NRC FORM 195
(2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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FROM: Duke Power Company
Charlotte, North Carolina
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PLANT NAME: Oconee Units 1-2-3

RJL 6/17/77

ENCLOSURE

Amdt. to OL/changes to Appendix A tech specs...notorixed 6/6/77...to incorporate changes to the pressurization heatup and cooldown limitations, based on the attached report, "Analysis of Capsule OC11-C from Duke Power Company Oconee Unit 2 Reactor Vessel Materials Surveillance Program"......
BAW-1437 - MAY 1977

(1/4")

See Rpts ON 50-270

FOR ACTION/INFORMATION

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NRC FORM 195 (2-76)

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

June 6, 1977

Regulatory Docket File

RECEIVED
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U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

Pursuant to 10CFR50, §50.90, please find attached a proposed amendment to the Technical Specifications for the Oconee Nuclear Station, Appendix A to Facility Operating Licenses DPR-38, -47, and -55. This proposed amendment incorporates changes to the pressurization heatup and cooldown limitations for Oconee Unit 2. These changes are based on the attached report, "Analysis of Capsule OC11-C from Duke Power Company Oconee Unit 2 Reactor Vessel Materials Surveillance Program," BAW-1437, June, 1977.

Replacement pages for the Oconee Nuclear Station Technical Specifications are attached. Proposed changes are identified by vertical lines in the margins.

Very truly yours,

William O. Parker, Jr.

MST:ge

Attachments

~~7717100011~~

7717100011

WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Facility Operating Licenses DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker

William O. Parker, Jr., Vice President

Subscribed and sworn to before me this 6th day of June, 1977.

Vivian B. Robbins

Notary Public

My Commission Expires:
My Commission Expires February 15, 1982

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 described in Specification 4.2.9. These revised limits shall be submitted to the NRC at least 90 days prior to exceeding four (Unit 1) effective full power years of operation or an
six (Unit 2)
integrated exposure of 1.7 x 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, 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 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 3.0×10^{10} n/cm^2 -- s at 2,568 MWt rated power and an integrated exposure of 3.0×10^{19} n/cm^2 for 40 years operation. (6) The calculated maximum values are 2.2×10^{10} n/cm^2 -- s and 2.2×10^{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 1.7×10^6 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 1.7×10^{18} n/cm^2 which is based on the 1.7×10^6 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 3.1.2-1C 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.

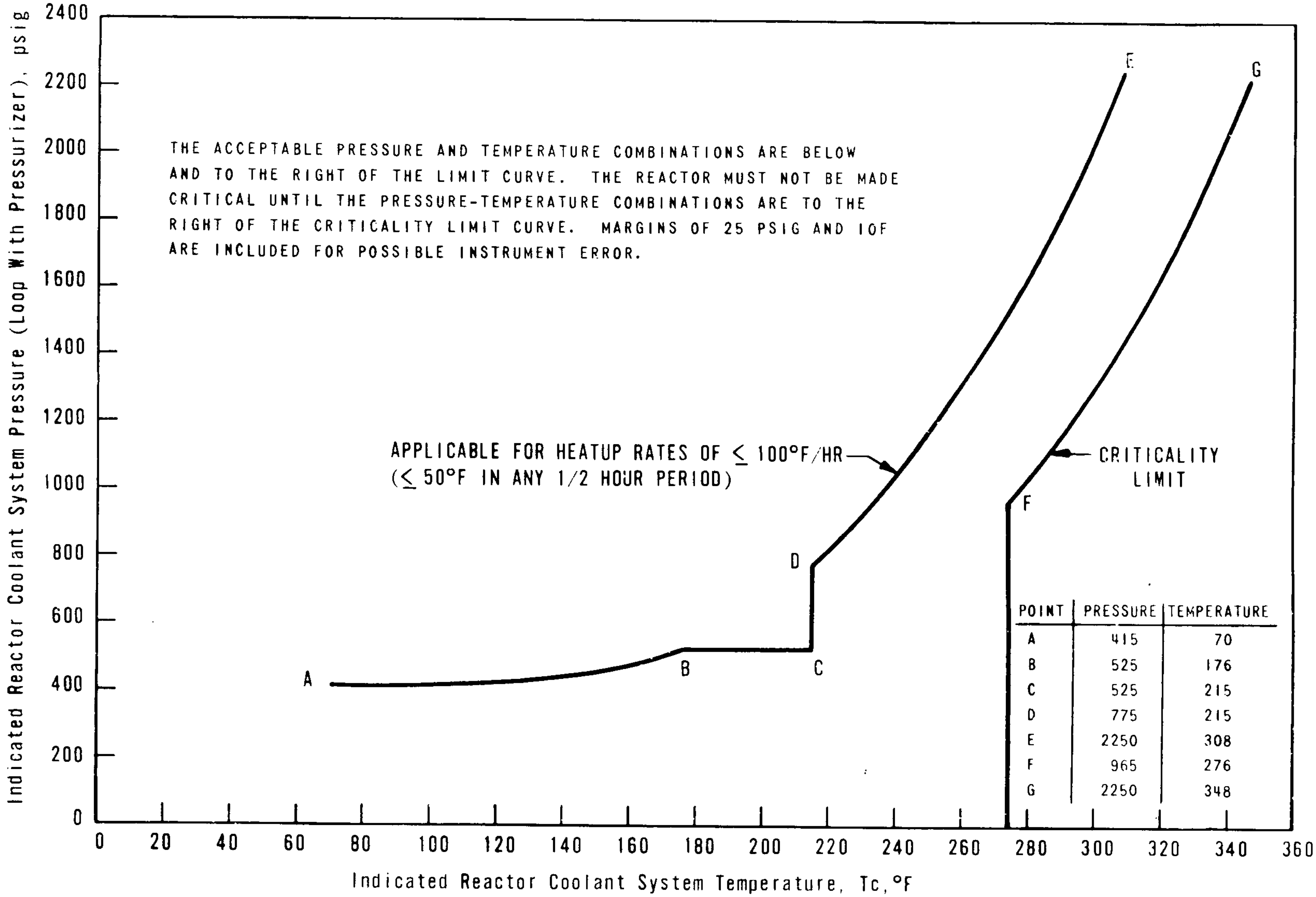
(4) FSAR Section 4.3.3.

(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, June, 1977.



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

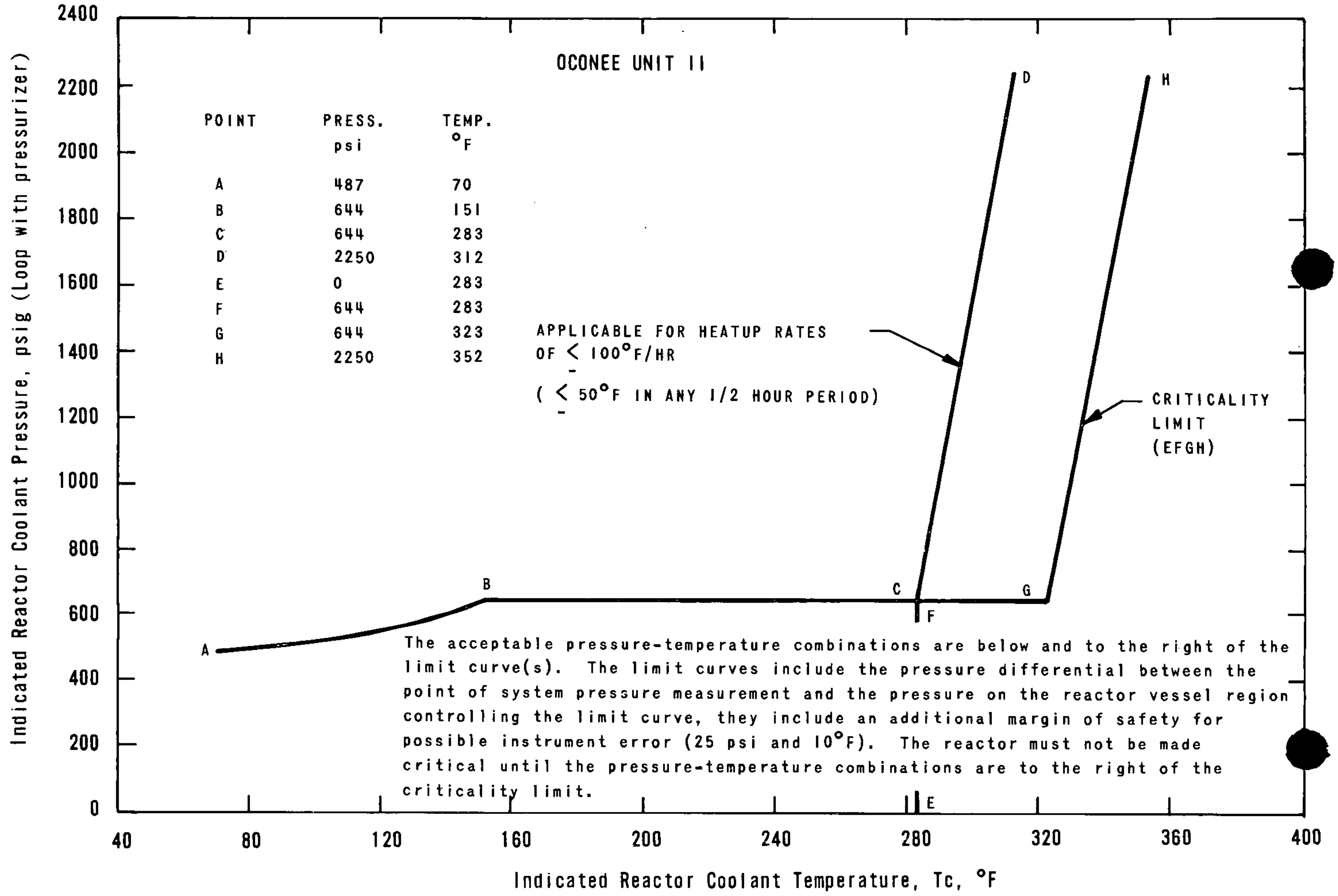
Unit 1 Only

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS, APPLICABLE FOR FIRST 4 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-1A

A: 38/38/35
2/23/77

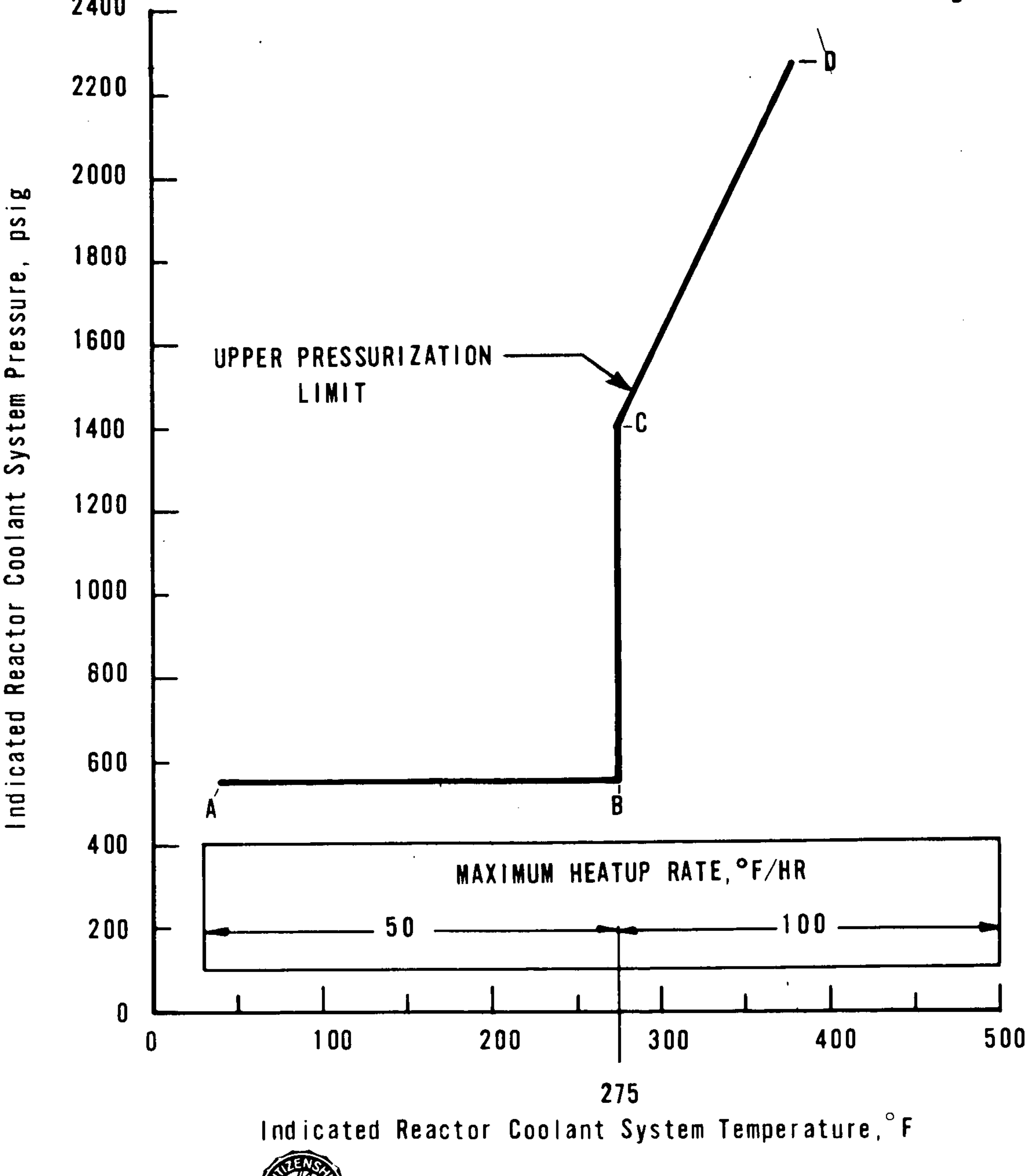


UNIT 2
REACTOR COOLANT SYSTEM
NORMAL OPERATION HEATUP LIMITATIONS
APPLICABLE FOR FIRST 6.0 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-1B

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



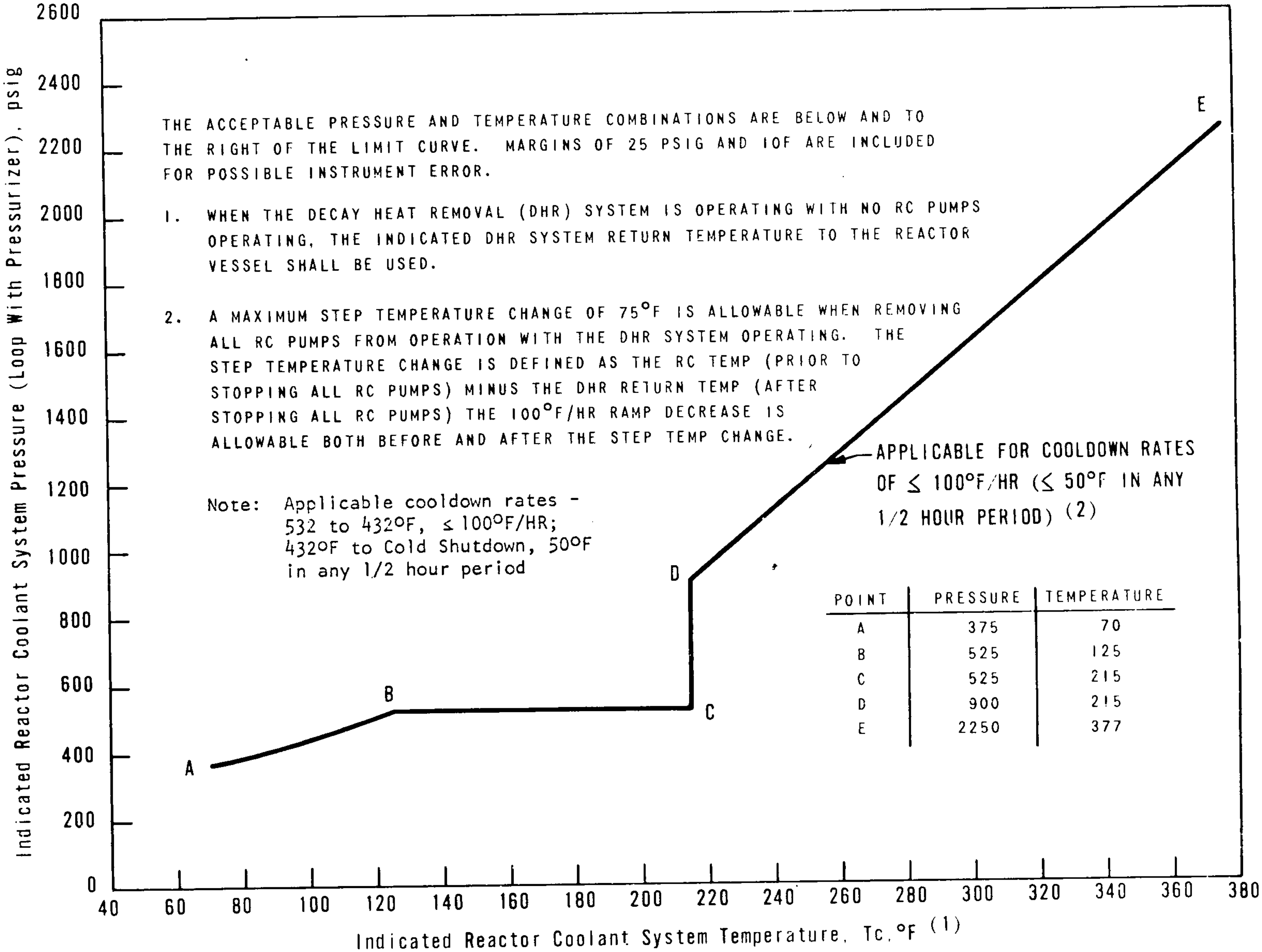
OCONEE NUCLEAR STATION

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



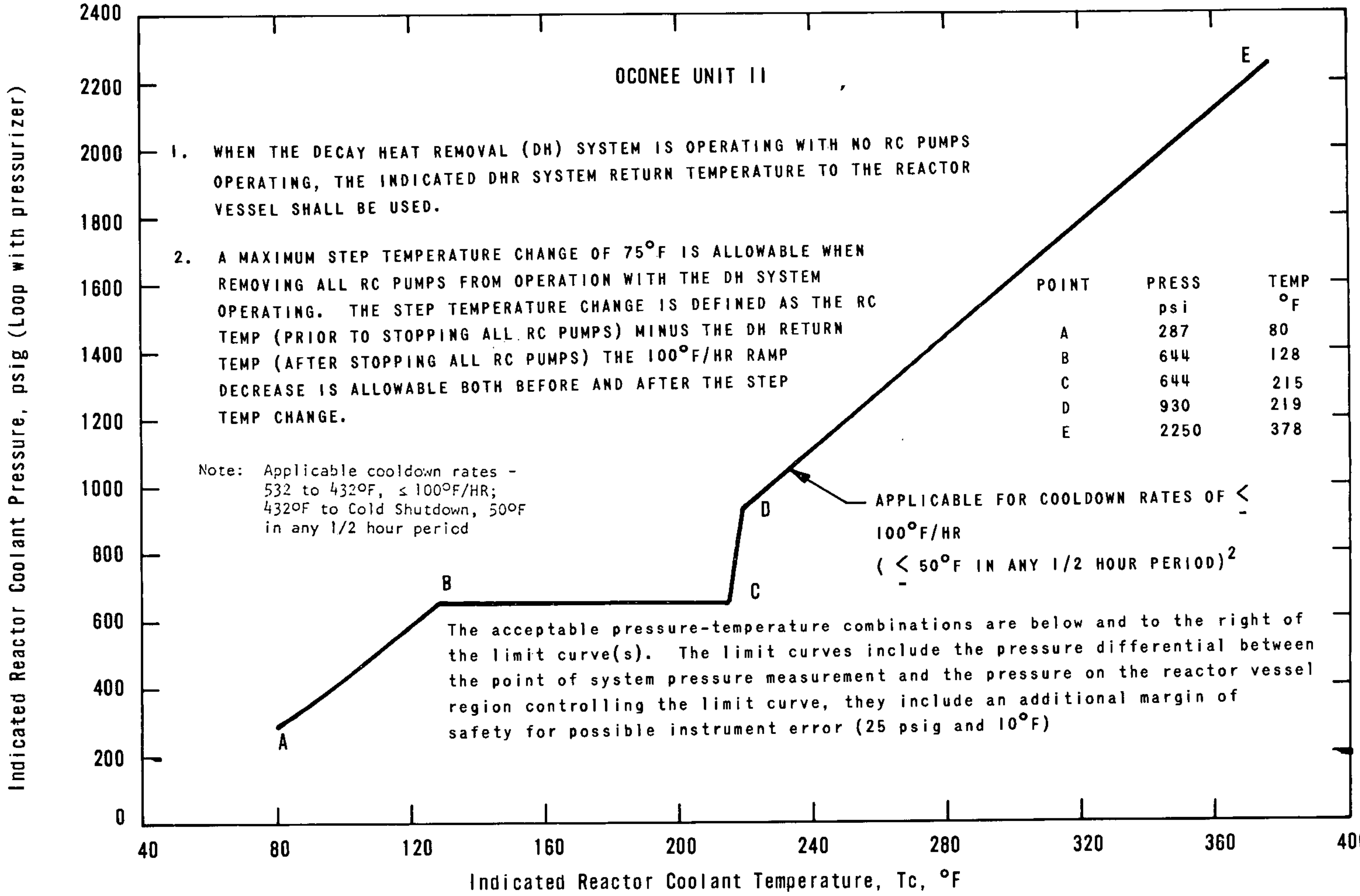
POINT	PRESSURE	TEMPERATURE
A	375	70
B	525	125
C	525	215
D	900	215
E	2250	377

Unit 1 Only

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS, APPLICABLE FOR FIRST 4 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-2A



UNIT 2
REACTOR COOLANT SYSTEM
NORMAL OPERATION COOLDOWN LIMITATIONS
APPLICABLE FOR FIRST 6.0 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

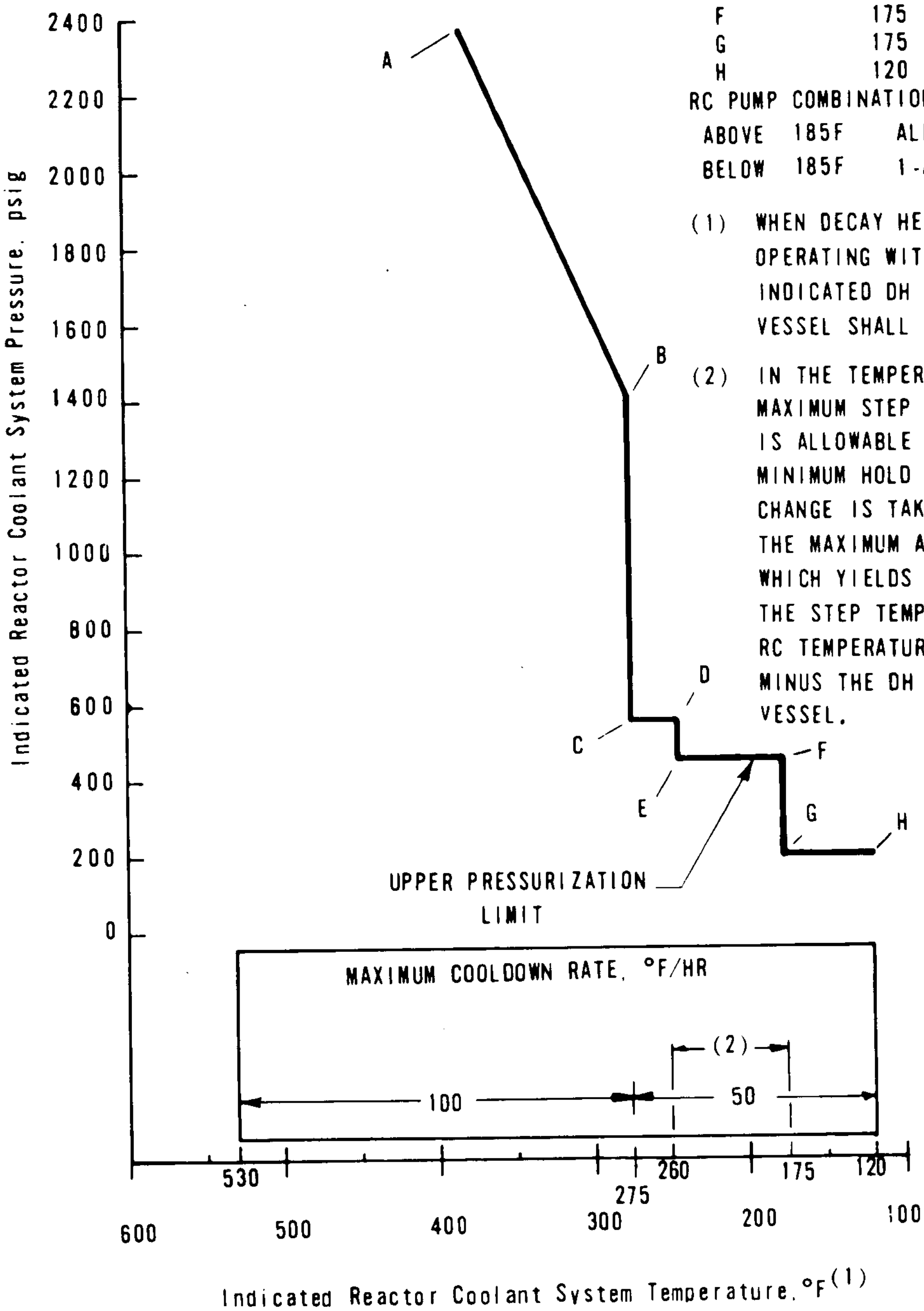
RC PUMP COMBINATIONS ALLOWABLE:

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.

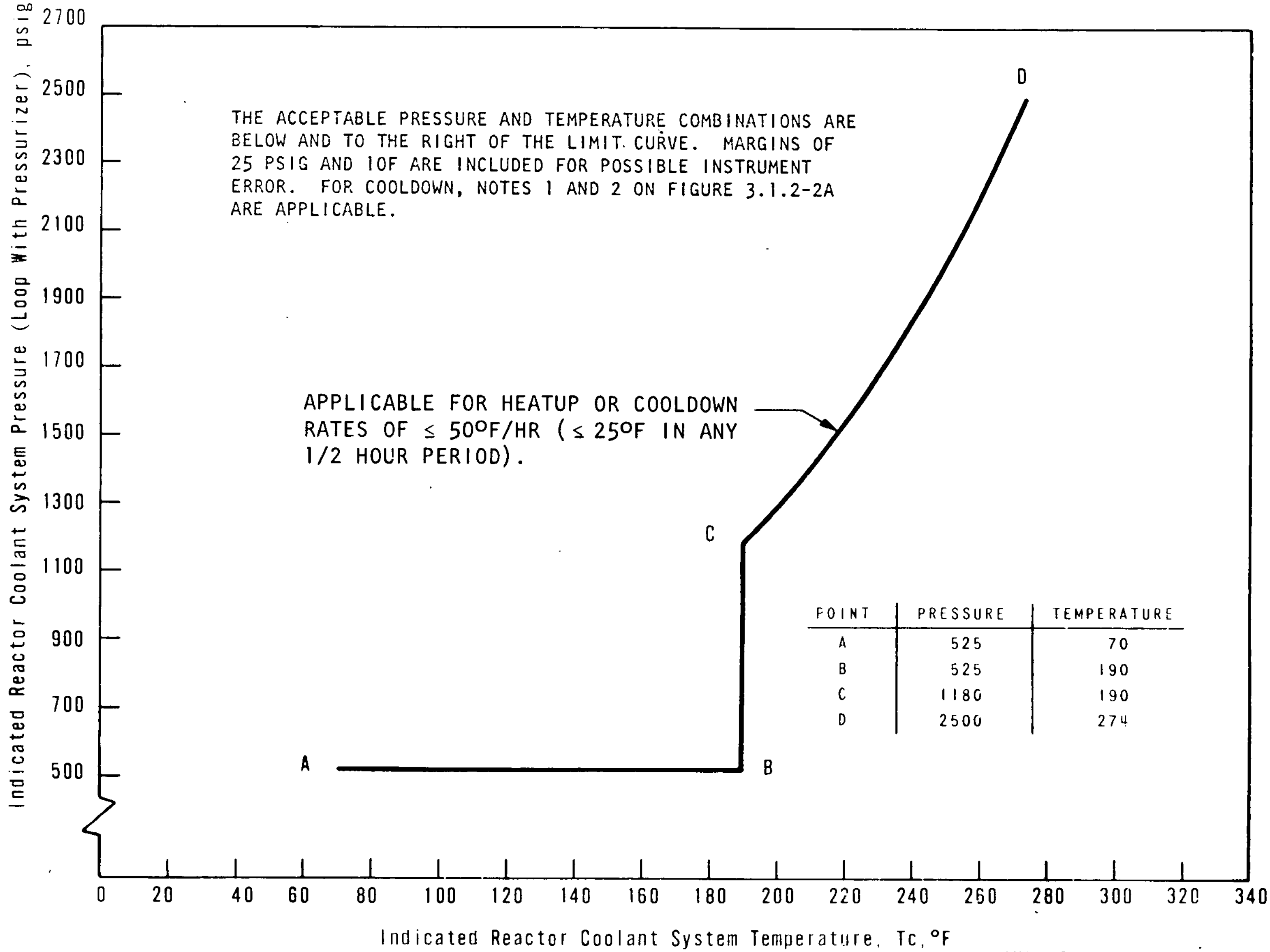


REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
(APPLICABLE UP TO DTT = 185°F)

Unit 3

DUKE POWER CITIZENSHIP SERVICE

OCONEE NUCLEAR STATION
Figure 3.1.2 - 2c



POINT	PRESSURE	TEMPERATURE
A	525	70
B	525	190
C	1180	190
D	2500	274

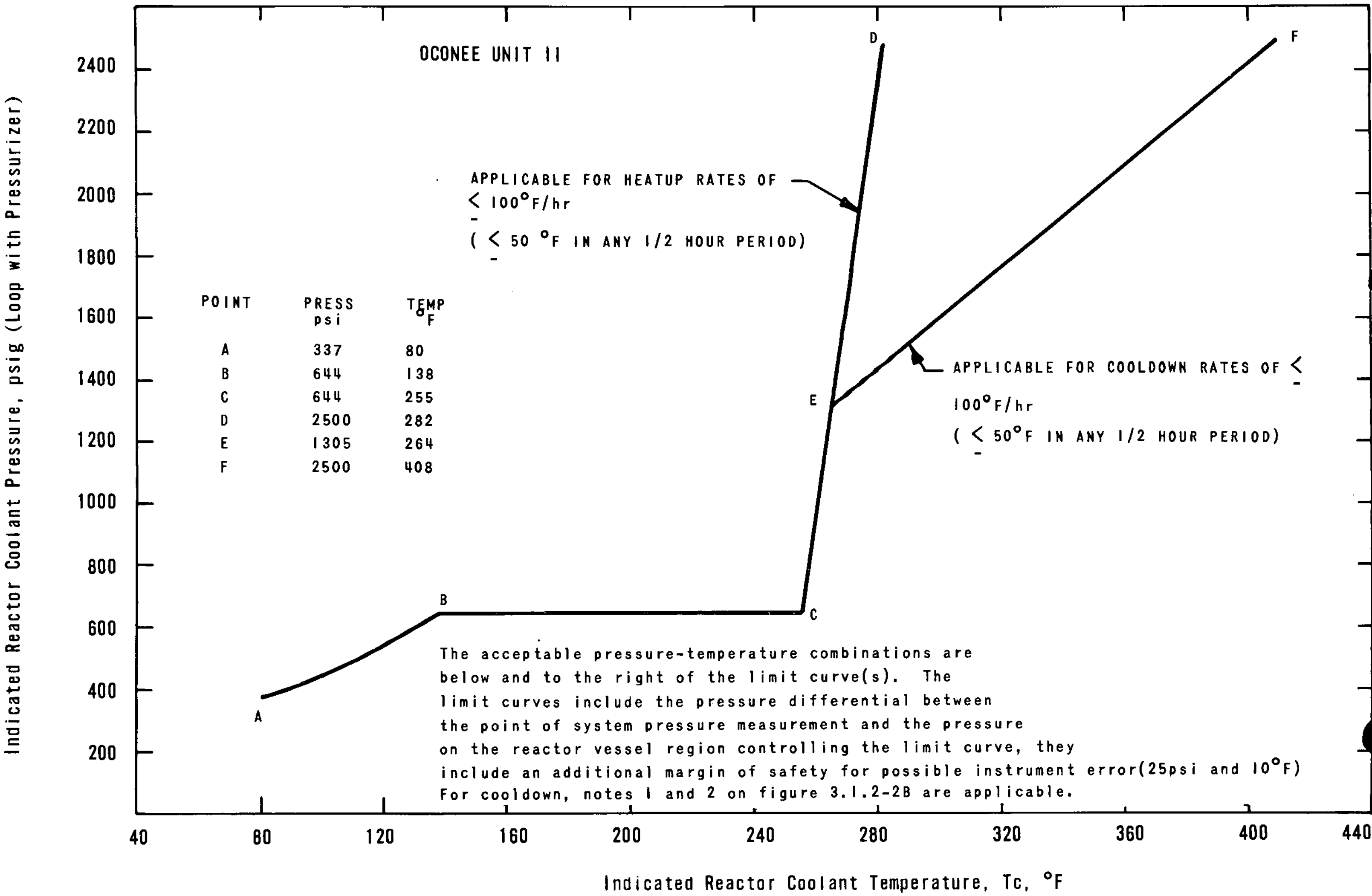
UNIT 1

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

A 38/38/35
2/23/77



UNIT 2
REACTOR COOLANT SYSTEM
INSERVICE LEAK AND HYDROSTATIC
TEST AND COOLDOWN LIMITATIONS
APPLICABLE FOR FIRST 6.0 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-3B

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 (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%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