May 14, 1991

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

Distribution
See next page

Mr. M.S. Tuckman
Vice President -
Nuclear Operations
Duke Power Company
P.O. Box 1007
Charlotte, North Carolina 28201-1007

Dear Mr. Tuckman:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 188, 188 AND 185 TO FACILITY OPERATING LICENSES DPR-38, DPR-47 AND DPR-55 - OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 (TACS 75344, 75345, 75346)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 188 , 188 and 185 to Facility Operating Licenses 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 Station's TSs in response to your request dated November 15, 1989, as supplemented March 28 and August 29, 1990.

The amendments revise the Pressure/Temperature Limits and the Low Temperature Overpressure Protection system operability requirements in Section 3.1 of the TSs and revise the associated bases.

A copy of the related Safety Evaluation is also enclosed. Notice of issuance of the enclosed amendments will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/S/

Leonard A. Wiens, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 188 to DPR-38
2. Amendment No. 188 to DPR-47
3. Amendment No. 185 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

LA:PDII-3	PE:PDII-3	PM:PDII-3	SRXB	EMCB	OGC	D:PDII-3
BClayton	FRinaldi	LWiens	RJones	CCheng		DMatthews
4/23/91	4/24/91	4/24/91	4/25/91	4/24/91	4/26/91	5/9/91

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

May 14, 1991

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

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Vice President -
Nuclear Operations
Duke Power Company
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Charlotte, North Carolina 28201-1007

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cc w/enclosures:
See next page

Mr. M.S. Tuckman
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

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DATED: May 14, 1991

AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE DPR-38 - Oconee Nuclear Station, Unit 1
AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE DPR-47 - Oconee Nuclear Station, Unit 2
AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE DPR-55 - Oconee Nuclear Station, Unit 3

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DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated November 15, 1989, as supplemented March 28 and August 29, 1990, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-38 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 188, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Matthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: May 14, 1991

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated November 15, 1989, as supplemented March 28 and August 29, 1990, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-47 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 188, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Matthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: May 14, 1991

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 185
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated November 15, 1989, as supplemented March 28 and August 29, 1990, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-55 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 185, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: May 14, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 185

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
3.1-3a through 3.1-7e	3.1-3a through 3.1-7e

3.1.2.7 Prior to exceeding fifteen (Unit 1)
fifteen (Unit 2)
fifteen (Unit 3)

effective full power years of operation.

Figures 3.1.2-1A (Unit 1), 3.1.2-2A (Unit 1)
3.1.2-1B (Unit 2), 3.1.2-2B (Unit 2)
3.1.2-1C (Unit 2), 3.1.2-2C (Unit 3)

and 3.1.2-3A (Unit 1)
3.1.2-3B (Unit 2)
3.1.2-3C (Unit 3)

and Technical Specification 3.1.2.1, 3.1.2.2 and 3.1.2.3 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G, Section V.B. and V.E.

3.1.2.8 The updated proposed technical specification referred to in 3.1.2.7 shall be submitted for NRC review at least 90 days prior to the end of the service period for Units 1, 2 and 3.

3.1.2.9 Two trains of the low temperature overpressure protection (LTOP) system shall be operable.

1. One LTOP train is comprised of the PORV with a lift setting of $\leq$ 500 psig for Units 1 and 2, $\leq$ 480 psig for Unit 3. The PORV shall be operable when:

   a) The temperature of one or more of the RCS cold legs is $\leq$ 325°F, and

   b) When RCS pressure is $\geq$ 100 psig or HPI pumps are operating.

2. The second LTOP train is comprised of the controls which assure that 10 minutes are available for operator action to mitigate an LTOP event. The second LTOP train shall be operable when the temperature of one or more of the RCS cold legs is $\leq$ 325°F and a RCS vent path equivalent to the PORV is not open. The following controls comprise the second LTOP train:

   a) RCS pressure is limited to $\leq$ 350 psig for Units 1 and 2, $\leq$ 345 psig for Unit 3 below an RCS temperature of 220°F.

   b) Deactivating train A and B of HPI.

   c) Deactivating both core flood tanks.

   d) Pressurizer level shall be controlled such that 10 minutes are available for operator action to mitigate an LTOP event.

Amendment No. 188(Unit 1)
Amendment No. 188(Unit 2)
Amendment No. 185(Unit 3)

e) Makeup flow shall be restricted such that 10 minutes are available for operator action to mitigate an LTOP event.

f) Alarms shall be provided such that 10 minutes are available for operator action to mitigate an LTOP event.

3. a.1 If the PORV is inoperable, the PORV shall be returned to operable status or the RCS shall be heated above 325°F within 24 hours.

a.2 If the provisions of a.1 above cannot be fulfilled, the RCS shall be depressurized to < 100 psig and HPI shall be removed from service within the next 12 hours.

b. If the second LTOP train is inoperable, action shall be initiated to restore the second train to operable status and compensatory measures shall be provided to monitor for initiation of an LTOP event within 4 hours.

Bases - Units 1, 2 and 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 5.2-1 of the FSAR.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in BAW-1699 and BAW-1697.

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 tests 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 cooldown limit curves are not applicable to conditions of off-normal operation (e.g., small LOCA and extended loss of feedwater) where cooling is achieved for extended periods of time by circulating water from the HPI through the core. If core cooling is restricted to meet the cooldown limits under other than normal operation, core integrity could be jeopardized.

The pressure-temperature limit lines shown on the figures specified in 3.1.2-1 for reactor criticality and one the figures referred to in 3.1.2-3 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 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 10 CFR 50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region, or in test reactors.

The limitations on steam generator pressure and temperature provides 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 requirements to perform leakage tests of systems outside of containment which could potentially contain radioactivity were established by the NRC following TMI. Oconee performs the leak test of LPI by establishing RCS pressure at about 300 psig and with LPI at this same pressure, checking for leakage. Such a test is within the scope of testing upon which the curves referenced in Specification 3.1.2.2 are based--that is, they are not routine evolutions, such as heatup and cooldown, but rather infrequent leak tests conducted on a refueling outage basis. As such, the hydrostatic/leak test pressure-temperature limitations are applicable for the RCS when performing leak tests of the LPI system.

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

The reactor vessel is protected against damage due to excessive pressures at low temperatures by the Low Temperature Overpressure Protection (LTOP) System. LTOP vulnerability is assumed when RCS cold leg temperature is $\leq$ 325°F and a RCS vent path equivalent to the PORV is not open.

The LTOP System consists of two trains. One train is the pressurizer PORV calibrated to a low setpoint of less than or equal to 500 psig for Units 1 and 2 and 480 psig for Unit 3. The PORV block valve must be open for the PORV to be operable. The capacity of the pressurizer PORV is sufficient to maintain the RCS pressure below the appropriate brittle fracture pressure limits during LTOP events in which boiling does not occur in the core. The remaining train is operator action and is based on an operating philosophy that precludes the plant from being in a water solid condition (except for system hydrotests). The fact that the Oconee units are operated with a steam or gas space in the pressurizer allows sufficient time for operator action to terminate an LTOP event prior to exceeding the appropriate brittle fracture pressure limits. Assuming an LTOP event was to occur at Oconee, and a single failure disables either train, the remaining train must be capable of maintaining RCS pressure below the appropriate brittle fracture pressure limits.

The Oconee LTOP System provides protection from pressure transients at low temperatures, by limiting the pressure of such a transient to below the limits set by 10CFR 50 Appendix G utilizing a conservative safety factor of 1.5. In addition, the following conditions are imposed by the NRC for the evaluation of the acceptability of LTOP Systems:

a. The most limiting initial conditions must be used.
b. The most limiting single failure, distinct from the initiating event, must be used.
c. No credit can be taken for mitigative operator action until 10 minutes after the operators become aware that a pressure transient is in progress.

For the Oconee units, the most limiting single failure is failure of the single pressurizer PORV to open at its low pressure setpoint. Operator awareness is assumed to be achieved by actuation of control room alarms. The following scenarios have the potential to result in an LTOP event:

1) Makeup Control Valve (HP-120) fails full open.

2) Erroneous opening of a core flood tank (CFT) discharge valve.

3) Erroneous actuation of the HPI system.

4) All pressurizer heaters erroneously energized.

5) Temporary loss of decay heat removal.

6) Thermal expansion of the RCS after starting an RCP due to stored energy in the steam generator.

Specification 3.1.2.9.2 requires that both CFTs and both HPI trains be isolated from the RCS, thus preventing these scenarios. Physical restriction of makeup flow, control of pressurizer level, and alarms ensure that at least 10 minutes are available for operator action to mitigate the remaining events. Units specific values required to meet the 10 minute operator action criterion are provided within the Selected Licensee Commitment Manual.

In order to assure 10 minutes are available for operator action, the operational restrictions of Specification 3.1.2.9.2 must be implemented:

Deactivating train A of HPI is accomplished by one of the following methods:

1) Shutting and deactivating valve HP-26 by tagging open the valve breaker and tagging the valve handwheel in the closed position, shutting valve HP-410 and tagging the valve switch in the closed position.
2) Deactivating all HPI pumps aligned to A HPI train and tagging the pump breakers open.

Amendment No. 188 (Unit 1)
Amendment No. 188 (Unit 2)
Amendment No. 185 (Unit 3)

Deactivating train B of HPI is accomplished by one of the following methods:

1) Shutting and deactivating valve HP-27 by tagging open the valve breaker and tagging the valve handwheel in the closed position, shutting valve HP-409 and tagging the valve switch in the closed position.
2) Deactivating all HPI pumps aligned to B HPI train and tagging the pump breakers open.

Deactivating both core flood tanks is accomplished by shutting valves CF-1 and CF-2, tagging open the valve breaker, and tagging the valves in the closed position. Alternately, core flood tanks may be deactivated by maintaining core flood tank pressure below the maximum allowable RCS pressure for the existing RCS temperature (per Figures 3.1.2-1 and 3.1.2-2).

Makeup flow must be restricted such that 10 minutes are available for operator action to mitigate the event.

Audible alarms must be provided such that 10 minutes are available for operator action to mitigate the event.

The intent of the action statements provided in Specification 3.1.2.9.3 is to place the reactor vessel in a condition in which it is not vulnerable to a LTOP event via the safest and most prompt course of action. In some cases, it may be more prudent to heat up above 325°F (cold leg temperature) rather than depressurize and open an RCS vent.

The allowable outage times (AOTs) provided in Specification 3.1.2.9.3 have been established based on the following considerations:

a. The most rapid LTOP scenarios exist with HPI pumps in operation, or with RCS pressure $\geq$ 100 psig. As such, the 24 hour AOT is sufficiently brief to assure a minimal period of operation while vulnerable to a single failure of the PORV.

b. In the event of '2nd train' inoperabilities, a time period of 4 hours is sufficient to return the train to operable status or to implement the compensatory measures.

<u>REFERENCES</u>

(1) Analysis of Capsule OCII-E from Duke Power Company Oconee Unit 2 Reactor Vessel Materials Surveillance Program, BAW-2051, October, 1988.

(2) Analysis of Capsule OCIII-B from Duke Power Company Oconee Unit 3 Reactor Vessel Materials Surveillance Program, BAW-1697, October, 1981.

(3) Analysis of Capsule OCI-C from Duke Power Company Oconee Unit 1 Reactor Vessel Materials Surveillance Program, BAW-2050, October, 1988.

TABLE 3.1-1

OPERATIONAL SPECIFICATIONS FOR PLANT HEATUP

I. RC Temperature Constraints

RC Temperature(1)	Maximum Heatup Rate
$T \leq 280°F$	50°F/HR
$T > 280°F$	100°F/HR

II. RC Pump Constraints

RC Temperature(1)	Allowed Pump Combination
$T > 250°F$	Any
$T \leq 250°F$	No more than 1 pump per loop

(1) RC Temperature is cold leg temperature if one or more RC pumps are in operation; otherwise it is the LPI cooler outlet temperature.

TABLE 3.1-2

OPERATIONAL SPECIFICATIONS FOR PLANT COOLDOWN

I. RC Temperature Constraints

RC Temperature(1)	Maximum Cooldown Rate(2)
T > 280°F	≤ 50°F in any 1/2 hour period
150°F < T ≤ 280°F	≤ 25°F in any 1/2 hour period
T ≤ 150°F	≤ 10°F in any 1 hour period
RCS depressurized (3)	≤ 50°F in any 1 hour period

II. RC Pump Constraints For Validity of Guidance

RC Temperature(1)	Allowed Pump Combinations
T > 270°F	Any
200°F < T ≤ 270°F	No more than 1 pump per loop
T ≤ 200°F	No more than 1 pump

(1) RC Temperature is cold leg temperature if one or more RC pumps are in operation or if on natural circulation cooldown; otherwise it is the LPI cooler outlet temperature.

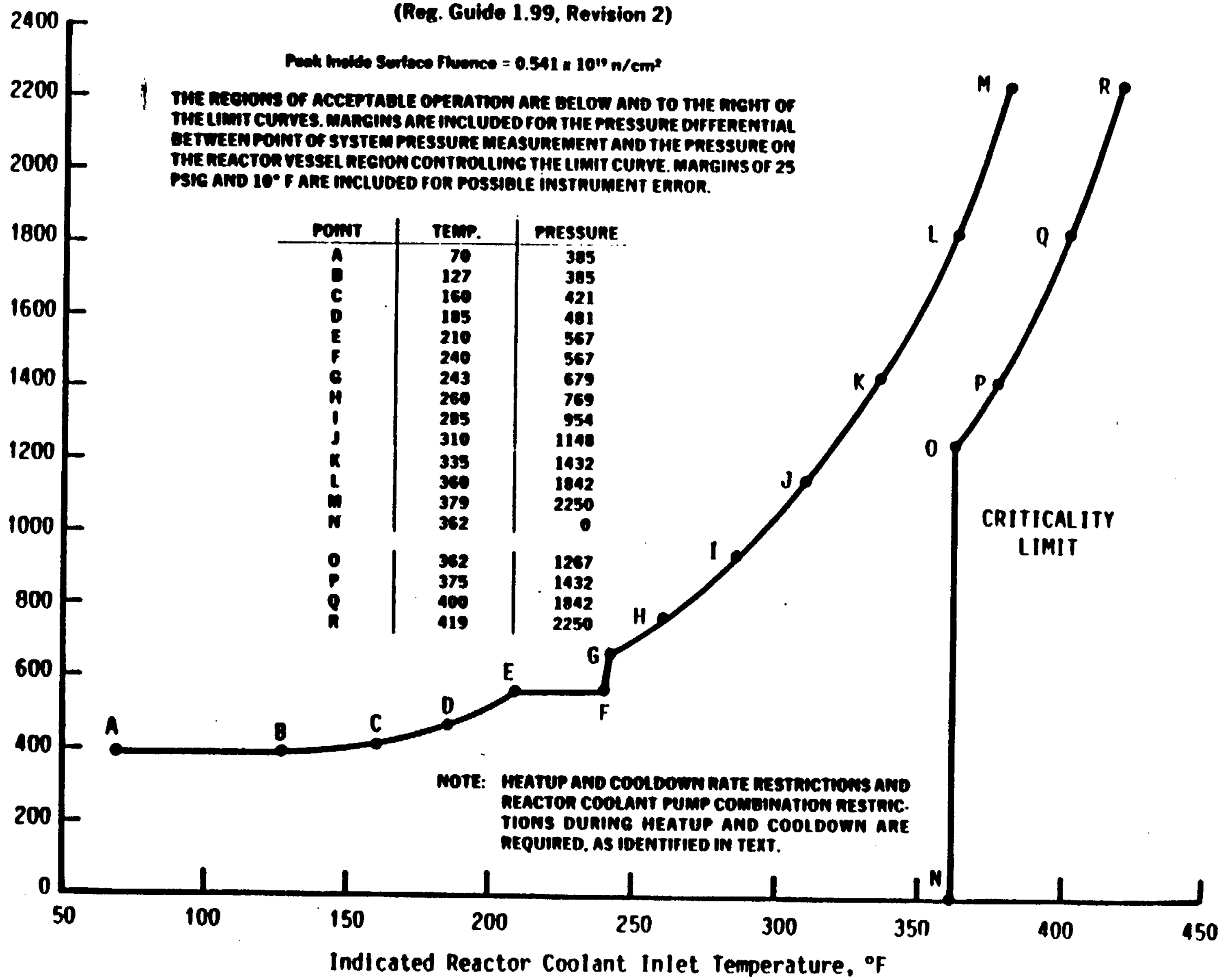
(2) These rate limits must be applied to the change in temperature indication from cold leg temperature to LPI cooler outlet temperature per Note (1).

(3) When the RCS is depressurized such that all three of the following conditions exist:

a) RCS temperature < 200°F,
b) RCS pressure < 50 psig,
c) All RC Pumps off,

the maximum cooldown rate shall be relaxed to ≤ 50°F in any 1 hour period.

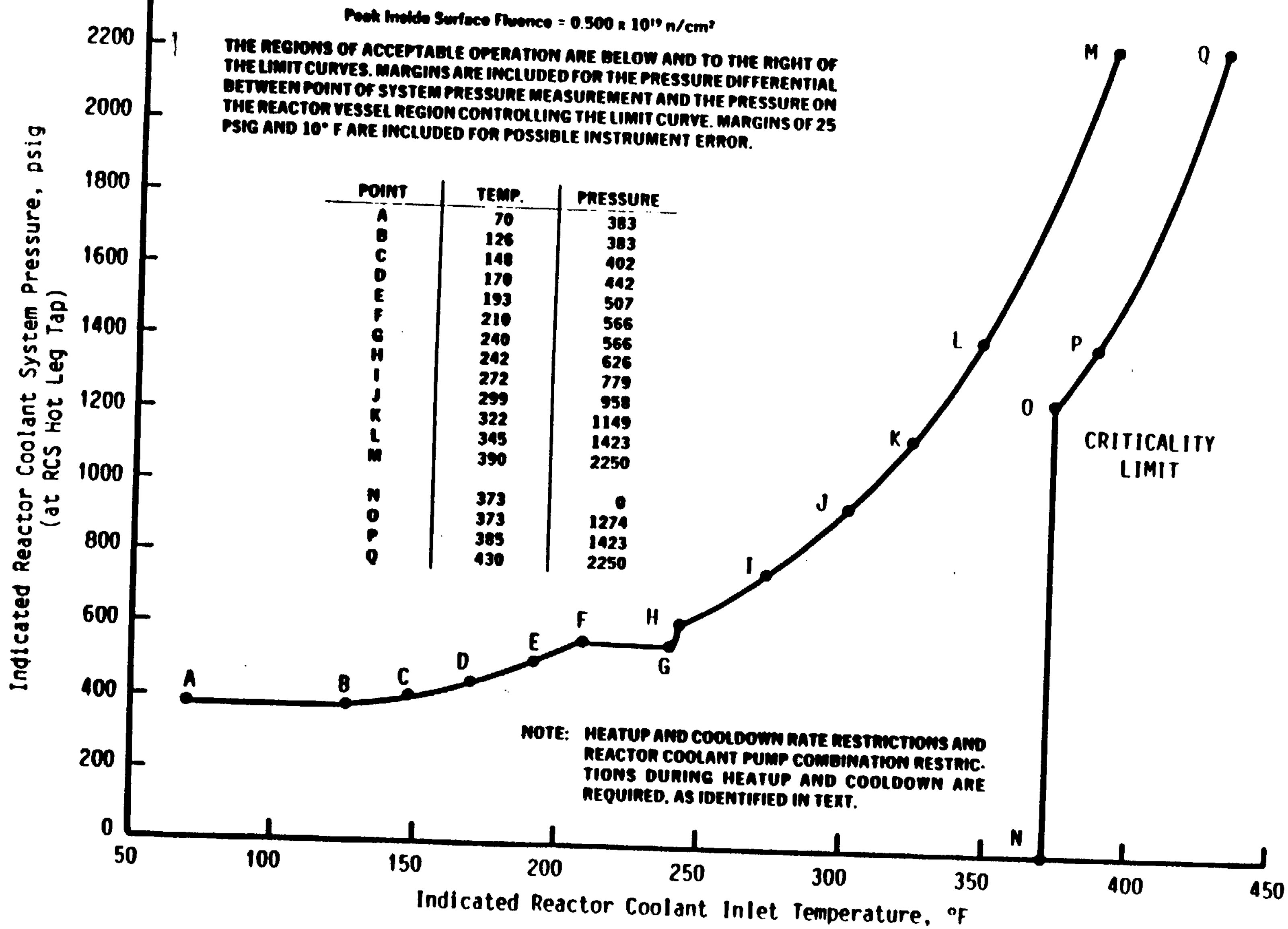
Unit 1 Oconee Nuclear Station
Reactor Coolant System Normal Operation Heatup
Limitations Applicable for First 15.0 EFPY
(Reg. Guide 1.99, Revision 2)



POINT	TEMP.	PRESSURE
A	70	385
B	127	385
C	160	421
D	185	481
E	210	567
F	240	567
G	243	679
H	260	769
I	285	954
J	310	1148
K	335	1432
L	360	1842
M	379	2250
N	362	0
O	362	1267
P	375	1432
Q	400	1842
R	419	2250

Figure 3.1.2-1A

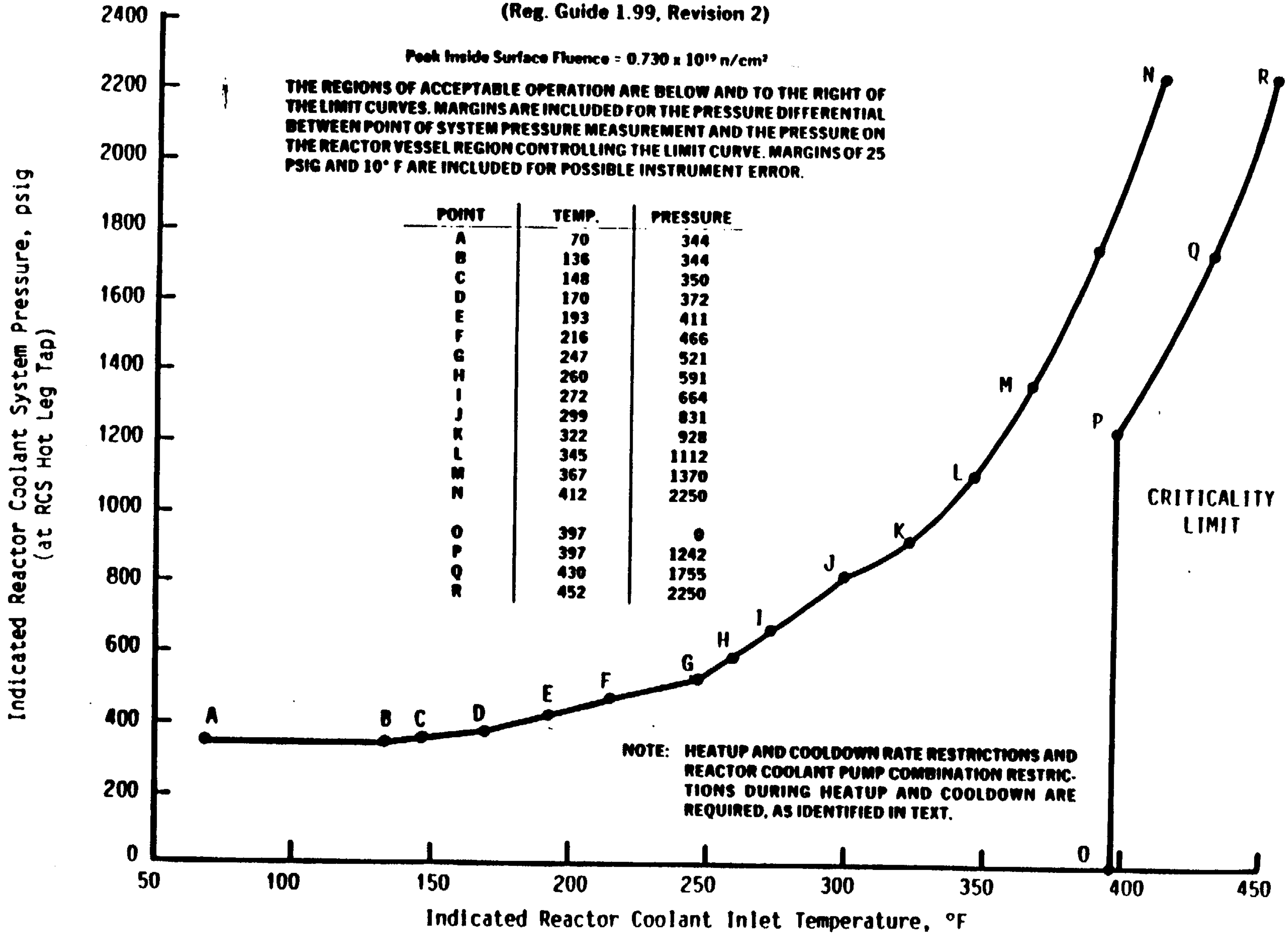
Unit 2 Oconee Nuclear Station
Reactor Coolant System Normal Operation
Heatup Limitations Applicable for First 15.0 EFPY
(Reg. Guide 1.99, Revision 2)



POINT	TEMP.	PRESSURE
A	70	383
B	126	383
C	148	402
D	170	442
E	193	507
F	210	566
G	240	566
H	242	626
I	272	779
J	299	958
K	322	1149
L	345	1423
M	390	2250
N	373	0
O	373	1274
P	385	1423
Q	430	2250

Figure 3.1.2-1B

Unit 3 Oconee Nuclear Station
Reactor Coolant System Normal Operation Heatup
Limitations Applicable for First 15.0 EFPY
(Reg. Guide 1.99, Revision 2)



POINT	TEMP.	PRESSURE
A	70	344
B	136	344
C	148	350
D	170	372
E	193	411
F	216	466
G	247	521
H	260	591
I	272	664
J	299	831
K	322	928
L	345	1112
M	367	1370
N	412	2250
O	397	0
P	397	1242
Q	430	1755
R	452	2250

Figure 3.1.2-1C

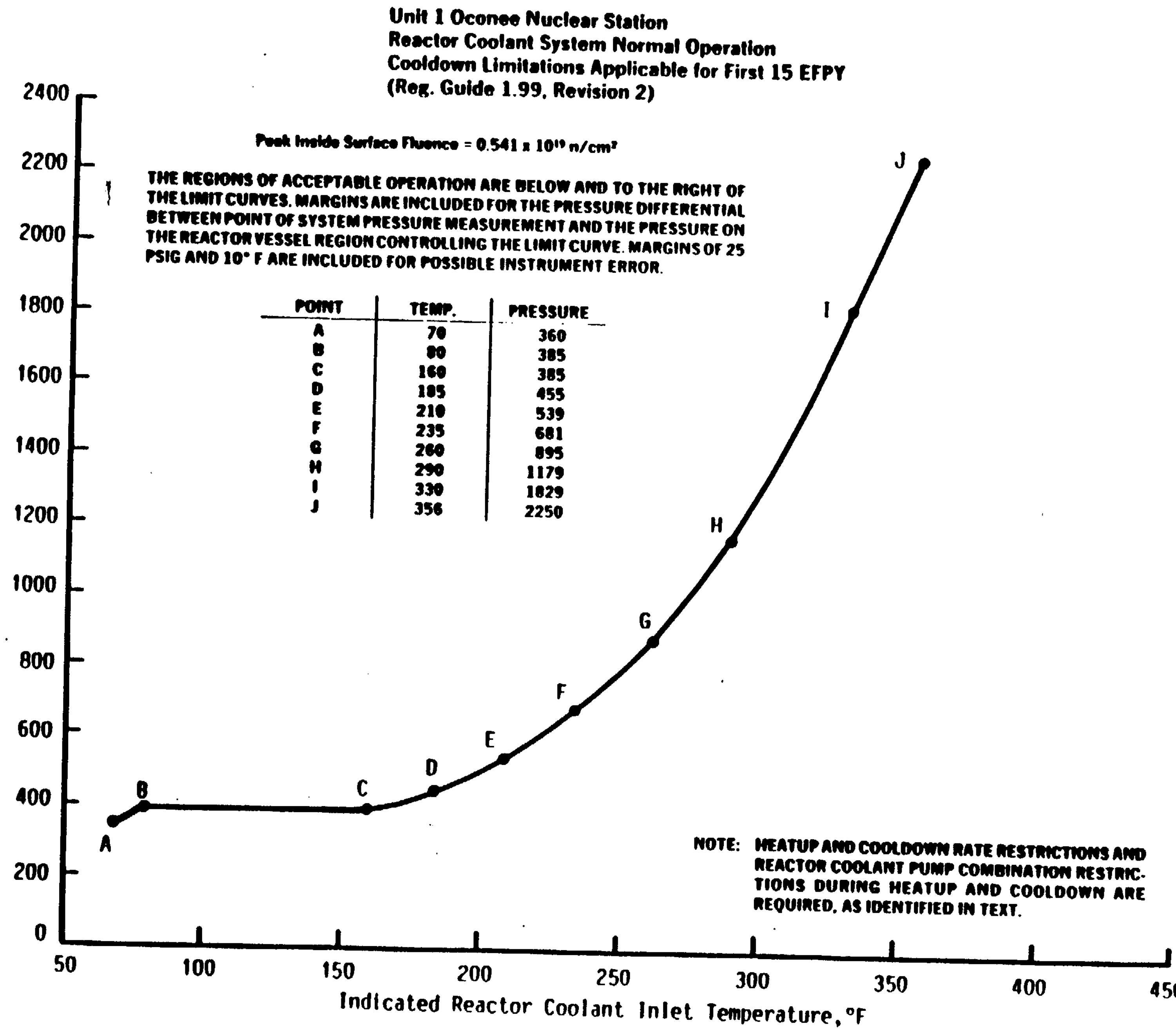
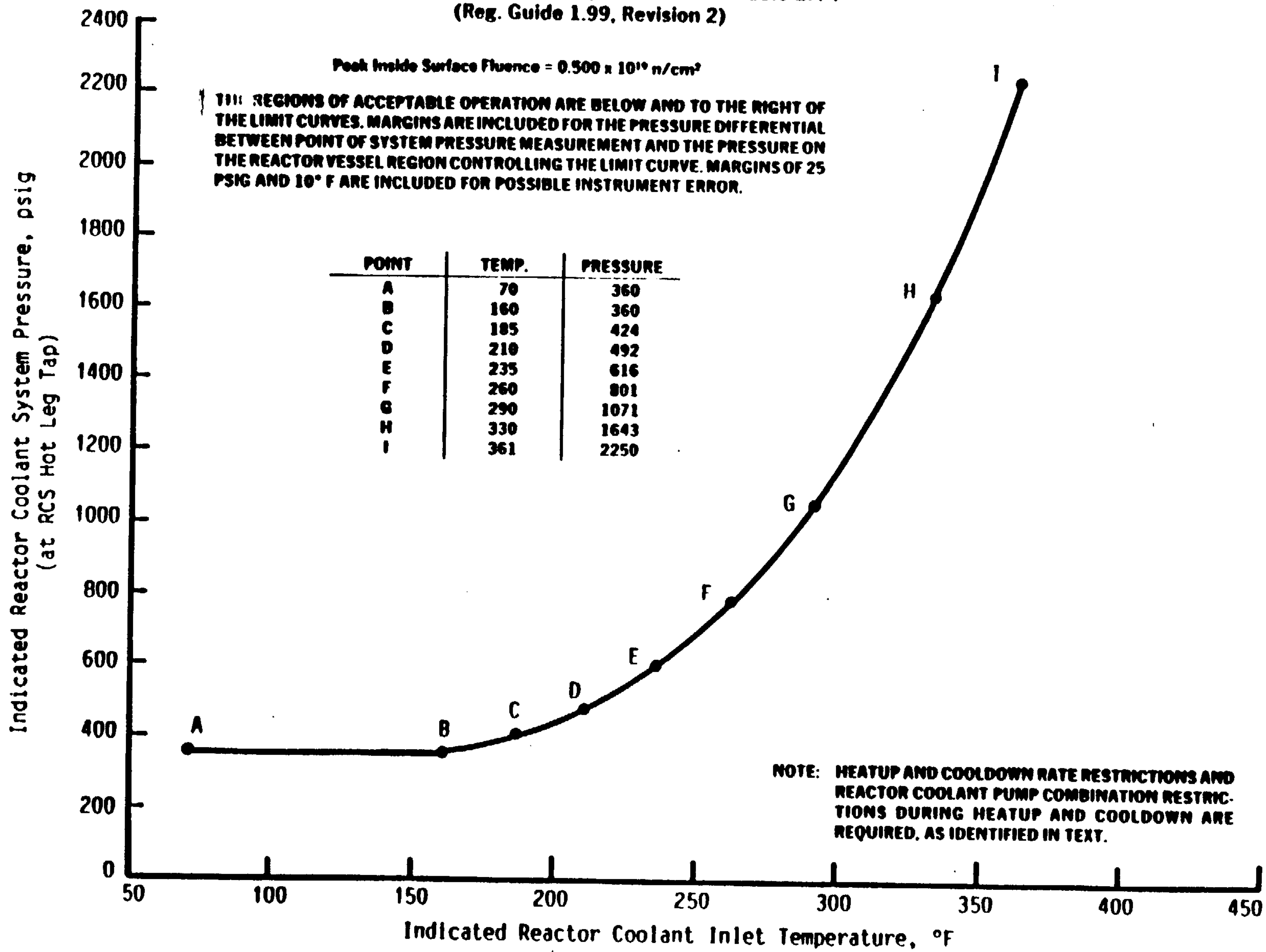


Figure 3.1.2-2A

Amendment No. 188 (Unit 1)
Amendment No. 188 (Unit 2)
Amendment No. 185 (Unit 3)

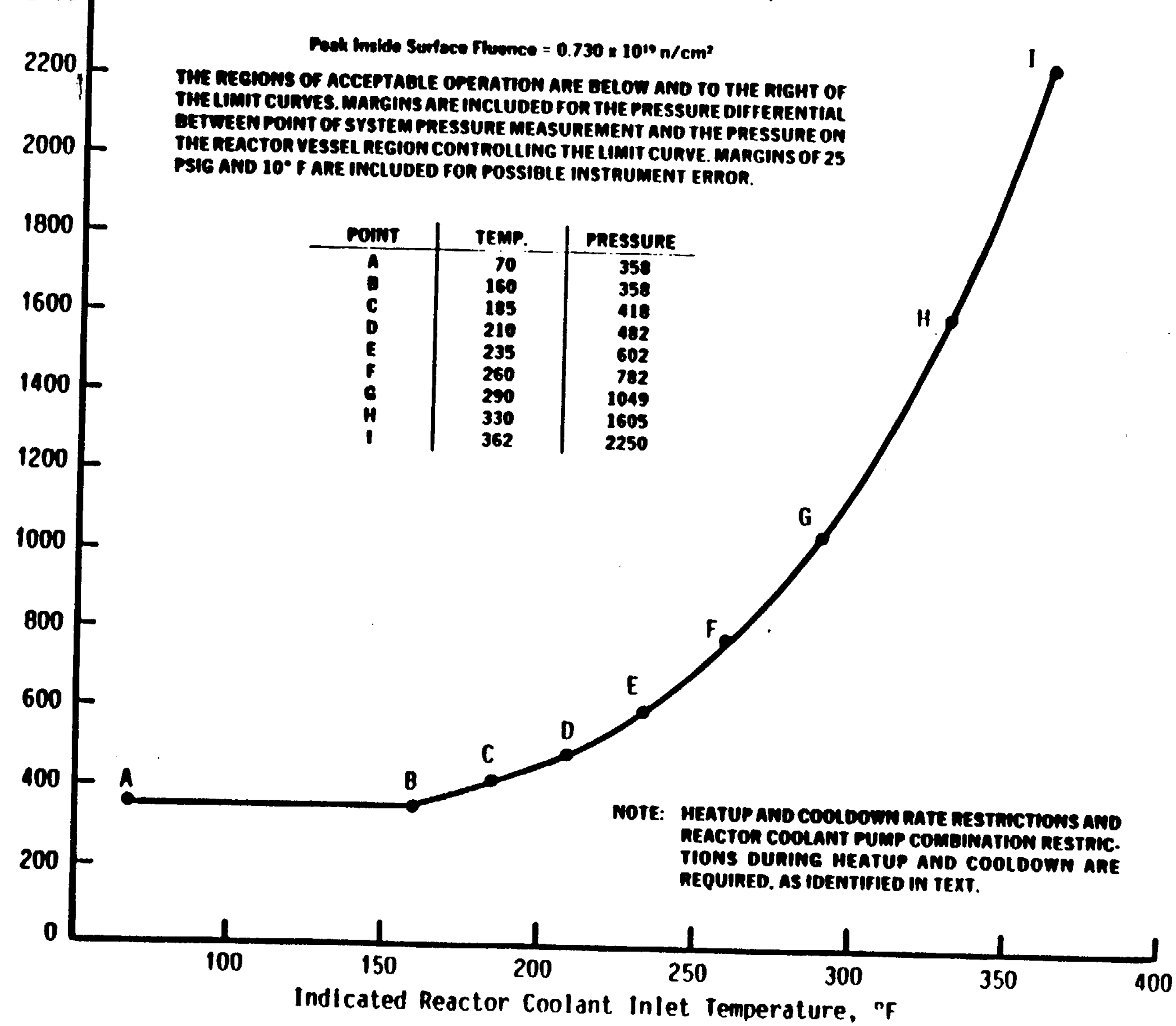
Unit 2 Oconee Nuclear Station
Reactor Coolant System Normal Operation Cooldown
Limitations Applicable for First 15.0 EFPY
(Reg. Guide 1.99, Revision 2)



POINT	TEMP.	PRESSURE
A	70	360
B	160	360
C	185	424
D	210	492
E	235	616
F	260	801
G	290	1071
H	330	1643
I	361	2250

Figure 3.1.2-2B

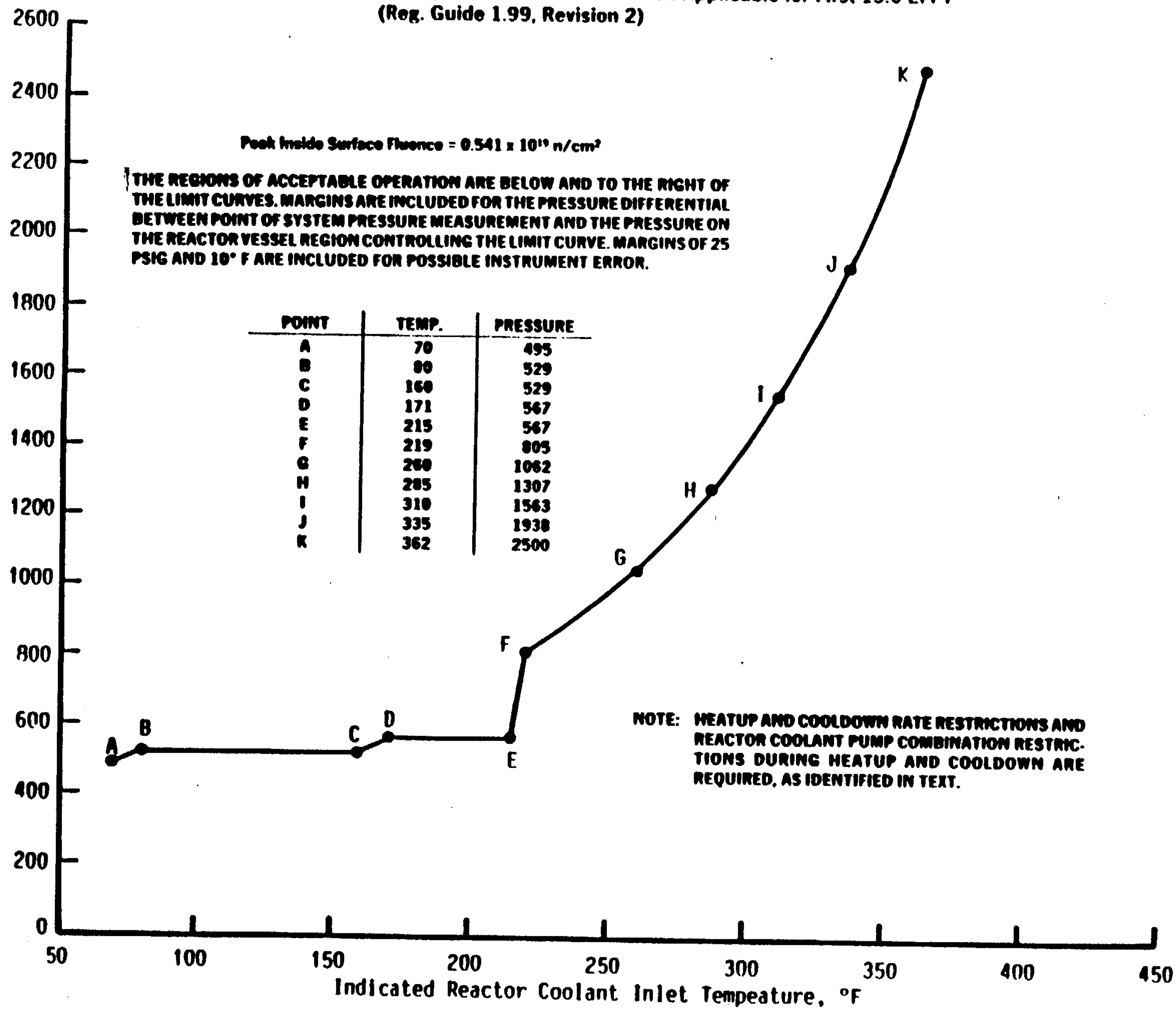
Unit 3 Oconee Nuclear Station
Reactor Coolant System Normal Operation Cooldown Limitations Applicable for First 15.0 EFPY
(Reg. Guide 1.99, Revision 2)



POINT	TEMP.	PRESSURE
A	70	358
B	160	358
C	185	418
D	210	482
E	235	602
F	260	782
G	290	1049
H	330	1605
I	362	2250

Figure 3.1.2-2C

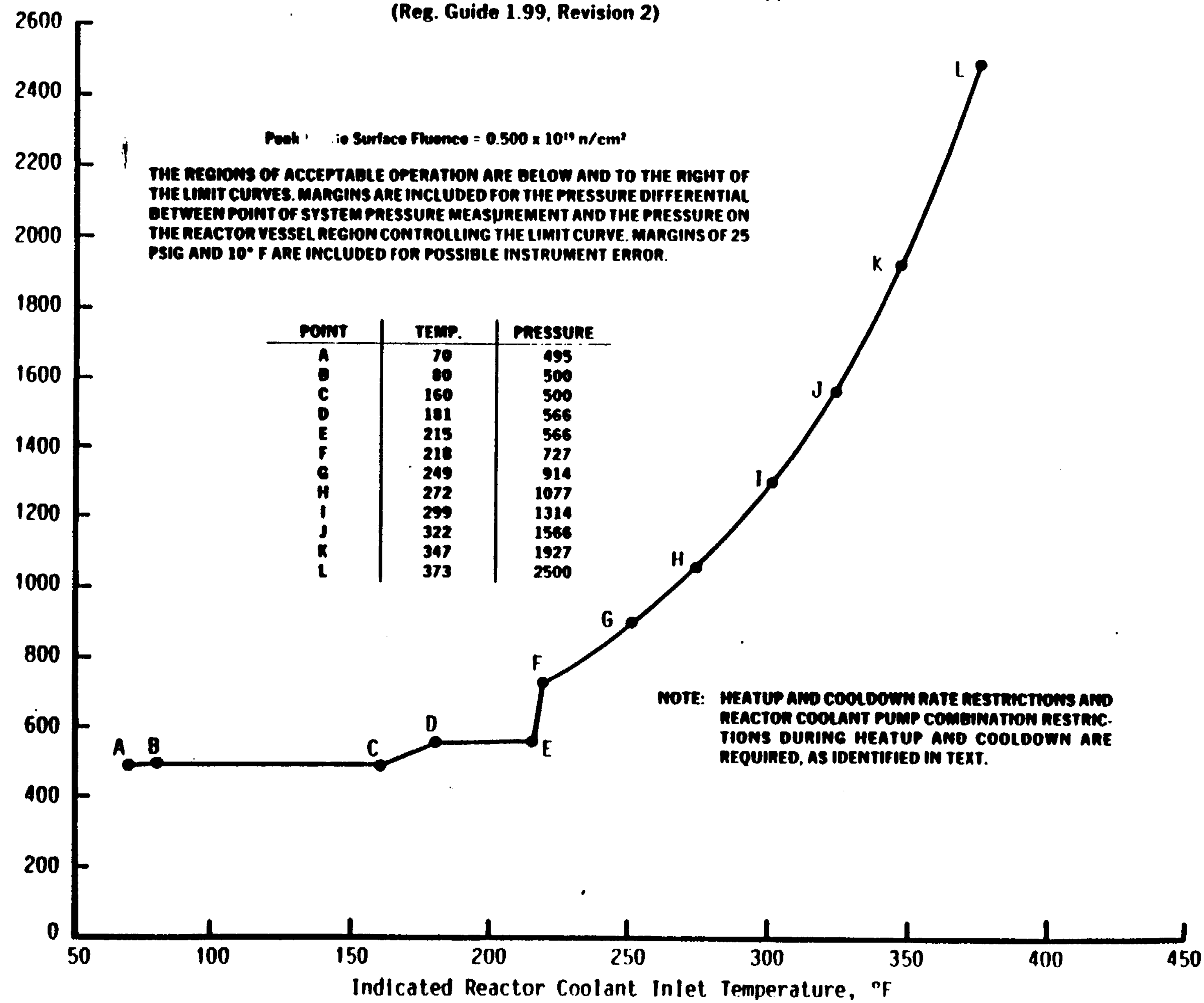
Unit 1 Oconee Nuclear Station
Reactor Coolant System Inservice Leak and Hydrostatic Test
Heatup and Cooldown Limitation Applicable for First 15.0 EFPY
(Reg. Guide 1.99, Revision 2)



POINT	TEMP.	PRESSURE
A	70	495
B	80	529
C	160	529
D	171	567
E	215	567
F	219	805
G	260	1062
H	285	1307
I	310	1563
J	335	1938
K	362	2500

Figure 3.1.2-3A

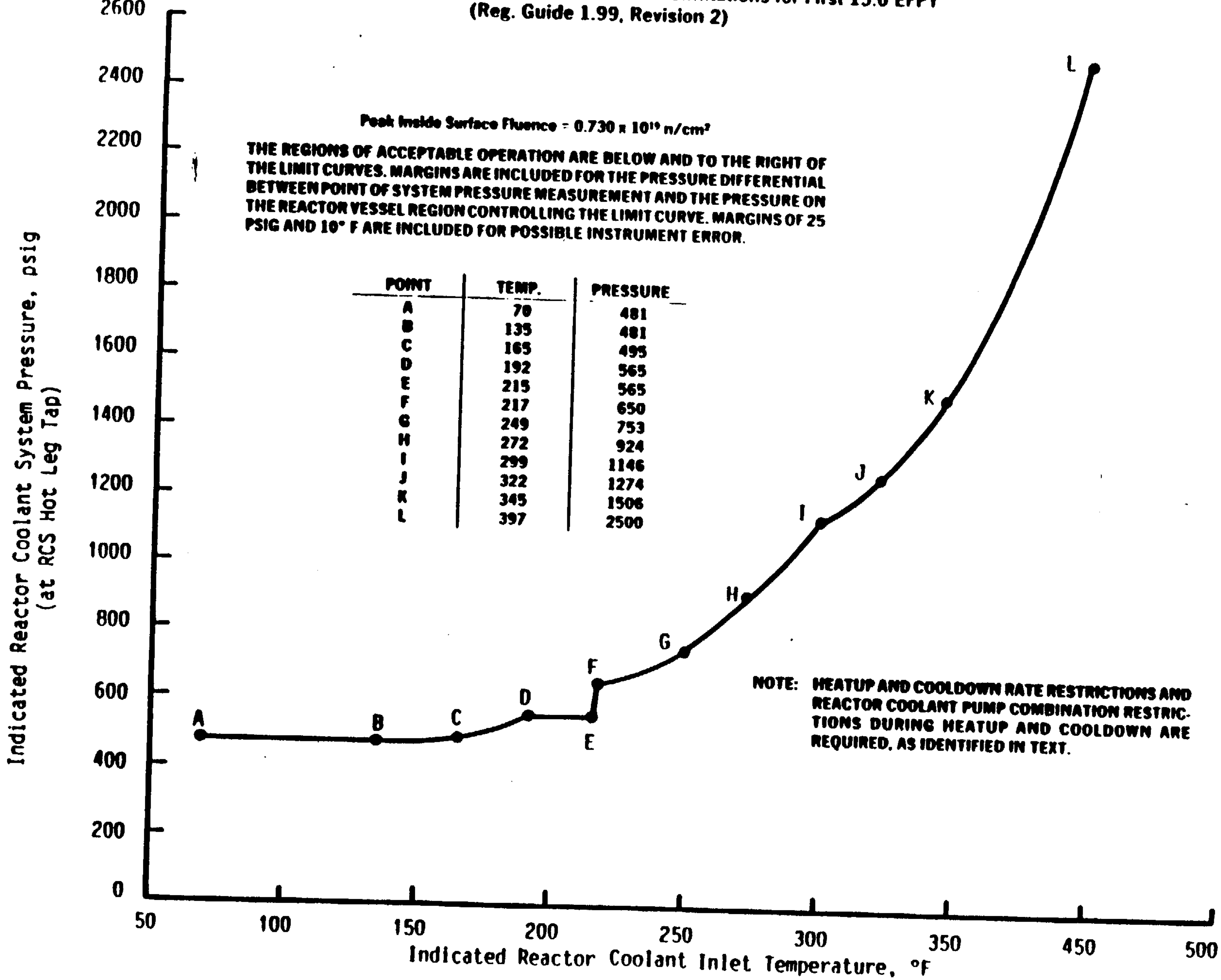
Unit 2 Oconee Nuclear Station
Reactor Coolant System Inservice Leak and Hydrostatic
Test Heatup and Cooldown Limitations Applicable for 15.0 EFPY
(Reg. Guide 1.99, Revision 2)



POINT	TEMP.	PRESSURE
A	70	495
B	80	500
C	160	500
D	181	566
E	215	566
F	218	727
G	249	914
H	272	1077
I	299	1314
J	322	1566
K	347	1927
L	373	2500

Figure 3.1.2-3B

Unit 3 Oconee Nuclear Station
Reactor Coolant System Inservice Leak and Hydrostatic Test
Heatup and Cooldown Limitations for First 15.0 EFPY
(Reg. Guide 1.99, Revision 2)



POINT	TEMP.	PRESSURE
A	70	481
B	135	481
C	165	495
D	192	565
E	215	565
F	217	650
G	249	753
H	272	924
I	299	1146
J	322	1274
K	345	1506
L	397	2500

Figure 3.1.2-3C

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE DPR-47

AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By letter dated November 15, 1989, as supplemented March 28 and August 29, 1990, Duke Power Company (the licensee) submitted a request for changes to the Oconee Technical Specifications (TS). The requested changes would revise the Pressure/Temperature (P/T) Limits and the Low Temperature Overpressure Protection (LTOP) system operability requirements in Section 3.1 of the TSs and revise the associated Bases. The March 28 and August 29, 1990, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The changes to P/T limits are in response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect On Plant Operation." The current P/T limit curves are based on an assumed design basis neutron fluence through 15 effective full power years (EFPYs). The proposed P/T limit curves were developed based on the guidance of Regulatory Guide (RG) 1.99, Rev. 2, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," and they are applicable for a period of up to 15 EFPYs. Prompted by the P/T Limit revisions, changes to the low temperature overpressure protection (LTOP) system requirements (TS 3.1.9.2) have been proposed to assure proper protection to the reactor vessel.

2.0 EVALUATION

The NRC staff has evaluated the proposed changes to the P/T limits utilizing the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50, ASTM Standards and the ASME Code, 10 CFR 50.36(c)(2), RG 1.99, Rev. 2, Standard Review Plan (SRP) Section 5.3.2, and GL 88-11. Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE).

Appendix H of 10 CFR Part 50 states testing requirements for reactor beltline material in surveillance capsules and provides a connection with ASTM Standards. GL 88-11 established the NRC requirement that the licensee use the guidance in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel material. 10 CFR 50.36(c)(2) states the limiting conditions for operation and SRP Section 5.3.2 provides the NRC staff position on P/T limits.

OCONEE UNIT 1 - P/T Limits

The NRC staff has evaluated the effect of neutron irradiation embrittlement on each beltline material in the Oconee 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 15 EFPY was weld SA-1585 with 0.21% copper (Cu), 0.59% nickel (Ni), and an initial RT_{ndt} of -12°F.

The licensee has removed four surveillance capsules from Oconee 1. The results from capsules F, E, A and C were published in Babcock and Wilcox (BAW) reports BAW-1421, Rev. 1, BAW-1436; BAW-1837, and BAW-2050, respectively. Surveillance capsule F contained Charpy impact and tensile specimens from plates. Surveillance capsules E, A and C contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and heat affected zone (HAZ) metal.

For the limiting beltline material, weld SA-1585, the staff calculated the ART to be 184.1°F at 1/4T (T = reactor vessel beltline thickness) and 140°F for 3/4T at 15 EFPY. The staff used a neutron fluence of 3.26E18 n/cm^2 at 1/4T and 1.18E18 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because the limiting weld material was not included in the surveillance capsules.

The licensee calculated the ARTs of 184°F and 140°F at 1/4T and 3/4T, respectively. Substituting the ARTs of 184°F and 140°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

OCONEE UNIT 2 - P/T Limits

The NRC staff has determined that the material with the highest ART at 15 EFPY was the circumferential weld between the lower shell nozzle shell belt and upper shell (WF-25) with 0.31% Cu, 0.59% Ni, and an initial RT_{ndt} of -6°F.

The licensee has removed three surveillance capsules from Oconee 2. The results from capsules C, A, and E in Oconee 2 were published in BAW reports BAW-1437, BAW-1699, and BAW-2051, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld WF-154, the staff calculated the ART to be 191°F at 1/4T and 138.3°F at 3/4T. The staff used a neutron fluence of 3.01E18 n/cm^2 at 1/4T and 1.09E18 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because the limiting material was not included in the surveillance capsules.

The licensee selected the weld, WF-25, as the limiting material and calculated the ARTs of 193°F and 139°F at 1/4T and 3/4T, respectively. The licensee's ARTs are conservative; therefore, they are acceptable. Substituting the ARTs of 193°F and 139°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

OCONEE UNIT 3 - P/T Limits

The NRC staff has determined that the material with the highest ART at 15 EFPY at 1/4T was the nozzle belt to upper shell circumferential weld (WF-200) with 0.26% Cu, 0.64% Ni, and an initial RT_{ndt} of 20°F. At 3/4T, the material with the highest ART at 15 EFPY was the upper to lower shell circumferential weld (WF-70) with 0.35% Cu, 0.59% Ni, and an initial RT_{ndt} of 20°F.

The licensee has removed two surveillance capsules from Oconee 3. The results from capsules A and B were published in BAW reports BAW-1438 and BAW-1697, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline materials, welds WF-200 and WF-70, the staff calculated the ART to be 202.2°F at 1/4T and 170.1°F for 3/4T at 15 EFPY. The staff used a neutron fluence of 3.18E18 n/cm^2 at 1/4T and 1.15E18 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because the limiting material was not included in the surveillance capsules.

The licensee selected welds WF-67 and WF-70 as the limiting materials at the 1/4T and 3/4T locations, respectively, and calculated the ARTS of 195°F at 1/4T and 171°F at 3/4T. The difference between the staff's ART and the licensee's ART at the 1/4T location (202°F vs. 195°F) is because the licensee used a lower chemistry factor in the ART calculation. Substituting the ARTs of 202.2°F and 171°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50. The licensee's P/T limits have sufficient safety margin such that with a lower ART the limits satisfy the Appendix G calculation.

OCONEE UNITS 1, 2 AND 3 - CLOSURE FLANGE LIMITS AND CHARPY USE AT EOL

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 60°F, the staff has determined that the proposed P/T limits for Oconee Units 1, 2 and 3 satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. Using the method in RG 1.99, Rev. 2, the staff predicted that the Charpy USE of weld metal SA-1430 and SA-1493 for Unit 1, SA-154 for Unit 2, and WF-200, WF-67 and WF-70 for Unit 3, will be below 50 ft-lb at the end of life. The licensee has joined the Babcock and Wilcox Owners Group (B&WOG) that is investigating the effect of low Charpy USE on fracture toughness of the reactor vessel beltline materials. The staff will withhold evaluation of the low Charpy USE until the B&WOG submits its findings on this issue for NRC staff review.

OCONEE UNITS 1, 2 AND 3 - LTOP

In light of the above revisions to the P/T curves, the LTOP system has been reviewed to assure proper protection of the reactor vessel. As a result of the review of the LTOP system, changes have been proposed to TS 3.1.9.2 to resolve conflicts which were identified in the current TSs concerning inadvertent actuation of the High Pressure Injection (HPI) system.

LTOP is provided by the power operated relief valve (PORV) on the pressurizer. The PORV is set at a pressure low enough to prevent violation of Appendix G heatup and cooldown curves should a reactor coolant system (RCS) pressure transient occur during low temperature operations. The licensee, in its November 15, 1989, submittal provided the results of considerations of the most limiting overpressure transients in determining the necessary TS changes for LTOP.

Currently, at Oconee, LTOP is provided by two methods, at least one of which must be operable when RCS cold leg temperature is less than or equal to 325°F. One method requires that both train A and train B of the HPI system be disabled. The other method consists of the PORV with a lift setting of less than or equal to 500 psig, pressurizer level less than or equal to 260 inches, and RCS pressure less than 400 psig. Because only one of the two LTOP methods is required to be operable in the existing TSs the current TSs are less conservative than intended, and operation is being controlled administratively to assure that the vessel is protected from LTOP events.

The licensee's proposed LTOP TSs require that two trains of LTOP be operable. The first train is comprised of a PORV with a reduced lift setting. This train must be operable when RCS cold leg temperature is less than or equal to 325°F and either RCS pressure is greater than or equal to 100 psig or HPI pumps are running. The second train is comprised of the controls which assure that 10 minutes are available for operator action to mitigate the consequences of an LTOP event. The following controls are included in the second LTOP train:

a. RCS pressure is limited to less than or equal to 350 psig for Units 1 and 2, and less than or equal to 345 psig for Unit 3, below an RCS temperature of 220°F.

b. Deactivating train A and B of HPI.

c. Deactivating both core flood tanks.

d. Pressurizer level shall be controlled such that 10 minutes are available for operator action to mitigate an LTOP event.

e. Makeup flow shall be restricted such that 10 minutes are available for operator action to mitigate an LTOP event.

f. Alarms shall be provided such that 10 minutes are available for operator action to mitigate an LTOP event.

The licensee has committed to include specific limiting values of maximum and minimum pressurizer level and maximum makeup flow rate in the Selected Licensee Commitments Manual which is found in Chapter 16 of the Final Safety Analysis Report.

The licensee has provided an evaluation and results of analyses to justify the proposal in its November 15, 1989 submittal. The evaluations assumed initiation of the most limiting mass addition and heat addition transients plus the most limiting single failure, failure of the pressurizer PORV. No credit was taken for operator action until 10 minutes after the event is recognized.

The most limiting mass addition transients were analyzed, and the results of these analyses are as follows. Erroneous actuation of the HPI system or the core flood discharge valves has been precluded by requiring that both the A and B trains of the HPI system and the core flood injection function be deactivated while RCS temperature is below 325°F. It has been determined that erroneous addition of nitrogen to the pressurizer results in a maximum pressure of only 150 psig and therefore is not a limiting transient. The last mass addition LTOP event analyzed was RCS makeup fails full open. This scenario was found to cause RCS pressure to increase above the new P/T limits in less than 10 minutes, when an initial pressurizer level of 220 inches is assumed. The most limiting makeup flow rate providing at least 10 minutes for operator action was calculated for each of the Units, and the flow indication uncertainty was subtracted to determine the maximum allowable flow rate with RCS temperature below 325°F. The makeup flow path will be restricted to the maximum allowable flow rate during low temperature operations.

Also included in the evaluation were the following limiting heat addition transients: loss of decay heat removal during startup and shutdown, all pressurizer heaters erroneously energized, and start of a reactor coolant pump (RCP) with stored thermal energy in the RCS. The loss of decay heat removal scenario was analyzed using conservative assumptions. One or more alarms will actuate on loss of decay heat removal, therefore, operator awareness is assumed at the beginning of the transient. It was determined that at least 10 minutes are available between the start of the transient and the time at which the pressurizer reaches the P/T limits. The initial conditions of this analysis determine the maximum allowable pressurizer pressure and level for operations with the RCS temperature below 220°F, and these maximum values are 350 psig and 220 inches, respectively. For the scenario of all pressurizer heaters erroneously energized, it was found that the rate of pressurization was inversely proportional to the initial pressurizer water level. As a result, a limit will be placed on the minimum allowable pressurizer level, such that 10 minutes will be available for operator action to mitigate the event. Finally, the start of an RCP with stored thermal energy in the RCS was analyzed. It was concluded that this transient will not cause the RCS pressure to exceed the P/T limits at any temperature.

These analyses were performed using the RETRAN-02 MOD003 computer code. RETRAN-02 MOD003 has been generically approved by the NRC staff on October 19, 1988. Currently Duke Power Company's application for the use of RETRAN-02 MOD003 is under review by the staff. The staff considers that reasonable assurance exists that the results of the licensee's analysis using RETRAN-02 MOD003 supports the proposed TSs on LTOP. However, if any concerns should arise during the staff review of Duke Power's RETRAN-02 MOD003 application, the staff may require a reassessment of the proposed TS changes.

The staff has also reviewed the proposed Bases for TS 3.1.2.9 and finds the discussion correctly describes the proposed LTOP features and is acceptable.

3.0 SUMMARY

The NRC staff concludes that the proposed P/T limits for the RCS for heatup, cooldown, leak test, and criticality are valid through 15 EFPYs because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Oconee Unit 1, 2 and 3 TSs.

The staff has determined that some beltline welds will have Charpy upper shelf energy below 50 ft-lb at end of life in all three Oconee units. The licensee has joined the B&WOG that is investigating the effects of low Charpy USE on fracture toughness of the reactor vessel beltline materials. The staff will withhold the conclusion on the matter until the staff has reviewed the B&WOG's report on the low USE in Oconee Units 1, 2 and 3.

Also, the staff concludes that the licensee proposed Technical Specification 3.1.9.2 are acceptable to support the updated Pressure-Temperature limits applicable for a period of up to 15 EFPY.

4.0 STATE CONSULTATION

In accordance with the commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (56 FR 4862 on February 6, 1991). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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