NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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DOCKET NUMBER: 50-269/270/287

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TO:	FROM:	DATE OF DOCUMENT / DATE RECEIVED
Mr. Edson G. Case	Duke Power Co. Charlotte, N. C. 28242 William O. Parker	DATE OF DOCUMENT: 09/14/77 DATE RECEIVED: 09/19/77

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DESCRIPTION: Ltr. Notorized 09/14/77...Trans The Following:

2p

ENCLOSURE: License No. DPR-38, 47, and 55 to Appl for Amend: tech specs proposed change concerning changes to the pressurization, heatup and cooldown limitations for Oconne Unit # 3...w/att Rept entitled: "Analysis of Capsule OCIII-A From Duke Power Co. Oconne Nuc. Station, Unit # 3 - Reactor Vessel Materials Surveillance Program - BAW-1438 - July 1977...

1/4"

DISTRIBUTION FOR MATERIAL ON REACTOR VESSEL DATA PER R. INGRAM 5-26-77

PLANT NAME: OCONEE UNITS 1-3
jcm 09/19/77

40 ENCL

FOR ACTION/INFORMATION

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

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

September 14, 1977

TELEPHONE: AREA 704
373-4083

Mr. Edson G. Case, Acting 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 3. These changes are based on the attached report, "Analysis of Capsule OCIII-A from Duke Power Company Oconee Unit 3 Reactor Vessel Materials Surveillance Program", BAW-1438, July, 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.

LJB:ge

Attachments

7726200 73

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 14th day of September, 1977.

Vivian B. Robbins

Notary Public

My Commission Expires:

Feb. 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 test 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
3.1.2-3C Unit 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 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.
six (Unit 2)
eight (Unit 3)

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[1], BAW-1437[2] and BAW-1438[3].

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 the 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.

REFERENCES

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

(2) Analysis of Capsule OC2-1C from Duke Power Company Oconee Unit 2 Reactor Vessel Materials Surveillance Program, BAW-1437, April, 1977.

(3) Analysis of Capsule OCIII-A from Duke Power Company Oconee Unit 3 Reactor Vessel Materials Surveillance Program, BAW-1438, July, 1977.

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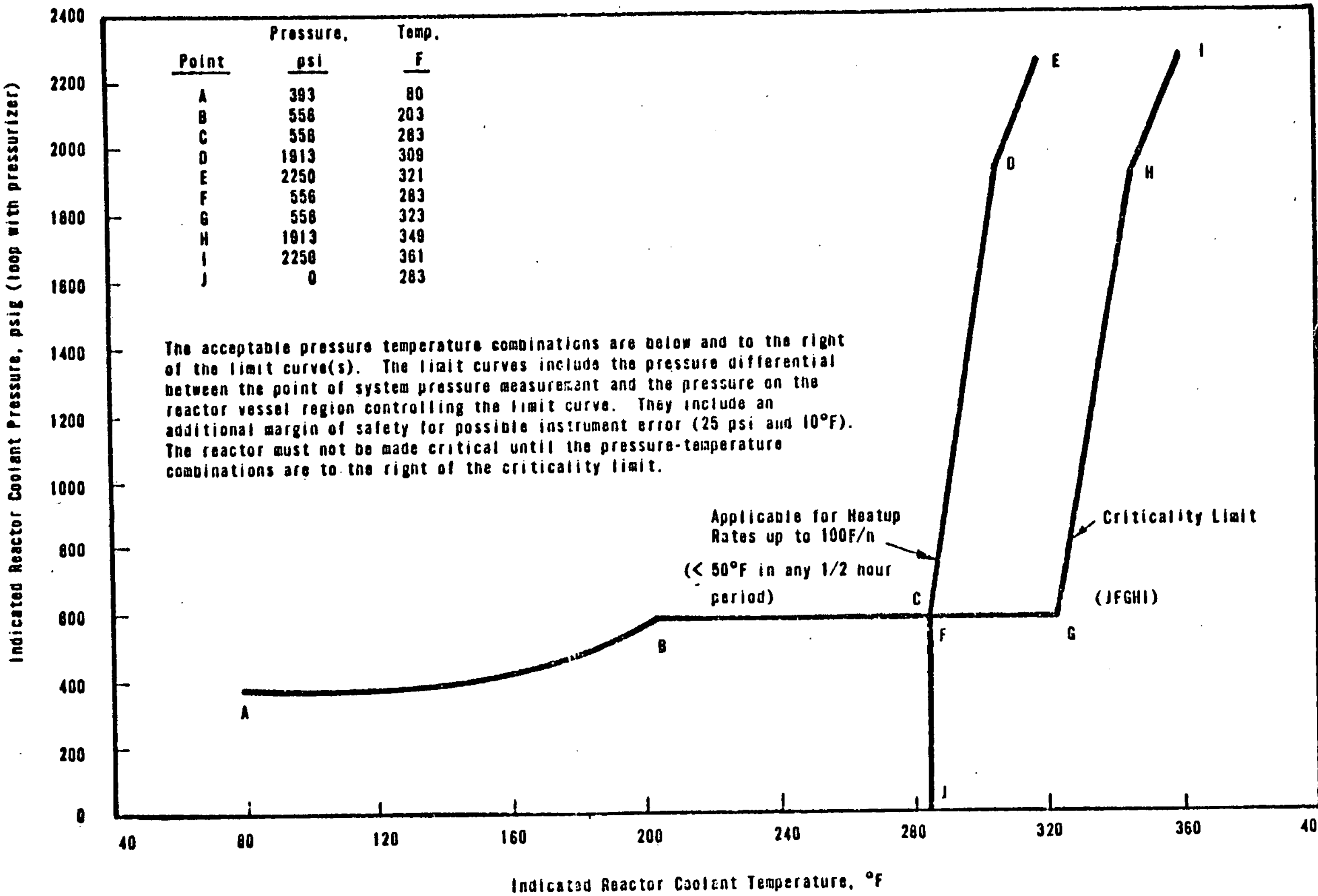
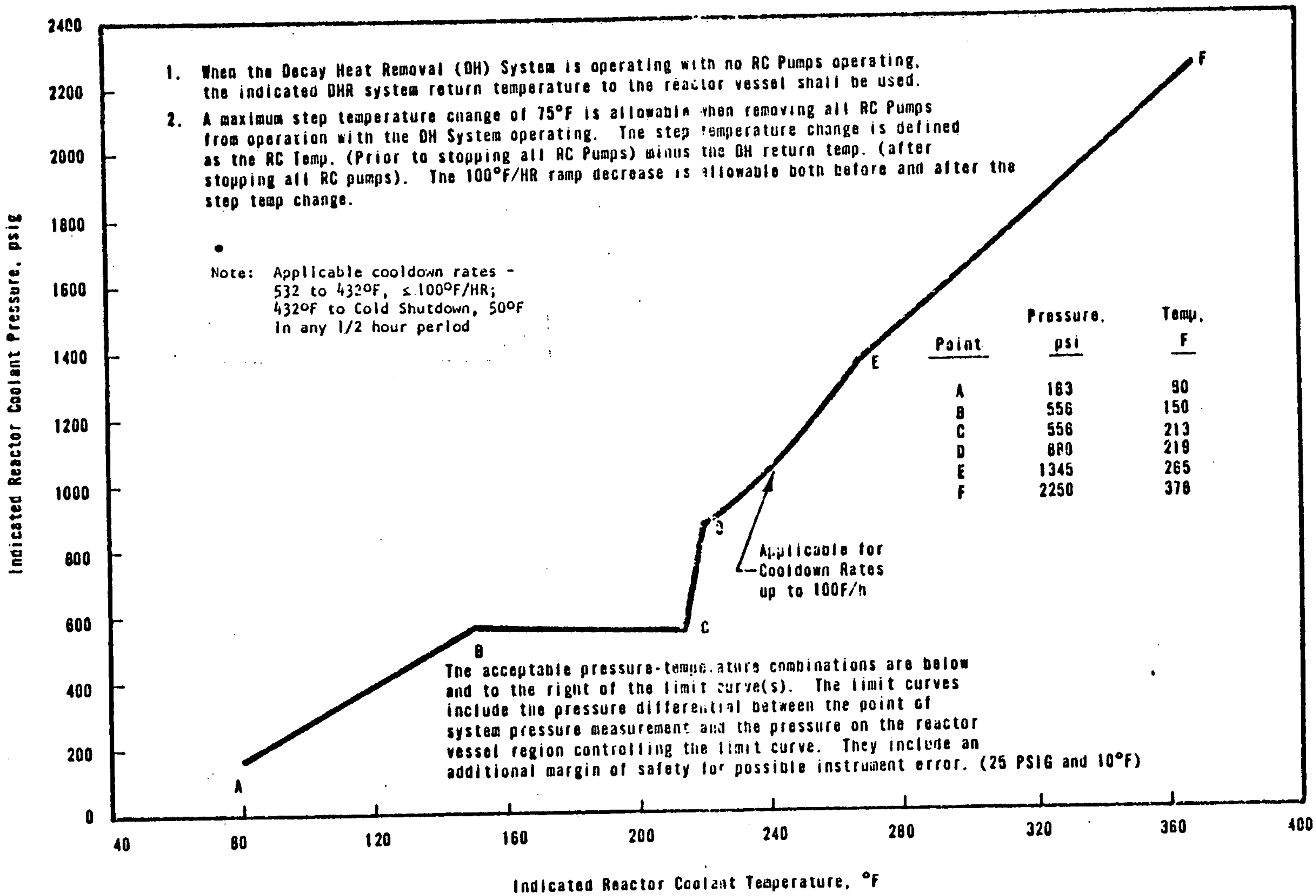


Point	Pressure, psi	Temp. F
A	393	80
B	556	203
C	556	283
D	1913	309
E	2250	321
F	556	283
G	556	323
H	1913	349
I	2250	361
J	0	283

UNIT 3
REACTOR COOLANT SYSTEM
NORMAL OPERATION HEATUP LIMITATIONS
APPLICABLE FOR FIRST 8.0 EFPY

OCONEE NUCLEAR STATION

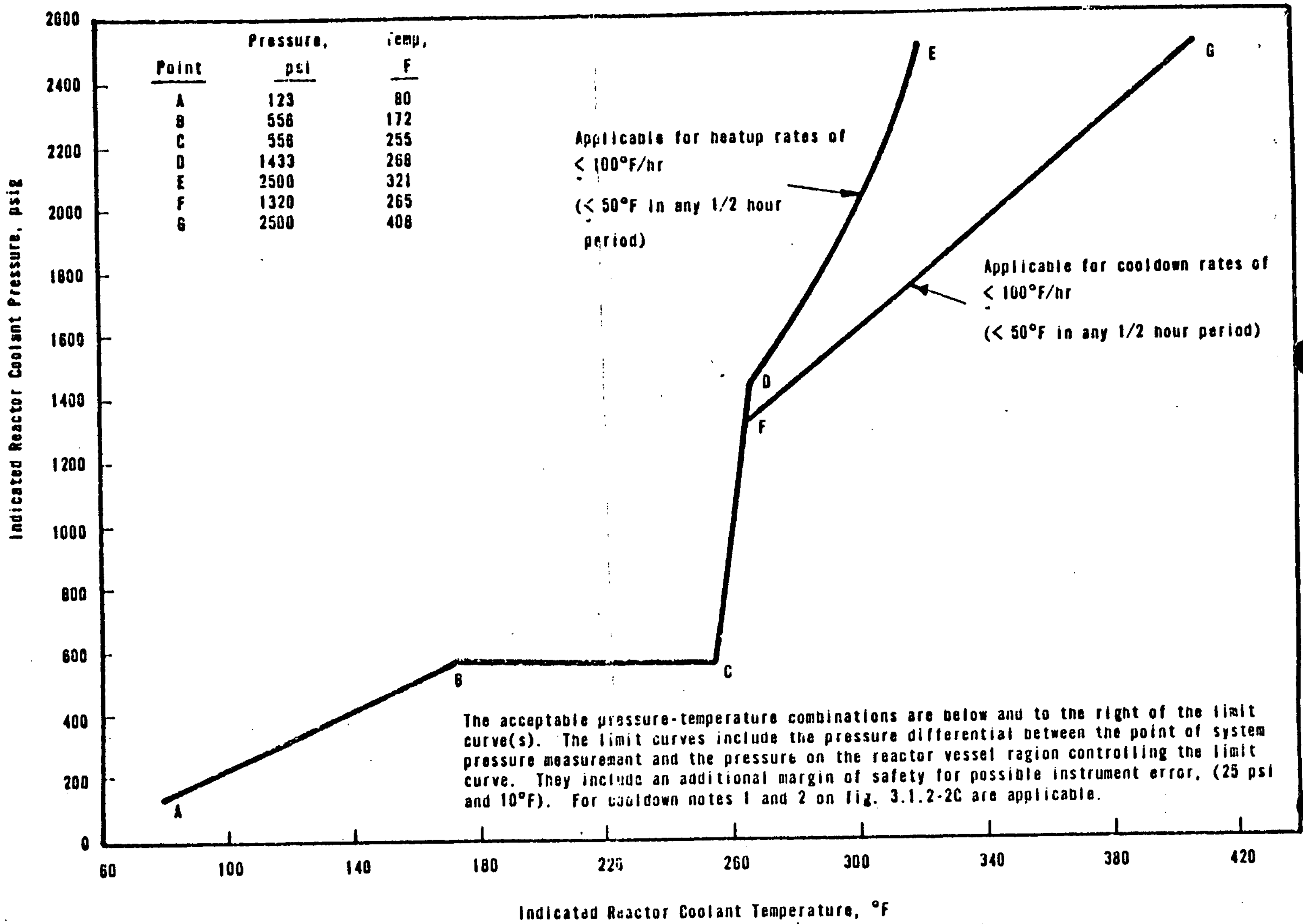
Figure 3.1.2-1C



UNIT 3
REACTOR COOLANT SYSTEM
NORMAL OPERATION COOLDOWN LIMITATIONS
APPLICABLE FOR FIRST 8.0 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-2C



UNIT 3
REACTOR COOLANT SYSTEM
INSERVICE LEAK AND HYDROSTATIC
TEST AND COOLDOWN LIMITATIONS
APPLICABLE FOR FIRST 8.0 EFPY

OCONEE NUCLEAR STATION

Figure 3.1.2-3C

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)
3.1.2-1B (Unit 2)
3.1.2-1C (Unit 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