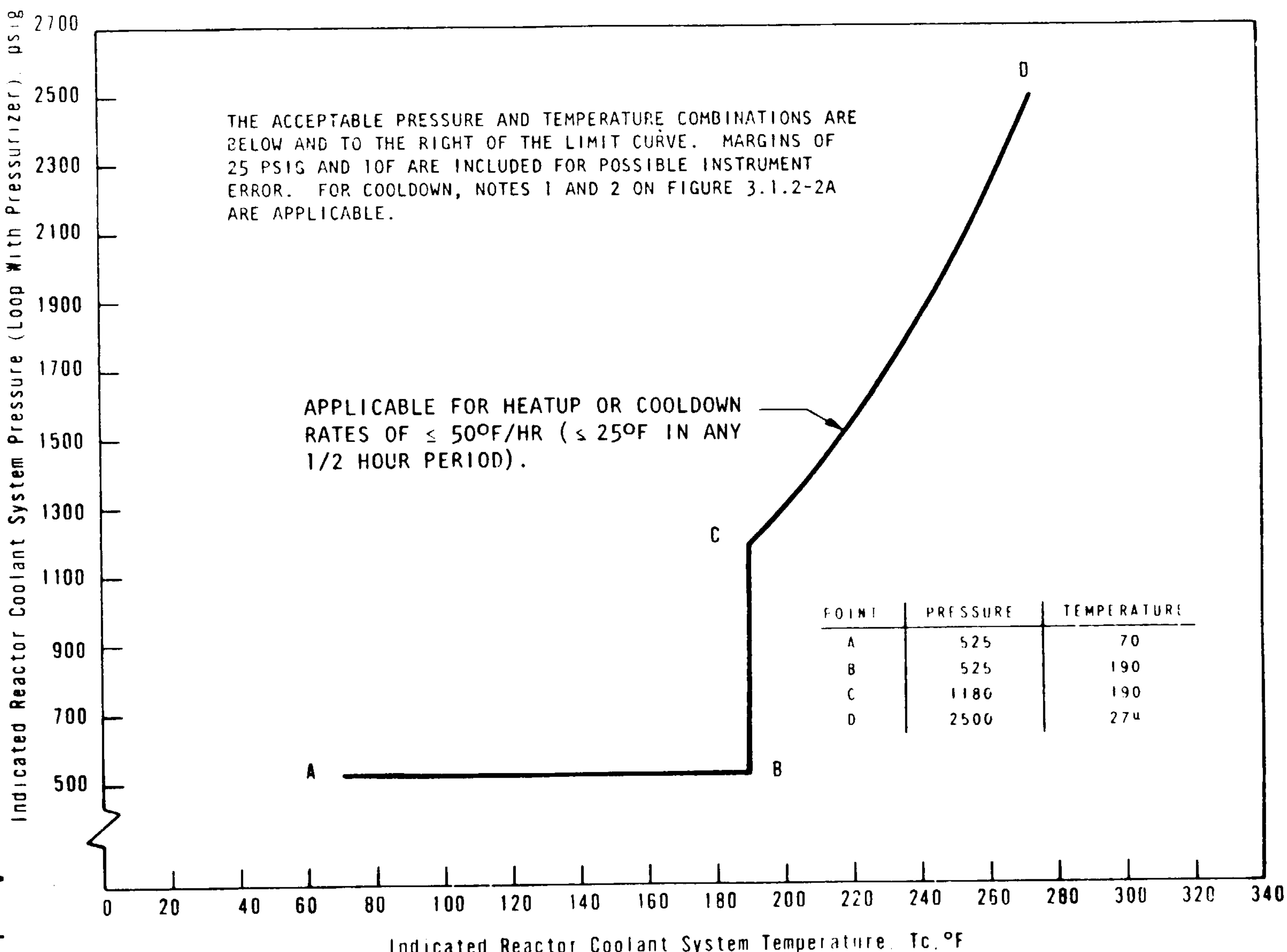


REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
(APPLICABLE UP TO DTT = 185°F)

