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. 38
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 October 1, 1975, 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.

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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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 38, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its 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: February 23, 1977

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 38 TO DPR-38

AMENDMENT NO. 38 TO DPR-47

AMENDMENT NO. 35 TO DPR-55

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

Revise Appendix A as follows:

1. Remove pages 3.1-3 through 3.1-9 and insert revised identically numbered pages.

2. Add pages 3.1-3a, 3.1-6a, 3.1-7a and 3.1-7b.

## 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 Units 2&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 Units 2&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 3.1.2-3.

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 limitations shall be updated based on the results of the reactor vessel materials surveillance program described in Specification 4.2.8.

Bases - Unit 1

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].

Figures 3.1.2-1A, 3.1.2-2A, 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 fifth effective full power year 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 Figure 3.1.2-1A for reactor criticality and on Figure 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, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of the reactor vessel in the core region.

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 Units 2 and 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 3.1.2-1B and 3.1.2-2B 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 $3.0 \times 10^{10}$ n/cm$^2$ -- s at 2,568 MWt rated power and an integrated exposure of $3.0 \times 10^{19}$ n/cm$^2$ for 40 years operation. (6) The calculated maximum values are $2.2 \times 10^{10}$ n/cm$^2$ -- s and $2.2 \times 10^{19}$ n/cm$^2$ integrated exposure for 40 years operation at 80 percent load. (4) Figure 3.1.2-1B is based on the design value which is considerably higher than the calculated value. The DTT value for Figure 3.1.2-1B 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 $1.7 \times 10^6$ 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 $1.7 \times 10^{18}$ n/cm$^2$ which is based on the $1.7 \times 10^6$ 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 the vessel in the core area. 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 3.1.2-1B and 3.1.2-2B are applicable to reactor core thermal ratings up to 2,568 MWt.

The pressure limit line on Figure 3.1.3-1B 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 3.1.2-1B. During this two year period, a fracture toughness criterion applicable to Oconee Units 2 and 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.

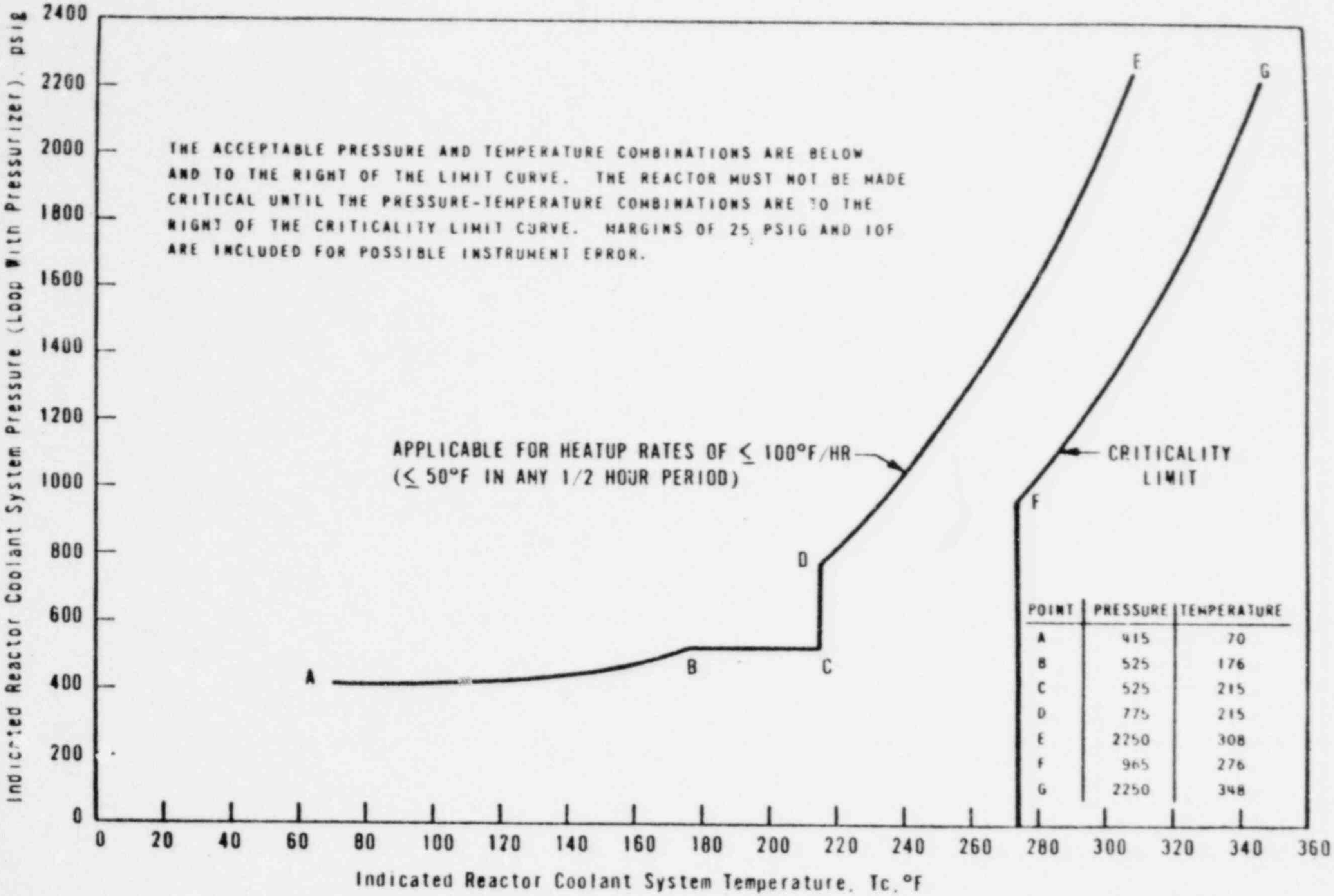
(3) FSAR Section 4.3.10.5.

(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.




| POINT | PRESSURE | TEMPERATURE |
|---|---|---|
| A | 415 | 70 |
| B | 525 | 176 |
| C | 525 | 215 |
| D | 775 | 215 |
| E | 2250 | 308 |
| F | 965 | 276 |
| G | 2250 | 348 |

Unit 1 Only

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS, APPLICABLE FOR FIRST 4 EFPY

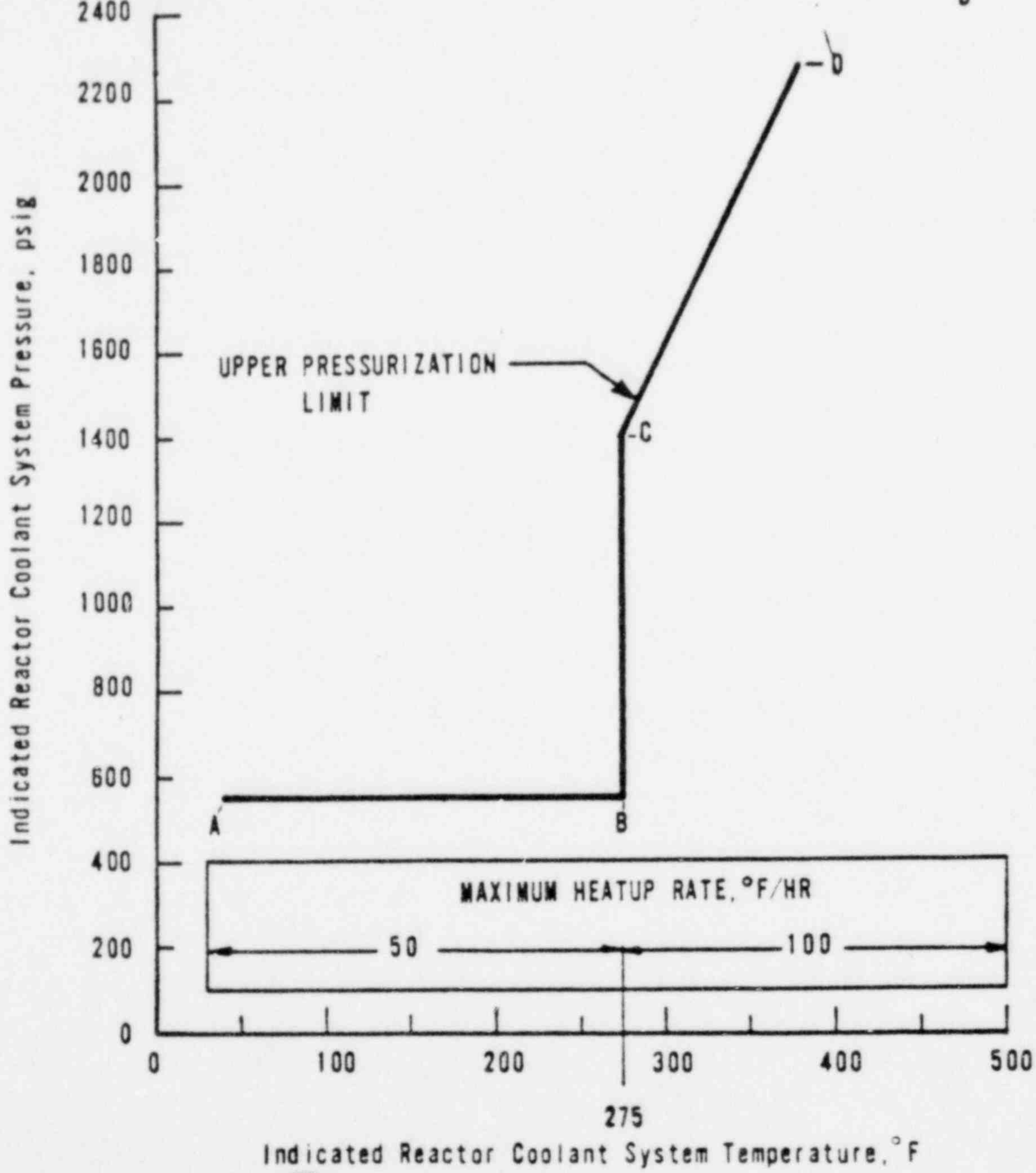
OCONEE NUCLEAR STATION

Figure 3.1.2-1A



Amendments Nos. 38, 38 & 35

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



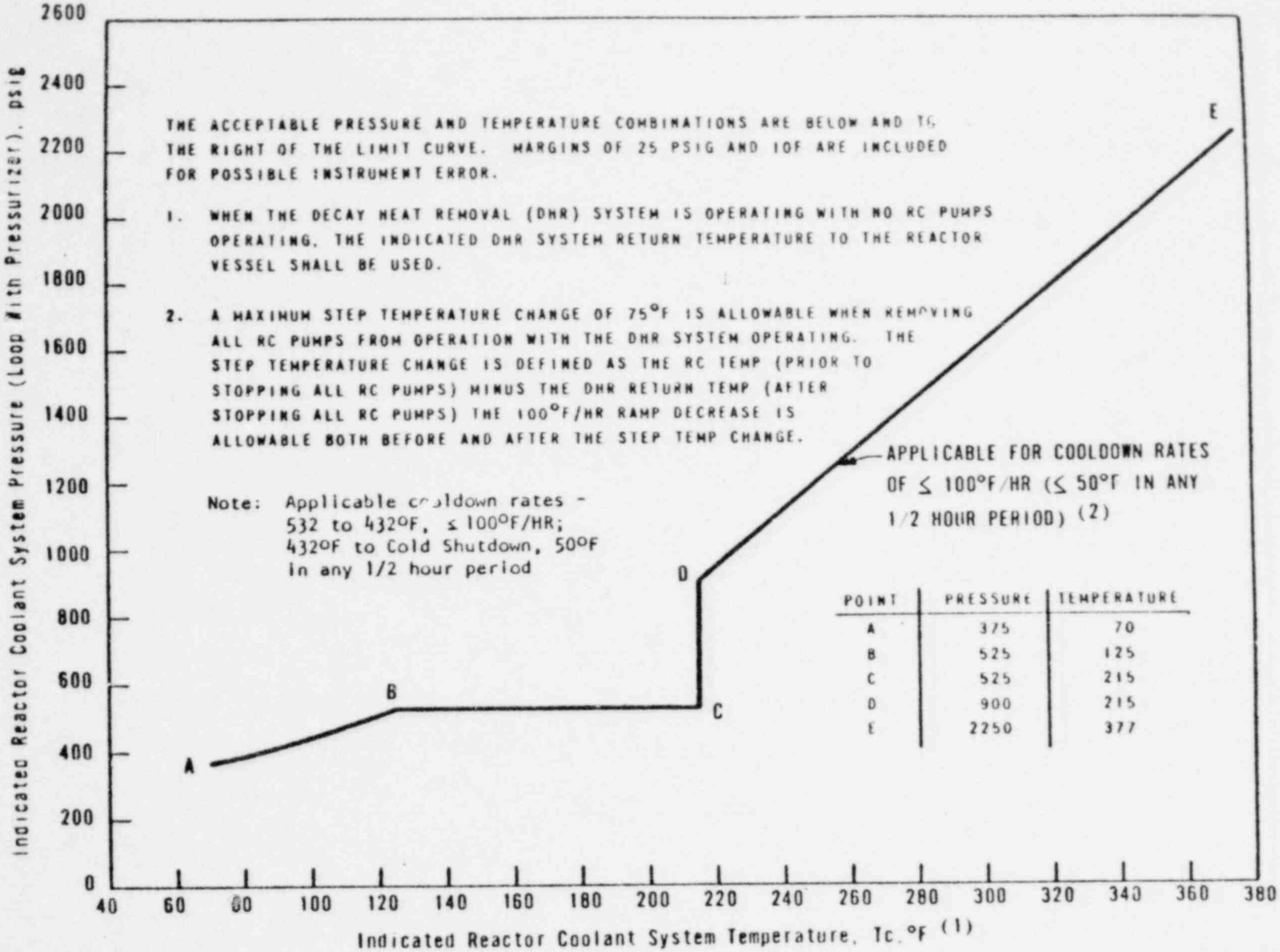

OCONEE NUCLEAR STATION

Units 2 & 3

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS

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

Figure 3.1.2-1B




| POINT | PRESSURE | TEMPERATURE |
|---|---|---|
| A | 375 | 70 |
| B | 525 | 125 |
| C | 525 | 215 |
| D | 900 | 215 |
| E | 2250 | 377 |

Unit 1 Only

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS, APPLICABLE FOR FIRST 4 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-2A

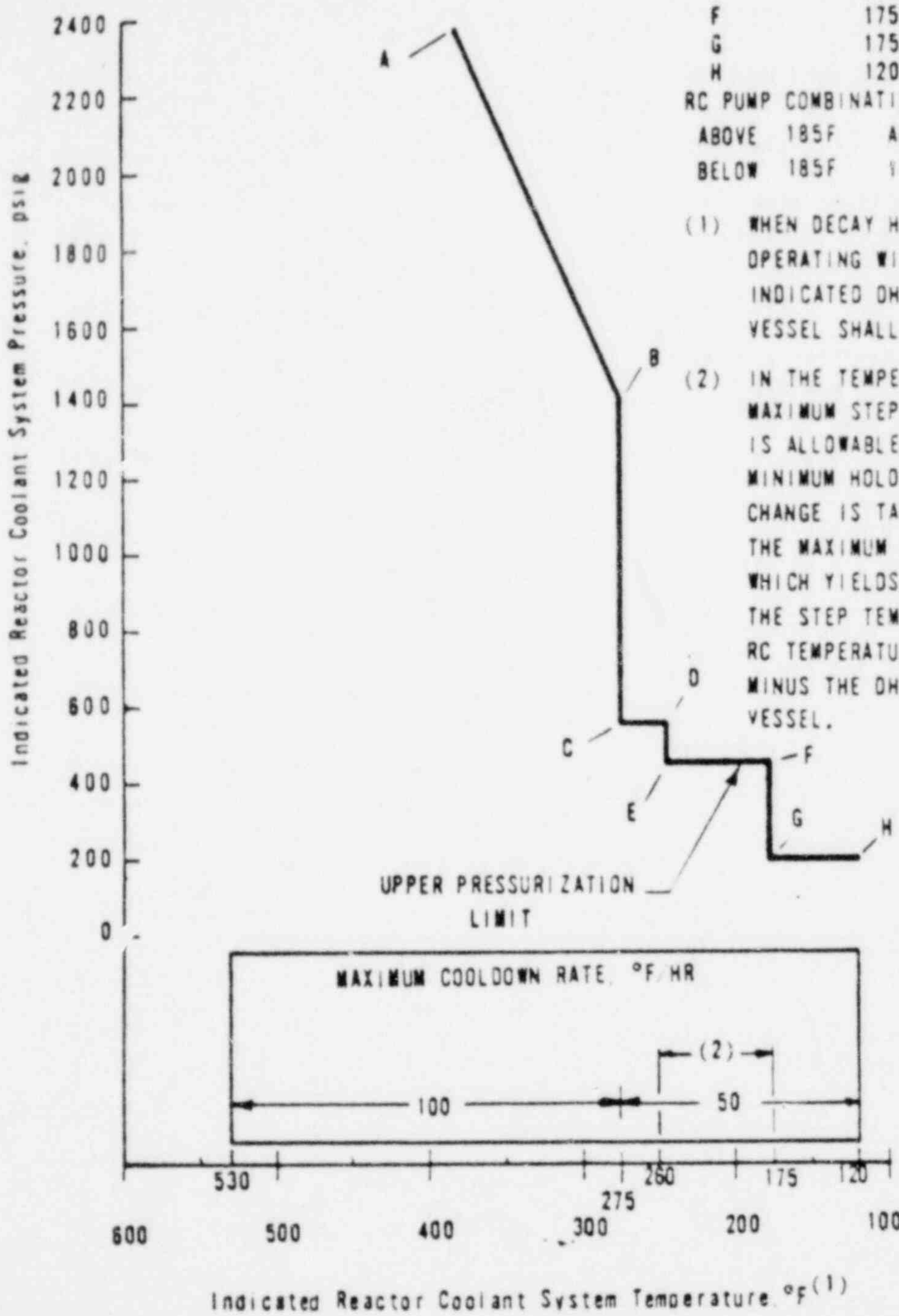
| POINT | TEMP | PRESS |
|---|---|---|
| A | 380 | 2275 |
| B | 275 | 1400 |
| C | 275 | 550 |
| D | 250 | 550 |
| E | 250 | 450 |
| F | 175 | 450 |
| G | 175 | 200 |
| H | 120 | 200 |

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 |

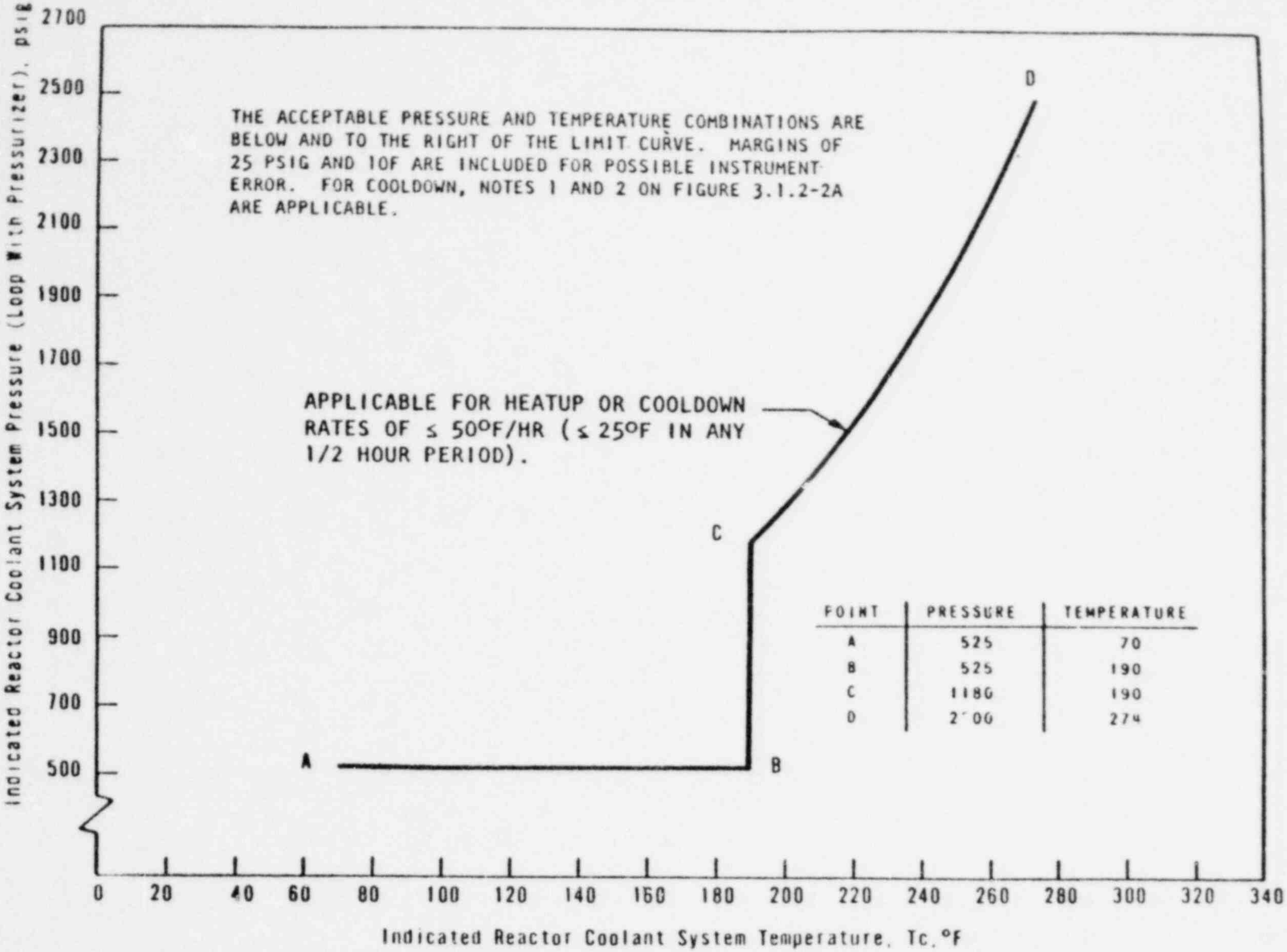
(1) WHEN DECAY HEAT REMOVAL SYSTEM (DH) IS OPERATING WITHOUT ANY RC PUMPS OPERATING, INDICATED DH RETURN TEMP. TO THE REACTOR VESSEL SHALL BE USED.

(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)

Units 2 & 3
OCONEE NUCLEAR STATION
Figure 3.1.2-2B




| POINT | PRESSURE | TEMPERATURE |
|---|---|---|
| A | 525 | 70 |
| B | 525 | 190 |
| C | 1180 | 190 |
| D | 2[illegible]00 | 274 |

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



Amendments Nos. 38, 38 & 35

3.1.3 Minimum Conditions for Criticality

Specification

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 (Units 2 and 3).

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%Δk/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.

Bases

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 0.1Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(1) and the small integrated Δk/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.

REFERENCES


(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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
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 October 1, 1975, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.38 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 23, 1977

UNITED STATES
NUCLEAR 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. 35
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 October 1, 1975, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 35 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its 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: February 23, 1977