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TO: Mr Rusche		ORIG none signed	CC	OTHER	SENT NRC PDR <u>XX</u>		SENT LOCAL PDR <u>XX</u>
CLASS	UNCLASS XXXXX	PROP INFO	INPUT	NO CYS REC'D 1	DOCKET NO: <u>50-269</u> /270/287		

DESCRIPTION:
Ltr notatized (not signed) 10-1-75.....
trans the following:
ACKNOWLEDGED
DO NOT REMOVE
PLANT NAME: Oconee 1, 2, & 3

ENCLOSURES:
Proposed Amdt to OL/Change to Tech Specs:
consisting of revised & addl pgs and info
concerning pressurization heatup & cooldown
limitations and to the reactor vessel material
surveillance program.....(40 cys encl rec'd)

FOR ACTION/INFORMATION 10-6-75 ehf


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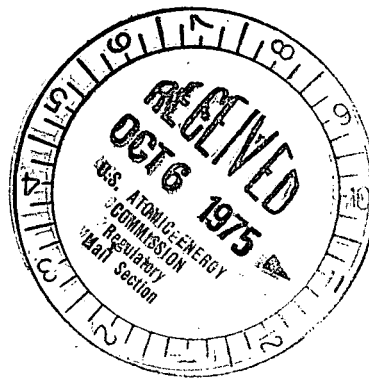
WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

October 1, 1975

TELEPHONE: AREA 704
373-4083

Mr. Benard C. Rusche, 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



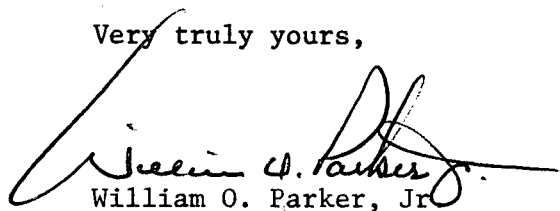
Dear Mr. Rusche:

Pursuant to 10CFR50, §50.90, please find attached a proposed amendment to the Technical Specifications for Oconee Nuclear Station, Appendix A to Facility Operating Licenses DPR-38, -47 and -55. This proposed amendment incorporates changes to the pressurization, heatup and cool-down limitations, and to the reactor vessel material surveillance program. These changes are based on the report, "Analysis of Capsule OC1-F from Duke Power Company Oconee Unit 1 Reactor Vessel Materials Surveillance Program", BAW-1421 Rev. 1, September 1975.

Attached are replacement pages for the Oconee Nuclear Station Technical Specifications. Proposed changes are identified by vertical lines in the margin of these replacement pages.

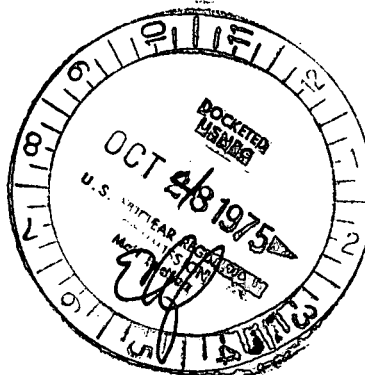
Forty copies of this request, including three signed originals are enclosed.

Very truly yours,


William O. Parker, Jr.

ROS:ge

Enclosure



~~21502~~

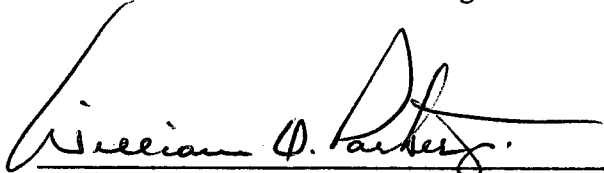
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Mr. Benard C. Rusche

Page 2

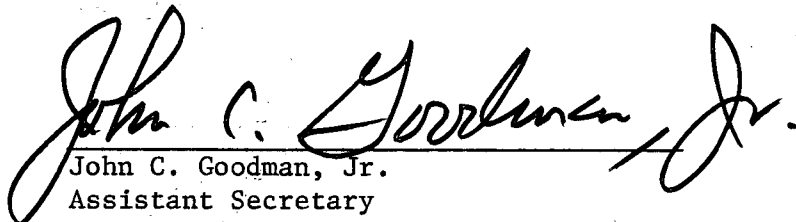
October 1, 1975

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 Technical Specifications, Appendix A to 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, Jr., Vice President

ATTEST:



John C. Goodman, Jr.
Assistant Secretary

Subscribed and sworn to before me this 1st day of October, 1975.



Notary Public

My Commission Expires:

October 24, 1977

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 is 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 3.0×10^{10} n/cm² -- s at 2,568 MWt rated power and an integrated exposure of 3.0×10^{19} n/cm² for 40 years operation. (6) The calculated maximum values are 2.2×10^{10} n/cm² -- s and 2.2×10^{19} n/cm² 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 1.7×10^6 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/cm² 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 3.1.2-1B 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:

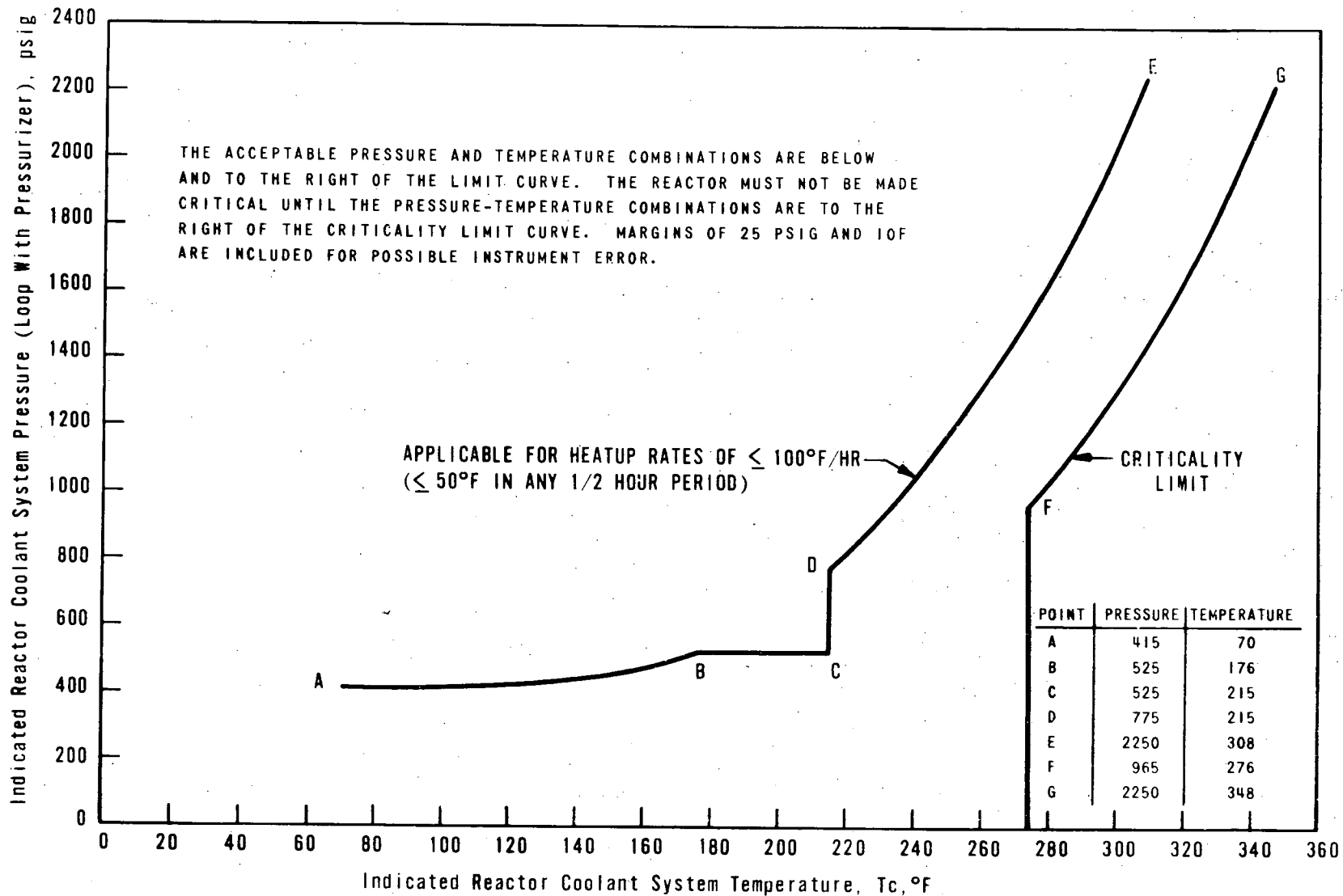
1. A 25 psi error is measured pressure.
2. System pressure is measured in either loop.
3. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations.

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 OCl-F from Duke Power Company Oconee Unit 1 Reactor Vessel Materials Surveillance Program, BAW-1421 Rev. 1, September 1975.



Unit 1 Only

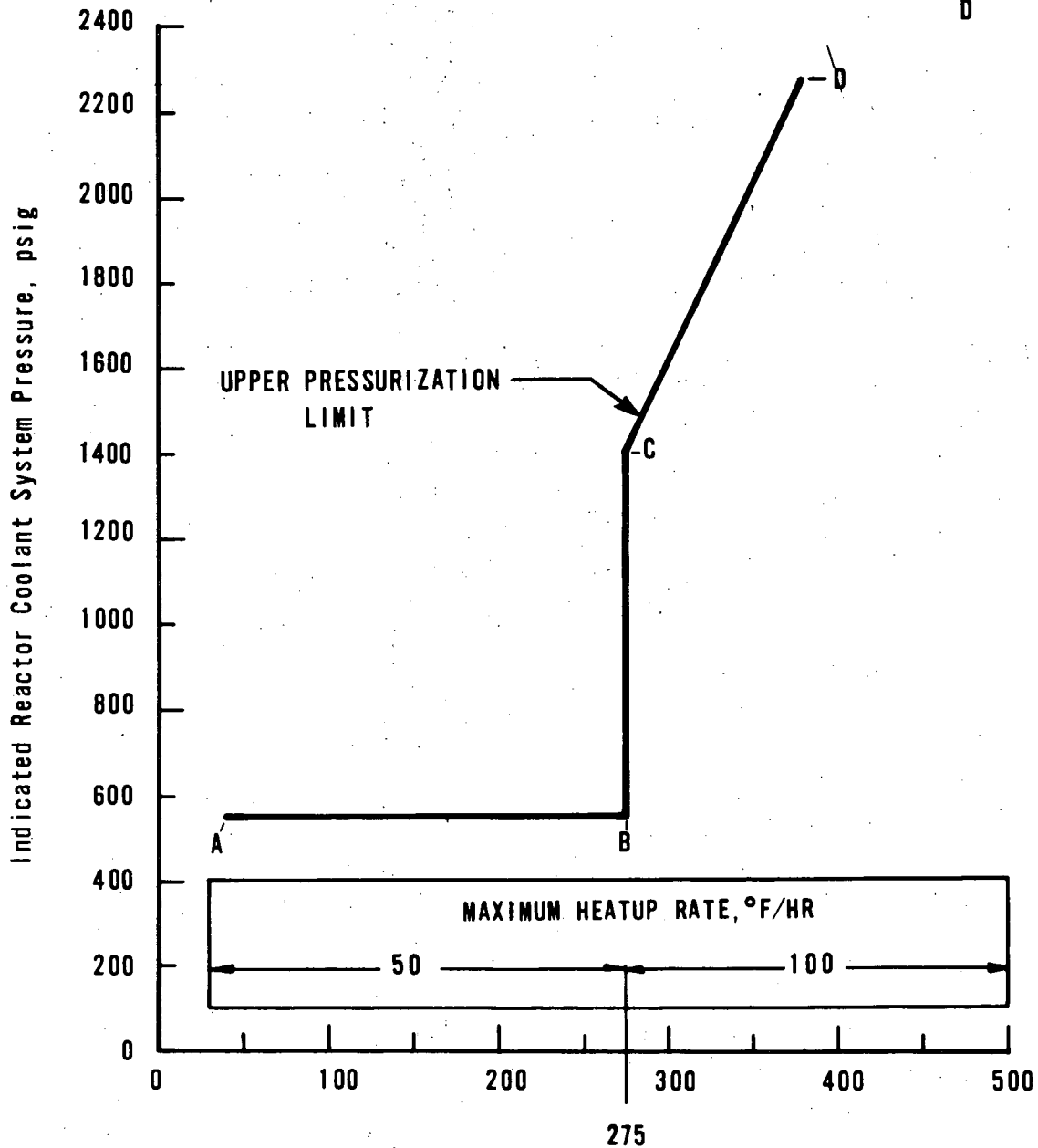
REACTOR COOLANT SYSTEM HEATUP
LIMITATIONS, APPLICABLE FOR
FIRST 5 EPFY

OCONEE NUCLEAR STATION

Figure 3.1.2-1A



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



Indicated Reactor Coolant System Temperature, °F



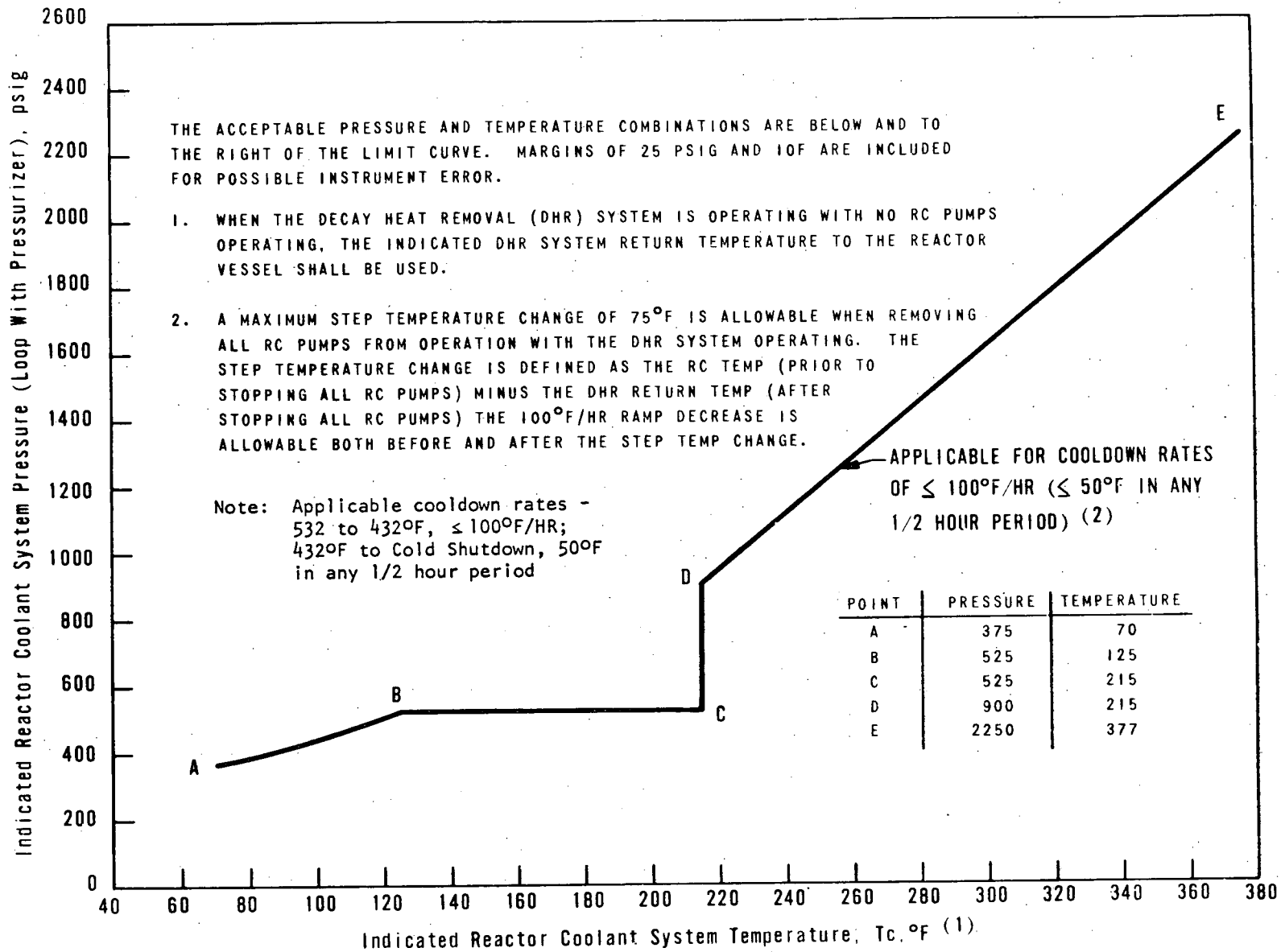
OCONEE NUCLEAR STATION

Units 2 & 3

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS

(APPLICABLE UP TO AN INTEGRATED EXPOSURE OF 1.7×10^{18} n/cm² OR DTT - 144 °F)

Figure 3.1.2-1B



Unit 1 Only
 REACTOR COOLANT SYSTEM COOLDOWN
 LIMITATIONS, APPLICABLE FOR
 FIRST 5 EFPY



OCONEE NUCLEAR STATION

Figure 3.1.2-2A

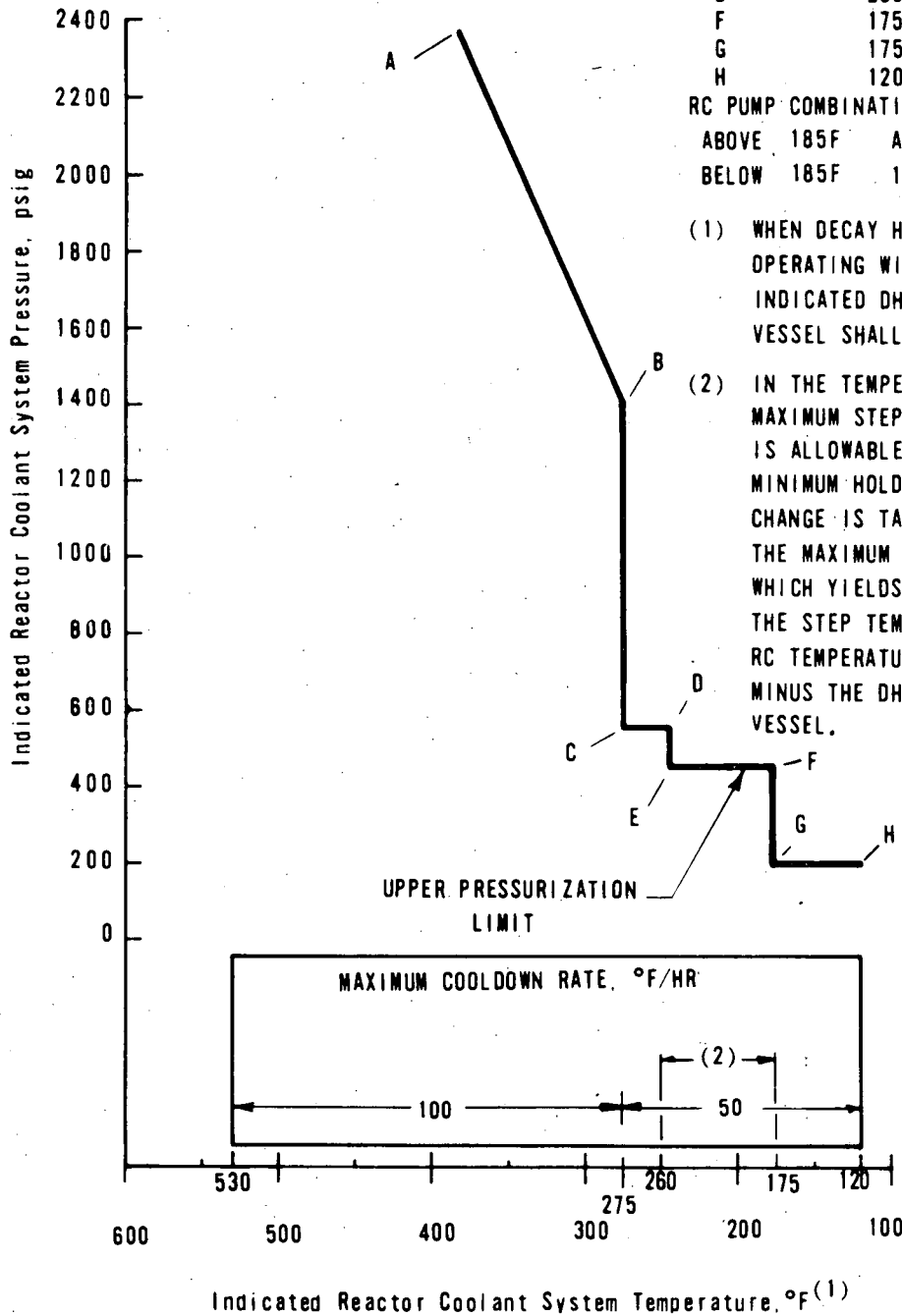
POINT	TEMP	PRESS
A	380	2275
B	275	1400
C	275	550
D	250	550
E	250	450
F	175	450
G	175	200
H	120	200

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.

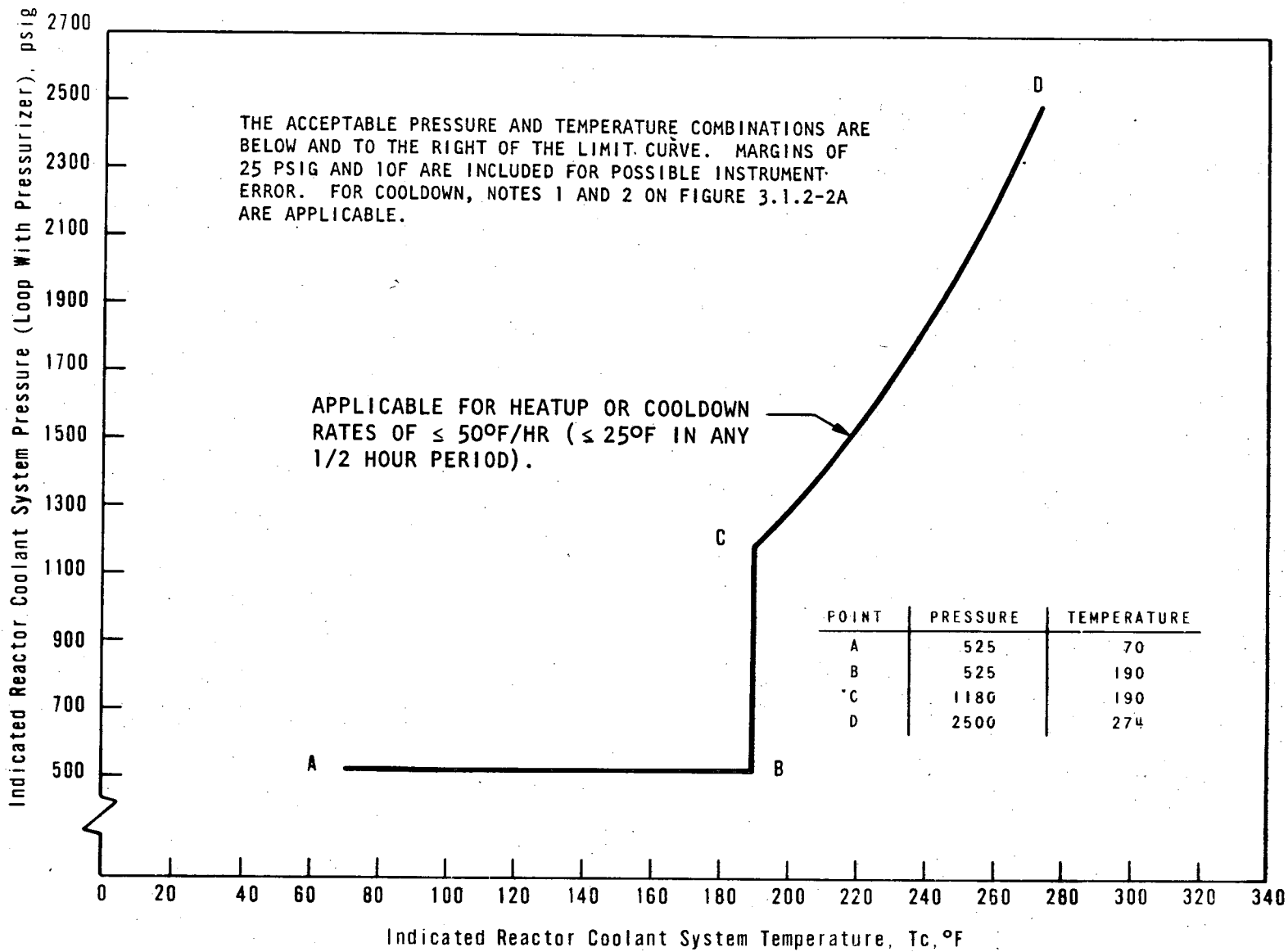


REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
(APPLICABLE UP TO DTT = 185°F)



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

3.1-7b



REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMITATIONS FOR INSERVICE HYDROSTATIC TESTS (NO FUEL ASSEMBLIES IN THE CORE), APPLICABLE FOR FIRST 5 EFPY



OCONEE NUCLEAR STATION

Figure 3.1.2-3

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% $\Delta k/k$ 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.

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 0.1 $\Delta k/k$.

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.

REFERENCES

- (1) FSAR, Section 3
- (2) FSAR, Section 3.2.2.1.4
- (3) FSAR, Supplement 3, Answer 14.4.1

- 4.2.3 The structural integrity of the Reactor Coolant System boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated, including evaluation of comparable areas of the Reactor Coolant System.
- 4.2.4 To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.
- 4.2.5 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.
- 4.2.6 The inservice inspection program shall be reviewed at the end of five years to consider incorporation of new inspection techniques and equipment which have been proved practical and the conclusions of this review and evaluation shall be discussed with the AEC/DOL.
- 4.2.7 At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an in-place, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed, if the interval measured from the previous such inspections is greater than 6 2/3 years.
- 4.2.8 Reactor vessel material surveillance capsules shall be withdrawn after 1,8,16 and 24 effective full power years of operation. The withdrawal schedules may be modified to coincide with those refueling outages most closely approaching the withdrawal schedule. Specimens thus withdrawn shall be tested in accordance with Appendix G of 10 CFR 50. A report of the test results shall be forwarded to the Commission in accordance with Specification 6.6.1.6.
- 4.2.9 During the first two refueling periods, two reactor coolant system piping elbows shall be ultrasonically inspected along their longitudinal welds (4 inches beyond each side) for clad bonding and for cracks in both the clad and base metal. The elbows to be inspected are identified in B&W Report 1364 dated December 1970.

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, 1970, including 1970 winter addenda, edition. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

The reactor vessel material surveillance program is based on the requirement of Appendix H to 10 CFR 50 and is described in B&W report BAW-10006A, Rev 3. The capsule withdrawal schedule is based on the following:

First capsule - At the time when the predicted shift of the adjusted reference temperature is approximately 50°F

Second and third capsules - At approximately one-third and two-thirds of the time interval between first and fourth capsule withdrawal.

Fourth capsule - Three-fourths of service life.

Fifth capsule - Standby

The specific capsule withdrawn at each of these intervals will be based on current need in best defining the material properties of the reactor vessel.

Early inspection of Reactor Coolant System piping elbows is considered desirable in order to reconfirm the integrity of the carbon steel base metal when explosively clad with sensitized stainless steel. If no degradation is observed during the two annual inspections, surveillance requirements will revert to Section XI of the ASME Boiler and Pressure Vessel Code.