Unit 3

OCONEE NUCLEAR STATION

Figure 3.1.2 - 2c

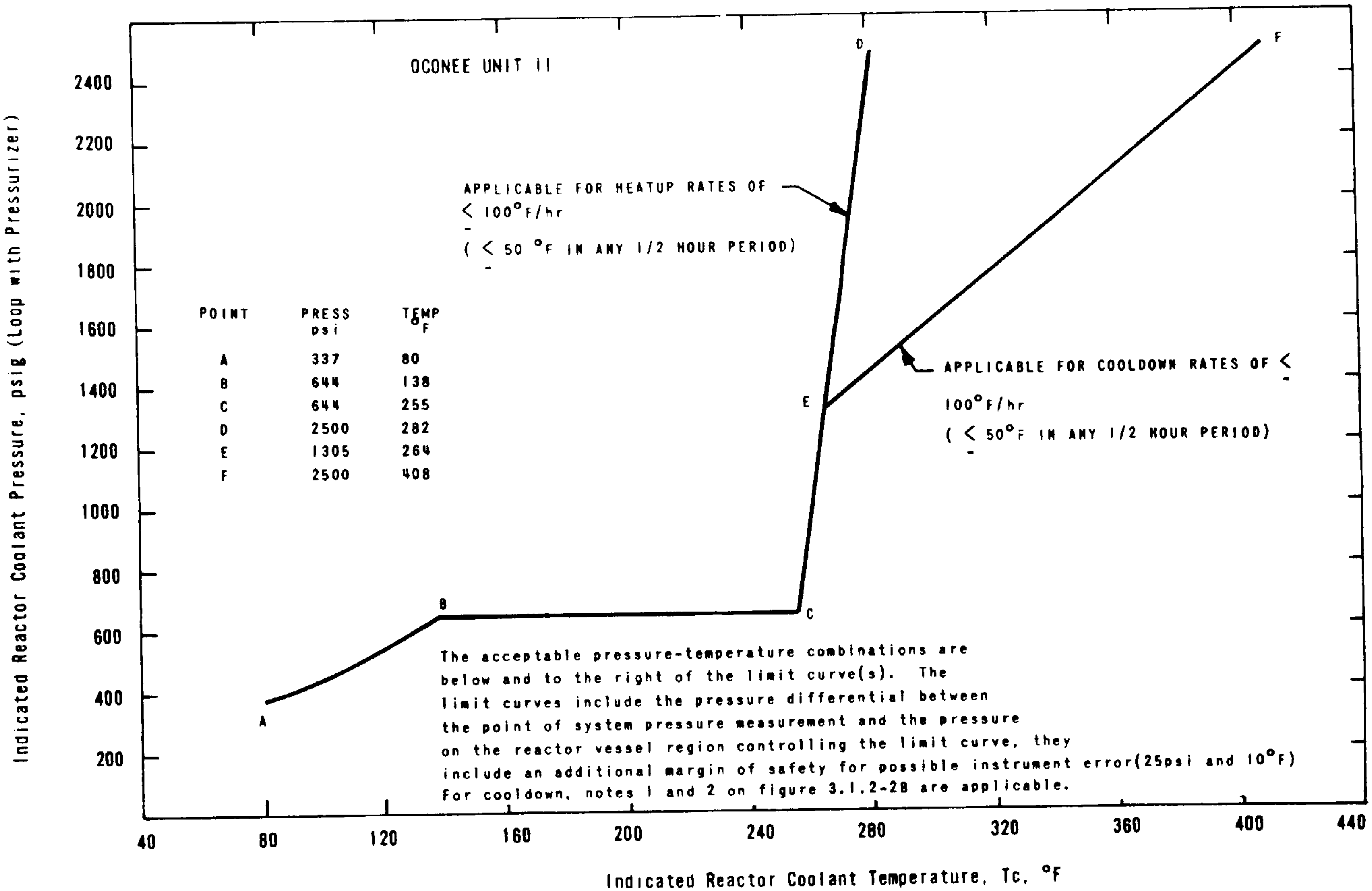


POINT	PRESSURE	TEMPERATURE
A	525	70
B	525	190
C	1180	190
D	2500	270

UNIT 1
REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMITATIONS FOR INSERVICE HYDROSTATIC TESTS (NO FUEL ASSEMBLIES IN THE CORE), APPLICABLE FOR FIRST 4 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-3



UNIT 2
REACTOR COOLANT SYSTEM
INSERVICE LEAK AND HYDROSTATIC
TEST AND COOLDOWN LIMITATIONS
APPLICABLE FOR FIRST 4.0 EFPY

OCONEE NUCLEAR STATION
Figure 3.1.2-3B

Amendments Nos. 51, 51 & 48

3.1.3 <u>Minimum Conditions for Criticality</u>

<u>Specification</u>

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above the criticality limit of 3.1.2-1A (Unit 1) or above DTT + 10°F (Unit 3).
3.1.2-1B (Unit 2)

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1%$\Delta k/k$ until a steam bubble is formed and a water level between 80 and 396 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. The regulating rods shall then be positioned within their position limits defined by Specification 3.5.2.5 prior to deboration.

<u>Bases</u>

At the beginning of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods.(1) Calculations show that above 525°F, the consequences are acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature,(2) startup and operation of the reactor when reactor coolant temperature is less than 525°F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient(2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1$\Delta k/k$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below the limits of Specification 3.1.2-1 provides increased assurance that the proper relationship between primary coolant pressure and temperature will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than 1% subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a startup accident.[3]

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

<u>REFERENCES</u>

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.4

(3) FSAR, Supplement 3, Answer 14.4.1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 51 TO LICENSE NO. DPR-38

AMENDMENT NO. 51 TO LICENSE NO. DPR-47

AMENDMENT NO. 48 TO LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

DOCKET NOS. 50-269, 50-270 AND 50-287

Introduction

By letter dated June 6, 1977, Duke Power Company (licensee) requested revisions to the Oconee Nuclear Station Technical Specifications which would incorporate changes to the Oconee Unit 2 pressurization, heatup and cooldown limitations (pressure-temperature operating limit curves) and to the reactor vessel material surveillance program.

Discussion

The existing Oconee Nuclear Station Technical Specifications curves for heatup and cooldown are applicable for 1.7×10^6 thermal megawatt-days (1.8 effective full power years). The proposed change will extend this time to 4 effective full power years (EFPY).

Evaluation

Heatup and Cooldown Limitations

The proposed revised Oconee Unit 2 pressure-temperature operating limit curves, Figures 3.1.2-1B, 3.1.2-2B and 3.1.2-3B are based on data from Babcock & Wilcox Report, "Analysis of Capsule OCII-C From Duke Power Company Oconee Nuclear Station, Unit 2, Reactor Vessel Materials Surveillance Program," BAW-1437, dated May 1977. The proposed curves are projected for 6 EFPY.

We have reviewed BAW-1437 and the proposed pressure-temperature operating limit curves. Capsule C from Unit 2 contained specimens of weld metal WF 209-1. This weld is not identical to the welds in Oconee Unit 2. However, this weld has a similar chemical composition and was made using weld procedure similar to those for high copper Oconee vessel welds. Therefore we conclude that the properties of the limiting weld metals in the Oconee reactor vessels will be affected by irradiation in a manner similar to these WF 209-1 specimens. Also, the Oconee Unit 2 and Unit 1 neutron flux, flux spectrum and weld material mechanical properties are similar. The Oconee Unit 1 pressure-temperature limits were approved on February 23, 1977, and were applicable for 4 EFPY. For the reasons stated in our Safety Evaluation Report issued February 23, 1977 on Oconee Unit 1, we have concluded that the proposed temperature operating limit curves for Unit 2 should also be limited to 4 EFPYs of operation.

Based on our review of the Oconee Unit 2 pressure-temperature limits and the similarity of the Oconee Unit 2 and Unit 1 designs, materials and operating conditions, we conclude that the operating limits proposed for Unit 2 are in conformance with Appendix G, 10 CFR Part 50, and are therefore acceptable.

Environmental Consideration

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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 and safety of the public.

Date: November 4, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-269, 50-270 AND 50-287

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 51, 51 and 48 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company which revised Technical Specifications for operation of the Oconee Nuclear Station Unit Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective within 30 days of the date of issuance.

These amendments revise the common Oconee Technical Specifications to incorporate changes to the Oconee Unit 2 pressurization heatup and cooldown limitations.

The application for these amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated June 6, 1977, (2) Amendment Nos. 51, 51 and 48 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, (3) the Commission's related Safety Evaluation and (4) the Commission's Safety Evaluation dated February 23, 1977. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555 and at the Oconee County Library, 201 South Spring, Walhalla, South Carolina 29691. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 4th day of November 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors