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

September 19, 1991

Posted
Amdt. 43 to NPF-68
(See Correction letter 10-15-91)

Docket Nos. 50-424
and 50-425

Mr. W. G. Hairston, III
Senior Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 43 AND 44 TO FACILITY OPERATING LICENSE NPF-68 AND AMENDMENT NOS. 23 AND 24 TO FACILITY OPERATING LICENSE NPF-81, VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 (TACS 79303/79304 AND 81240/81241)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 43 and 44 to Facility Operating License No. NPF-68 and Amendment Nos. 23 and 24 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated November 29, 1990, as supplemented January 29, March 6, March 27, March 29, April 19, August 8, and August 19, 1991, to provide for use of VANTAGE-5 fuel and increased operational flexibility.

Consistent with your request to implement changes to the TSs in two phases, Amendment Nos. 43 and 23 are effective beginning with the initial loading of VANTAGE-5 fuel into Vogtle Unit 1 Cycle 4, and Amendment Nos. 44 and 24 are effective beginning with the initial loading of VANTAGE-5 fuel into Vogtle Unit 2 Cycle 3. Your staff is aware that any future amendments affecting the TS pages changed by Amendment Nos. 44 and 24 will require appropriate updating of these TS pages if NRC approval is requested prior to Unit 2 Cycle 3.

Enclosed is the NRC staff's Safety Evaluation of your proposal. In this Safety Evaluation, we do agree with your proposed TS changes with the exception of the change to take credit for control rod insertion during long-term core cooling phase of loss-of-coolant-accident (LOCA).

To verify that the reactor will remain subcritical during the long term cooling phase following a LOCA, you proposed to apply leak-before-break (LBB) methodology to demonstrate mechanical integrity of the reactor upper internals and thus establish feasibility for control rod insertion during reactor coolant system line breaks smaller than 1.0 square feet. On the basis of this analysis, you proposed to change the TS Bases 3/4.5.4 to take credit for control rod insertion instead of providing increased boron concentration in the refueling water storage tank (RWST). As discussed in Section 2.1.4 of our Safety Evaluation, the NRC staff does not currently support the use of LBB methodology with respect to either

short-term or long-term effects of the FSAR (Final Safety Analysis Report) Chapter 15 events. Consequently, the proposed change to the TS Bases 3/4.5.4 for taking credit for control rod insertion has not been implemented in the TS amendments. Sufficient RWST boron concentration should be provided to maintain subcriticality of the reactor core during post LOCA conditions without taking credit for control rod insertion.

Notice of issuance of the amendments will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Darl Hood

Darl S. Hood, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment Nos. 43 and 44 to NPF-68
2. Amendment Nos. 23 and 24 to NPF-81
3. Safety Evaluation

cc w/enclosures:
See next page

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Notice of issuance of the amendments will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Darl S. Hood, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment Nos. 43 and 44 to NPF-68
2. Amendment Nos. 23 and 24 to NPF-81
3. Safety Evaluation

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Mr. W. G. Hairston, III
Georgia Power Company

Vogtle Electric Generating Plant

cc:
Mr. J. A. Bailey
Manager - Licensing
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Mr. W. B. Shipman
General Manager, Vogtle Electric
Generating Plant
P. O. Box 1600
Waynesboro, Georgia 30830

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW., Suite 2900
Atlanta, Georgia 30323

Office of the County Commissioner
Burke County Commission
Waynesboro, Georgia 30830

Office of Planning and Budget
Room 615B
270 Washington Street, SW.
Atlanta, Georgia 30334

Mr. C. K. McCoy
Vice President - Nuclear, Vogtle Project
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 572
Waynesboro, Georgia 30830

Mr. R. P. McDonald
Executive Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Mr. J. Leonard Ledbetter, Director
Environmental Protection Division
Department of Natural Resources
205 Butler Street, SE., Suite 1252
Atlanta, Georgia 30334

Attorney General
Law Department
132 Judicial Building
Atlanta, Georgia 30334

Mr. Alan R. Herdt
Project Branch #3
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW., Suite 2900
Atlanta, Georgia 30323

Mr. Dan Smith
Program Director of
Power Production
Oglethorpe Power Corporation
2100 East Exchange Place
P. O. Box 1349
Tucker, Georgia 30085-1349

Charles A. Patrizia, Esq.
Paul, Hastings, Janofsky & Walker
12th Floor
1050 Connecticut Avenue, NW.
Washington, DC 20036

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Facility Operating License No. NPF-68 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated November 29, 1990, as supplemented January 29, March 6, March 27, March 29, April 19, August 8, and August 19, 1991, 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 2.C.(2) of Facility Operating License No. NPF-68 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 43 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective beginning with the initial loading of VANTAGE-5 fuel into Vogtle Unit 1 Cycle 4.

FOR THE NUCLEAR REGULATORY COMMISSION

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: September 19, 1991

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Facility Operating License No. NPF-81 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated November 29, 1990, as supplemented January 29, March 6, March 27, March 29, April 19, August 8, and August 19, 1991, 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 2.C.(2) of Facility Operating License No. NPF-81 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 23 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective beginning with the initial loading of VANTAGE-5 fuel into Vogtle Unit 1 Cycle 4.

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: September 19, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. NPF-68

AND LICENSE AMENDMENT NO. 23

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NOS. 50-424 AND 50-425

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages effective beginning with initial loading of VANTAGE-5, Unit 1 Cycle 4. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

| Remove Pages | Insert Pages |
|---|---|
| III | III |
| V | V |
| XV/XVI* | XV/XVI* |
| XXIII | XXIII |
| 2-1 - 2-2 | 2-1 - 2-2 |
| | 2-2a |
| 2-3 - 2-11 | 2-3 - 2-11 |
| | 2-12 |
| | 2-13 |
| | 2-14 |
| | 2-15 |
| | 2-16 |
| | 2-17 |
| | 2-18 |
| | 2-19 |
| B 2-1 | B 2-1 |
| | B 2-1a |
| B 2-2 | B 2-2 |
| 3/4 1-11 | 3/4 1-11 |
| 3/4 1-12 | 3/4 1-12 |
| 3/4 1-13 | 3/4 1-13 |
| 3/4 1-19 | 3/4 1-19 |
| 3/4 2-1 | 3/4 2-1 |
| | 3/4 2-1a |
| 3/4 2-2 - 3/4 2-6 | 3/4 2-2 - 3/4 2-6 |
| 3/4 2-7 | 3/4 2-7 |
| | 3/4 2-7a |
| 3/4 2-13 | 3/4 2-13 |
| 3/4 3-33 / 3/4 3-34* | 3/4 3-33 / 3/4 3-34* |

| Remove Pages | Insert Pages |
|---|---|
| 3/4 5-1 / 3/4 5-2* | 3/4 5-1 / 3/4 5-2* |
| 3/4 5-9* / 3/4 5-10 | 3/4 5-9* / 3/4 5-10 |
| B 3/4 2-1 - B 3/4 2-6 | B 3/4 2-1 - B 3/4 2-6 |
| | B 3/4 2-7 |
| | B 3/4 2-8 |
| | B 3/4 2-9 |
| | B 3/4 2-10 |
| | B 3/4 2-11 |
| | B 3/4 2-12 |
| B 3/4 4-1 | B 3/4 4-1 |
| B 3/4 5-1* / B 3/4 5-2 | B 3/4 5-1* / B 3/4 5-2 |
| 6-21 | 6-21 |
| 6-21a | 6-21a |
| | 6-21b |
| | 6-21c |

* Overleaf pages containing no changes

# INDEX

## SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS


| SECTION | PAGE |
|---|---|
| 2.1 SAFETY LIMITS | |
| 2.1.1 REACTOR CORE | 2-1 |
| 2.1.2 REACTOR COOLANT SYSTEM PRESSURE | 2-1 |
| FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT (UNIT 1) | 2-2 |
| FIGURE 2.1-1a REACTOR CORE SAFETY LIMIT (UNIT 2) | 2-2a |
| 2.2 LIMITING SAFETY SYSTEM SETTINGS | |
| 2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS | 2-3 |
| TABLE 2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS (UNIT 1) | 2-4 |
| TABLE 2.2-1a REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS (UNIT 2) | 2-12 |


## BASES


| SECTION | PAGE |
|---|---|
| 2.1 SAFETY LIMITS | |
| 2.1.1 REACTOR CORE | B 2-1 |
| 2.1.2 REACTOR COOLANT SYSTEM PRESSURE | B 2-2 |
| 2.2 LIMITING SAFETY SYSTEM SETTINGS | |
| 2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS | B 2-3 |

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS


| SECTION | | PAGE |
|---|---|---|
| 3/4.2 POWER DISTRIBUTION LIMITS | | |
| 3/4.2.1 | AXIAL FLUX DIFFERENCE (UNIT 1) | 3/4 2-1 |
| 3/4.2.1 | AXIAL FLUX DIFFERENCE (UNIT 2) | 3/4 2-1a |
| 3/4.2.2 | HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$ | 3/4 2-3 |
| 3/4.2.3 | NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$ | 3/4 2-8 |
| 3/4.2.4 | QUADRANT POWER TILT RATIO | 3/4 2-10 |
| 3/4.2.5 | DNB PARAMETERS | 3/4 2-13 |
| 3/4.3 INSTRUMENTATION | | |
| 3/4.3.1 | REACTOR TRIP SYSTEM INSTRUMENTATION | 3/4 3-1 |
| TABLE 3.3-1 | REACTOR TRIP SYSTEM INSTRUMENTATION | 3/4 3-2 |
| TABLE 4.3-1 | REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS | 3/4 3-9 |
| 3/4.3.2 | ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION | 3/4 3-15 |
| TABLE 3.3-2 | ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION | 3/4 3-17 |
| TABLE 3.3-3 | ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS | 3/4 3-28 |
| TABLE 4.3-2 | ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS | 3/4 3-36 |
| 3/4.3.3 | MONITORING INSTRUMENTATION | |
| | Radiation Monitoring For Plant Operations | 3/4 3-45 |
| TABLE 3.3-4 | RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS | 3/4 3-46 |
| TABLE 4.3-3 | RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS | 3/4 3-48 |
| | Movable Incore Detectors | 3/4 3-49 |
| | Seismic Instrumentation (Common System) | 3/4 3-50 |

INDEX

BASES


| SECTION | PAGE |
|---|---|
| 3/4.0 APPLICABILITY | B 3/4 0-1 |
| 3/4.1 REACTIVITY CONTROL SYSTEMS | |
| 3/4.1.1 BORATION CONTROL | B 3/4 1-1 |
| 3/4.1.2 BORATION SYSTEMS | B 3/4 1-2 |
| 3/4.1.3 MOVABLE CONTROL ASSEMBLIES | B 3/4 1-3 |
| 3/4.2 POWER DISTRIBUTION LIMITS (UNIT 1) | B 3/4 2-1 |
| 3/4.2.1 AXIAL FLUX DIFFERENCE | B 3/4 2-1 |
| 3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$ | B 3/4 2-2 |
| 3/4.2.4 QUADRANT POWER TILT RATIO | B 3/4 2-5 |
| 3/4.2.5 DNB PARAMETERS | B 3/4 2-5 |
| 3/4.2 POWER DISTRIBUTION LIMITS (UNIT 2) | B 3/4 2-7 |
| 3/4.2.1 AXIAL FLUX DIFFERENCE | B 3/4 2-7 |
| 3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$ | B 3/4 2-8 |
| 3/4.2.4 QUADRANT POWER TILT RATIO | B 3/4 2-11 |
| 3/4.2.5 DNB PARAMETERS | B 3/4 2-11 |
| 3/4.3 INSTRUMENTATION | |
| 3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION | B 3/4 3-1 |
| 3/4.3.3 MONITORING INSTRUMENTATION | B 3/4 3-3 |
| 3/4.3.4 TURBINE OVERSPEED PROTECTION | B 3/4 3-6 |




Correction letter of 10-15-91

INDEX

ADMINISTRATIVE CONTROLS

SECTION


6.4.2 SAFETY REVIEW BOARD (SRB)

- Function..... 6-9
- Composition..... 6-10
- Alternates..... 6-10
- Consultants..... 6-10
- Meeting Frequency..... 6-10
- Quorum..... 6-10
- Review..... 6-11
- Audits..... 6-11
- Records..... 6-12

6.5 REPORTABLE EVENT ACTION..... 6-13

6.6 SAFETY LIMIT VIOLATION..... 6-13

6.7 PROCEDURES AND PROGRAMS..... 6-13

6.8 REPORTING REQUIREMENTS

6.8.1 ROUTINE REPORTS..... 6-17

- Startup Report..... 6-17
- Annual Reports..... 6-17
- Annual Radiological Environmental Surveillance Report..... 6-18
- Semiannual Radioactive Effluent Release Report..... 6-19
- Monthly Operating Reports..... 6-21
- Core Operating Limits Report (UNIT 1)..... 6-21
- Core Operating Limits Report (UNIT 2)..... 6-21b

6.8.2 SPECIAL REPORTS..... 6-21c

6.9 RECORD RETENTION..... 6-22

## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS*

### 2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER (NI-0041, NI-0042, NI-0043, NI-0044), pressurizer pressure (PI-0455A, B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), and the highest operating loop coolant temperature ($T_{avg}$) (TI-0412, TI-0422, TI-0432, TI-0442) shall not exceed the limits shown in Figure 2.1-1 (UNIT 1) or Figure 2.1-1a (UNIT 2).

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.6.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure (PI-0408, PI-0418, PI-0428, PI-0438) shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.6.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.6.1.

*Where specific instrument numbers are provided in parentheses they are for information only, and apply to each unit unless specifically noted (to assist in identifying associated instrument channels or loops) and are not intended to limit the requirements to the specific instruments associated with the number.

THIS PAGE APPLICABLE TO UNIT 1 ONLY

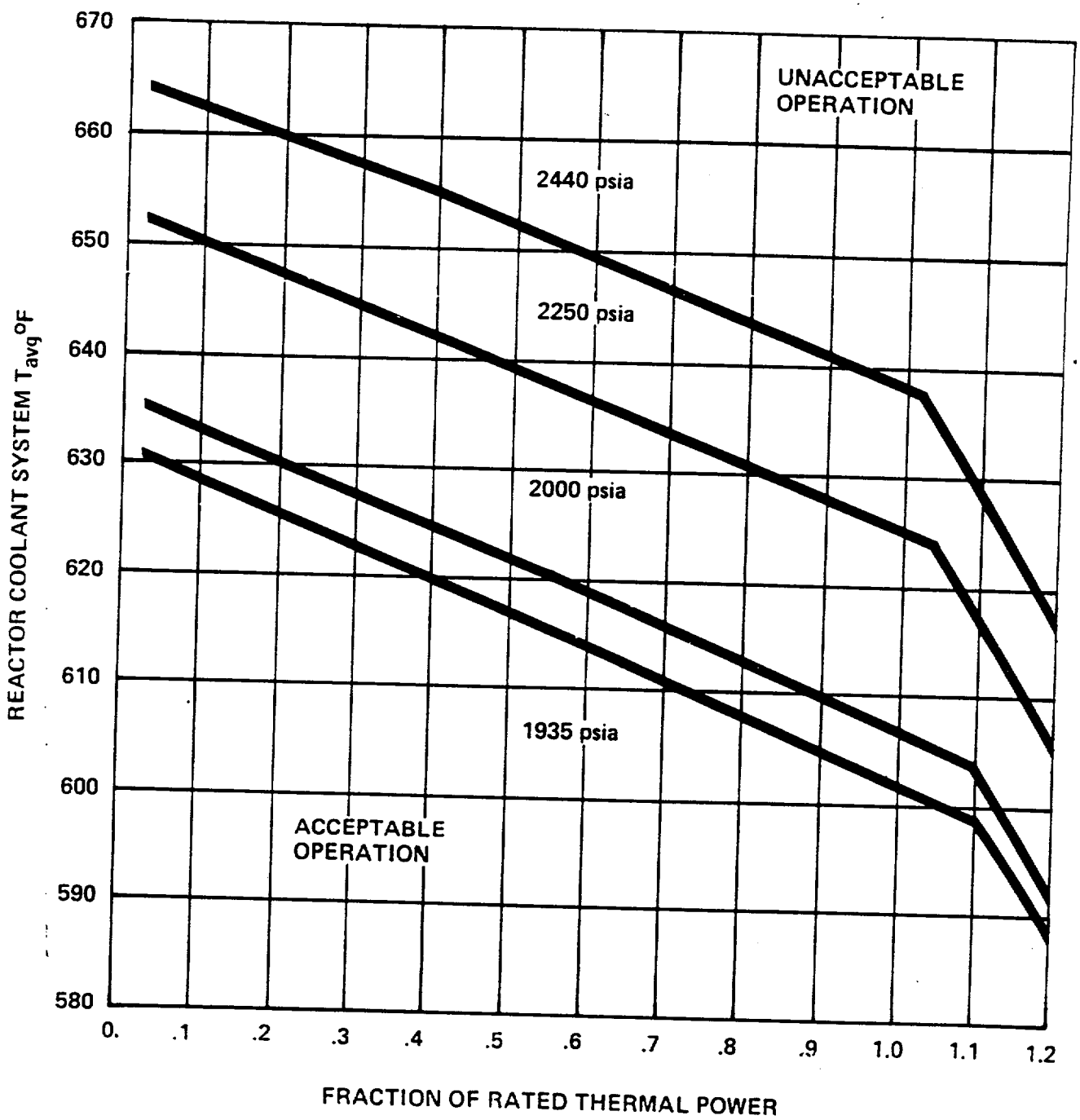


FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - UNIT 1

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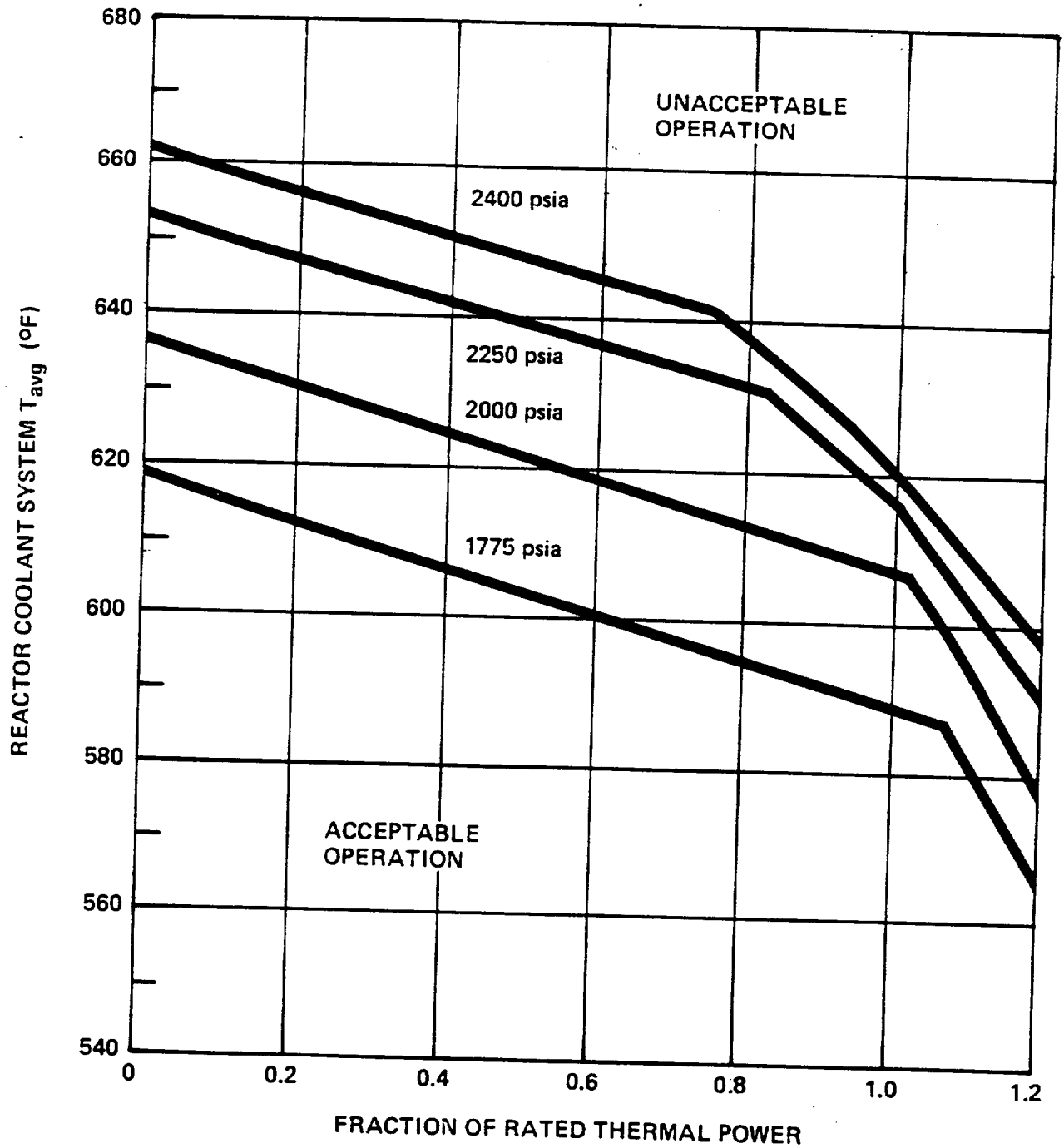


FIGURE 2.1-1a

REACTOR CORE SAFETY LIMIT - UNIT 2

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1 (UNIT 1) or Table 2.2-1a (UNIT 2).

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1 or Table 2.2-1a, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1 or Table 2.2-1a, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 or Table 2.2-1a and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 or Table 2.2-1a for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 or Table 2.2-1a for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 or Table 2.2-1a for the affected channel.

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 - UNIT 1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | N.A. | N.A. | N.A. | N.A. | N.A. |
| 2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | | | | | |
| a. High Setpoint | 7.5 | 4.56 | 0 | ≤109% of RTP# | ≤111.3% of RTP# |
| b. Low Setpoint | 8.3 | 4.56 | 0 | ≤25% of RTP# | ≤27.3% of RTP# |
| 3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | 1.6 | 0.50 | 0 | ≤5% of RTP# with a time constant ≥2 seconds | ≤6.3% of RTP# with a time constant ≥2 seconds |
| 4. Power Range, Neutron Flux, High Negative Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | 1.6 | 0.50 | 0 | ≤5% of RTP# with a time constant ≥2 seconds | ≤6.3% of RTP# with a time constant ≥2 seconds |
| 5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B) | 17.0 | 8.41 | 0 | ≤25% of RTP# | ≤31.1% of RTP# |
| 6. Source Range, Neutron Flux (NI-0031B, NI-0032B) | 17.0 | 10.01 | 0 | $\leq 10^5$ cps | $\leq 1.4 \times 10^5$ cps |
| 7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C) | 10.7 (UNIT 1) | 7.04 (UNIT 1) | 1.96 + 1.17 (UNIT 1) | See Note 1 | See Note 2 |
| 8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B) | 4.3 (UNIT 1) | 1.54 | 1.96 (UNIT 1) | See Note 3 | See Note 4 |

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 1

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A) | 3.1 | 0.71 | 1.67 | ≥1960 psig** | ≥1950 psig |
| 10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A) | 3.1 | 0.71 | 1.67 | ≤2385 psig | ≤2395 psig |
| 11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461) | 8.0 | 2.18 | 1.67 | ≤92% of instrument span | ≤93.9% of instrument span |
| 12. Reactor Coolant Flow-Low (LOOP1 FI-0414 FI-0415 FI-0416; LOOP2 FI-0424 FI-0425 FI-0426; LOOP3 FI-0434 FI-0435 FI-0436; LOOP4 FI-0444 FI-0445 FI-0446) | 2.5 | 1.87 | 0.60 | ≥90% of loop design flow* | ≥89.4% of loop design flow* |
| 13. Steam Generator Water Level Low-Low (LOOP1 LI-0517 LI-0518 LI-0519 LI-0551; LOOP2 LI-0527 LI-0528 LI-0529 LI-0552; LOOP3 LI-0537 LI-0538 LI-0539 LI-0553; LOOP4 LI-0547 LI-0548 LI-0549 LI-0554) | 18.5 (21.8)*** | 17.18 (18.21)*** | 1.67 | ≥18.5% (37.8)*** of narrow range instrument span | ≥17.8% (35.9)*** of narrow range instrument span |
| 14. Undervoltage - Reactor Coolant Pumps | 6.0 | 0.58 | 0 | >9600 volts (70% bus voltage) | >9481 volts (69% bus voltage) |
| 15. Underfrequency - Reactor Coolant Pumps | 3.3 | 0.50 | 0 | ≥57.3 Hz | ≥57.1 Hz |

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrument that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 1

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 16. Turbine Trip | | | | | |
| a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163) | N.A. | N.A. | N.A. | ≥580 psig | ≥500 psig |
| b. Turbine Stop Valve Closure | N.A. | N.A. | N.A. | ≥96.7% open | ≥96.7% open |
| 17. Safety Injection Input from ESF | N.A. | N.A. | N.A. | N.A. | N.A. |
| 18. Reactor Trip System Interlocks | | | | | |
| a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B) | N.A. | N.A. | N.A. | $\geq 1 \times 10^{-10}$ amp | $\geq 6 \times 10^{-11}$ amp |
| b. Low Power Reactor Trips Block, P-7 | | | | | |
| 1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A. | ≤10% of RTP# | ≤12.3% of RTP# |
| 2) P-13 input (PI-0505, PI-0506) | N.A. | N.A. | N.A. | <10% RTP# Turbine Impulse Pressure Equivalent | <12.3% RTP# Turbine Impulse Pressure Equivalent |
| c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A. | ≤48% of RTP# | ≤50.3% of RTP# |

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 1

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| d. Power Range Neutron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A. | ≤50% of RTP# | ≤52.3% of RTP# |
| e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A | ≥10% of RTP# | ≥7.7% of RTP# |
| f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506) | N.A. | N.A. | N.A. | <10% RTP# Turbine Impulse Pressure Equivalent | <12.3% RTP# Turbine Impulse Pressure Equivalent |
| 19. Reactor Trip Breakers | N.A. | N.A. | N.A | N.A. | N.A. |
| 20. Automatic Trip and Interlock Logic | N.A. | N.A. | N.A. | N.A. | N.A. |

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS - UNIT 1

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T \left(\frac{1}{1 + \tau_6 S}\right) - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | Measured ΔT by RTD Manifold Instrumentation |
| $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ | = | Lead-lag compensator on measured ΔT; |
| $\tau_1, \tau_2$ | = | Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s; |
| $\frac{1}{1 + \tau_3 S}$ | = | Lag compensator on measured ΔT; |
| $\tau_3$ | = | Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s; |
| $\Delta T_0$ | = | Indicated ΔT at RATED THERMAL POWER; |
| $K_1$ | $\leq$ | 1.12 (UNIT 1); |
| $K_2$ | = | 0.0224/°F (UNIT 1); |
| $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ | = | The function generated by the lead-lag compensator for $T_{avg}$ dynamic compensation; |
| $\tau_4, \tau_5$ | = | Time constants utilized in the lead-lag compensator for $T_{avg}$, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s; |
| $T$ | = | Average temperature, °F; |
| $\frac{1}{1 + \tau_6 S}$ | = | Lag compensator on measured $T_{avg}$; |
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6 = 0$ s; |

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 1

NOTE 1: (Continued)

$T'$ $\leq$ 588.4°F (UNIT 1) (Nominal $T_{avg}$ operating temperature);

$K_3$ = 0.00115/psig (UNIT 1);

$P$ = Pressurizer pressure, psig;

$P'$ = 2235 psig (Nominal RCS operating pressure);

$S$ = Laplace transform variable, $s^{-1}$;

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -32.0% (UNIT 1) and + 11.0% (UNIT 1), $f_1(\Delta I) = 0$, where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -32.0% (UNIT 1), the ΔT Trip Setpoint shall be automatically reduced by 3.25% (UNIT 1) of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 11.0% (UNIT 1), the ΔT Trip Setpoint shall be automatically reduced by 1.97% (UNIT 1) of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.1% (UNIT 1) of ΔT span.

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 1

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{(1+\tau_3 S)}\right) \leq \Delta T_o \{K_4 - K_5 \frac{(\tau_7 S)}{(1+\tau_7 S)} \left(\frac{1}{(1+\tau_6 S)}\right) T - K_6 [T \left(\frac{1}{(1+\tau_6 S)}\right) - T''] - f_2(\Delta I)\}$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | Measured ΔT by RTD manifold instrumentation; |
| $\frac{1+\tau_1 S}{1+\tau_2 S}$ | = | Lead-lag compensator on measured ΔT; |
| $\tau_1, \tau_2$ | = | Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s; |
| $\frac{1}{1+\tau_3 S}$ | = | Lag compensator on measured ΔT; |
| $\tau_3$ | = | Time constants utilized in the lag compensator for ΔT, $\tau_3$=0s; |
| $\Delta T_o$ | = | Indicated ΔT at RATED THERMAL POWER; |
| $K_4$ | $\leq$ | 1.08 (UNIT 1), |
| $K_5$ | $\geq$ | 0.02/°F for increasing average temperature and $\geq$ 0 for decreasing average temperature, |
| $\frac{\tau_7 S}{1+\tau_7 S}$ | = | The function generated by the rate-lag compensator for $T_{avg}$ dynamic compensation, |
| $\tau_7$ | = | Time constants utilized in the rate-lag compensator for $T_{avg}$, $\tau_7 \geq 10$ s, |
| $\frac{1}{1+\tau_6 S}$ | = | Lag compensator on measured $T_{avg}$; |

THIS PAGE APPLICABLE TO UNIT 1 ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued) - UNIT 1

NOTE 3: (Continued)

| | | |
|---|---|---|
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6 = 0$ s; |
| $K_6$ | $\geq$ | 0.0020/°F (UNIT 1) for $T > T''$ and $K_6 = 0$ for $T \leq T''$, |
| T | = | Average Temperature, °F; |
| T" | = | Indicated $T_{avg}$ at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.4°F (UNIT 1), |
| S | = | Laplace transform variable, $s^{-1}$; and |
| $f_2(\Delta I)$ | = | 0 for all ΔI. |

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% (UNIT 1) of ΔT span.

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a - UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | N.A. | N.A. | N.A. | N.A. | N.A. |
| 2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | | | | | |
| a. High Setpoint | 7.5 | 4.56 | 0 | ≤109% of RTP# | ≤111.3% of RTP# |
| b. Low Setpoint | 8.3 | 4.56 | 0 | ≤25% of RTP# | ≤27.3% of RTP# |
| 3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | 1.6 | 0.50 | 0 | ≤5% of RTP# with a time constant ≥2 seconds | ≤6.3% of RTP# with a time constant ≥2 seconds |
| 4. Power Range, Neutron Flux, High Negative Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | 1.6 | 0.50 | 0 | ≤5% of RTP# with a time constant ≥2 seconds | ≤6.3% of RTP# with a time constant ≥2 seconds |
| 5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B) | 17.0 | 8.41 | 0 | ≤25% of RTP# | ≤31.1% of RTP# |
| 6. Source Range, Neutron Flux (NI-0031B, NI-0032B) | 17.0 | 10.01 | 0 | $\leq 10^5$ cps | $\leq 1.4 \times 10^5$ cps |
| 7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C) | 6.6 (UNIT 2) | 3.37 (UNIT 2) | 1.95 + 0.50 (UNIT 2) | See Note 1 | See Note 2 |
| 8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B) | 4.9 (UNIT 2) | 1.54 | 1.95 (UNIT 2) | See Note 3 | See Note 4 |

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A) | 3.1 | 0.71 | 1.67 | ≥1960 psig** | ≥1950 psig |
| 10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A) | 3.1 | 0.71 | 1.67 | ≤2385 psig | ≤2395 psig |
| 11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461) | 8.0 | 2.18 | 1.67 | ≤92% of instrument span | ≤93.9% of instrument span |
| 12. Reactor Coolant Flow-Low (LOOP1 FI-0414 FI-0415 FI-0416; LOOP2 FI-0424 FI-0425 FI-0426; LOOP3 FI-0434 FI-0435 FI-0436; LOOP4 FI-0444 FI-0445 FI-0446) | 2.5 | 1.87 | 0.60 | >90% of loop design flow* | >89.4% of loop design flow* |
| 13. Steam Generator Water Level Low-Low (LOOP1 LI-0517 LI-0518 LI-0519 LI-0551; LOOP2 LI-0527 LI-0528 LI-0529 LI-0552; LOOP3 LI-0537 LI-0538 LI-0539 LI-0553; LOOP4 LI-0547 LI-0548 LI-0549 LI-0554) | 18.5 (21.8)*** | 17.18 (18.21)*** | 1.67 | ≥18.5% (37.8)*** of narrow range instrument span | ≥17.8% (35.9)*** of narrow range instrument span |
| 14. Undervoltage - Reactor Coolant Pumps | 6.0 | 0.58 | 0 | >9600 volts (70% bus voltage) | >9481 volts (69% bus voltage) |
| 15. Underfrequency - Reactor Coolant Pumps | 3.3 | 0.50 | 0 | ≥57.3 Hz | ≥57.1 Hz |

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrument that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 16. Turbine Trip | | | | | |
| a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163) | N.A. | N.A. | N.A. | ≥580 psig | ≥500 psig |
| b. Turbine Stop Valve Closure | N.A. | N.A. | N.A. | ≥96.7% open | ≥96.7% open |
| 17. Safety Injection Input from ESF | N.A. | N.A. | N.A. | N.A. | N.A. |
| 18. Reactor Trip System Interlocks | | | | | |
| a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B) | N.A. | N.A. | N.A. | $\geq 1 \times 10^{-10}$ amp | $\geq 6 \times 10^{-11}$ amp |
| b. Low Power Reactor Trips Block, P-7 | | | | | |
| 1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A. | ≤10% of RTP# | ≤12.3% of RTP# |
| 2) P-13 input (PI-0505, PI-0506) | N.A. | N.A. | N.A. | <10% RTP# Turbine Impulse Pressure Equivalent | <12.3% RTP# Turbine Impulse Pressure Equivalent |
| c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A. | ≤48% of RTP# | ≤50.3% of RTP# |

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS - UNIT 2

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| d. Power Range Neutron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A. | ≤50% of RTP# | ≤52.3% of RTP# |
| e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A | ≥10% of RTP# | ≥7.7% of RTP# |
| f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506) | N.A. | N.A. | N.A. | <10% RTP# Turbine Impulse Pressure Equivalent | <12.3% RTP# Turbine Impulse Pressure Equivalent |
| 19. Reactor Trip Breakers | N.A. | N.A. | N.A | N.A. | N.A. |
| 20. Automatic Trip and Interlock Logic | N.A. | N.A. | N.A. | N.A. | N.A. |

#RTP = RATED THERMAL POWER

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a (Continued)

TABLE NOTATIONS - UNIT 2

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | Measured ΔT by RTD Manifold Instrumentation (UNIT 2); |
| $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ | = | Lead-lag compensator on measured ΔT; |
| $\tau_1, \tau_2$ | = | Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s; |
| $\frac{1}{1 + \tau_3 S}$ | = | Lag compensator on measured ΔT; |
| $\tau_3$ | = | Time constants utilized in the lag compensator for ΔT, $\tau_3 = 0$ s; |
| $\Delta T_o$ | = | Indicated ΔT at RATED THERMAL POWER; |
| $K_1$ | $\leq$ | 1.10 (UNIT 2); |
| $K_2$ | = | 0.012/°F (UNIT 2); |
| $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ | = | The function generated by the lead-lag compensator for $T_{avg}$ dynamic compensation; |
| $\tau_4, \tau_5$ | = | Time constants utilized in the lead-lag compensator for $T_{avg}$, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s; |
| $T$ | = | Average temperature, °F; |
| $\frac{1}{1 + \tau_6 S}$ | = | Lag compensator on measured $T_{avg}$; |
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6 = 0$ s; |

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 1: (Continued)

$T'$ $\leq$ 588.5°F (UNIT 2) (Nominal $T_{avg}$ operating temperature);

$K_3$ = 0.00056/psig (UNIT 2);

$P$ = Pressurizer pressure, psig;

$P'$ = 2235 psig (Nominal RCS operating pressure);

$S$ = Laplace transform variable, $s^{-1}$;

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -33.5% (UNIT 2) and + 6.5% (UNIT 2), $f_1(\Delta I) = 0$, where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds - 33.5% (UNIT 2), the ΔT Trip Setpoint shall be automatically reduced by 1.27% (UNIT 2) of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 6.5% (UNIT 2), the ΔT Trip Setpoint shall be automatically reduced by 0.83% (UNIT 2) of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.5% (UNIT 2).

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{(1)}{(1 + \tau_3 S)} \leq \Delta T_0 \{K_4 - K_5 \frac{(\tau_7 S)}{(1 + \tau_7 S)} \frac{(1)}{(1 + \tau_6 S)} T - K_6 [T \frac{(1)}{(1 + \tau_6 S)} - T''] - f_2(\Delta I)\}$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | Measured ΔT by RTD manifold instrumentation (UNIT 2); |
| $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ | = | Lead-lag compensator on measured ΔT; |
| $\tau_1, \tau_2$ | = | Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s; |
| $\frac{1}{1 + \tau_3 S}$ | = | Lag compensator on measured ΔT; |
| $\tau_3$ | = | Time constants utilized in the lag compensator for ΔT, $\tau_3$=0s; |
| $\Delta T_0$ | = | Indicated ΔT at RATED THERMAL POWER; |
| $K_4$ | $\leq$ | 1.089 (UNIT 2), |
| $K_5$ | $\geq$ | 0.02/°F for increasing average temperature and $\geq$ 0 for decreasing average temperature, |
| $\frac{\tau_7 S}{1 + \tau_7 S}$ | = | The function generated by the rate-lag compensator for $T_{avg}$ dynamic compensation, |
| $\tau_7$ | = | Time constants utilized in the rate-lag compensator for $T_{avg}$, $\tau_7 \geq 10$ s, |
| $\frac{1}{1 + \tau_6 S}$ | = | Lag compensator on measured $T_{avg}$; |

THIS PAGE APPLICABLE TO UNIT 2 ONLY

TABLE 2.2-1a (Continued)

TABLE NOTATIONS (Continued) - UNIT 2

NOTE 3: (Continued)

| | | |
|---|---|---|
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6 = 0$ s; |
| $K_6$ | $\geq$ | 0.0013/°F (UNIT 2) for $T > T''$ and $K_6 = 0$ for $T \leq T''$, |
| T | = | Average Temperature, °F; |
| T" | = | Indicated $T_{avg}$ at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F) (UNIT 2), |
| S | = | Laplace transform variable, $s^{-1}$; and |
| $f_2(\Delta I)$ | = | 0 for all ΔI. |

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.4% (UNIT 2) of ΔT span.

## 2.1.1 REACTOR CORE - UNIT 1

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through correlations which have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNBR thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show reactor core safety limits for a range of THERMAL POWER, REACTOR COOLANT SYSTEM pressure and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is less than the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR satisfies the DNB design criterion (all the other line segments in each curve). The VANTAGE 5 fuel is analyzed using the WRB-2 correlation with design limit DNBR values of 1.24 and 1.23 for the typical cell and thimble cells, respectively. The LOPAR fuel is analyzed using the WRB-1 correlation with design limit DNBR values of 1.23 and 1.22 for the typical and thimble cells, respectively.

C. The hot channel exit quality is not greater than the upper limit of the quality range (including the effect of uncertainties) of the DNB correlations. This is not a limiting criterion for this plant.

## 2.1 SAFETY LIMITS

BASES

### 2.1.1 REACTOR CORE - UNIT 2

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the W-3 (R Grid) correlation. The W-3 (R Grid) DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1a show reactor core safety limits which are determined for a range of reactor operating conditions. The core limits represent the loci of points of THERMAL POWER, REACTOR COOLANT SYSTEM pressure and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR is not less than the design limit (all the other line segments in each curve).

C. The hot channel exit quality is not greater than the upper limit of the quality range of the W-3 (R-Grid) correlation which is 15% (middle line segment on Reactor Coolant System pressure curves, 2400 psia and 2250 psia; this is not a limiting criterion for this plant).

REACTOR CORE (Continued)

These curves are based on an enthalpy hot channel factor, $F_{\Delta H}^{RTP}$, and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F_{\Delta H}^{N}$ at reduced power based on the expression:

$$F_{\Delta H}^{N} = F_{\Delta H}^{RTP} [1 + PF_{\Delta H} (1-P)]$$

Where: $F_{\Delta H}^{RTP}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}^{N}$ specified in the COLR, and

P is the fraction of RTP.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the $f_1$ ($\Delta I$) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature $\Delta T$ trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

## BORATED WATER SOURCE - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 9504 gallons (19% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 99404 gallons (9% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm, and

3) A minimum solution temperature of 44°F (Unit 1) or 54°F (Unit 2) (TI-10982).

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

### SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $\leq$72°F, verify the boric acid storage tank solution temperature is $\geq$ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when it is the source of borated water and the outside air temperature is less than 40°F (UNIT 1) or 50°F (UNIT 2).

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 36674 gallons (81% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 631478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm,

3) A minimum solution temperature of 44°F (UNIT 1) or 54°F (UNIT 2),

4) A maximum solution temperature of 116°F (TI-10982), and

5) RWST Sludge Mixing Pump Isolation Valves capable of closing on RWST low-level.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the Boric Acid Storage Tank inoperable and being used as one of the above required borated water sources, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT for MODE 5 at 200°F; restore the Boric Acid Storage Tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $\leq$ 72°F, verify the boric acid storage tank solution temperature is $\geq$ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when the outside air temperature is less than 40°F (UNIT 1) or 50°F (UNIT 2).

c. At least once per 18 months by verifying that the Sludge Mixing Pump Isolation Valves automatically close upon RWST low-level test signal.

## ROD DROP TIME

### LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to 2.7 (UNIT 1) or 2.2 (UNIT 2) seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg}$ (TI-0412, TI-0422, TI-0432, TI-0442) greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

### SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

THIS PAGE APPLICABLE TO UNIT 1 ONLY

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AXIAL FLUX DIFFERENCE - UNIT 1

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 ABOVE 50 PERCENT RATED THERMAL POWER.*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR,

   1. Either restore the indicated AFD to within the limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux* - High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2) At least once per hour until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside its limits.

*See Special Test Exceptions Specification 3.10.2.

THIS PAGE APPLICABLE TO UNIT 2 ONLY

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AXIAL FLUX DIFFERENCE - UNIT 2

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (flux difference units) about the target flux difference. The target band is specified in the CORE OPERATING LIMITS REPORT (COLR).

The indicated AFD may deviate outside the required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits specified in the COLR and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.* #

ACTION:

a. With the indicated AFD outside of the required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits specified in the COLR and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux* - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

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* See Special Test Exceptions Specification 3.10.2.

\# Surveillance testing of the Power Range Neutron Flux Channel may be performed (below 90% of RATED THERMAL POWER) pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION - UNIT 2

ACTION (Continued)

c. With the indicated AFD outside of the required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the required target band and the cumulative penalty deviation has been reduced to less than 1 hour in the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

## 3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

### LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} [K(Z)] \text{ for } P \leq 0.5$$

Where: $F_Q^{RTP}$ = the $F_Q$ limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the normalized $F_Q(Z)$ as a function of core height specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of $K_4$) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS - UNIT 1

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Determining the computed heat flux hot channel factor $F_Q^C(Z)$, as follows:

Increase the measured $F_Q(Z)$ obtained from the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties.

c. Verifying that $F_Q^C(Z)$, obtained in Specification 4.2.2.2b. above, satisfies the relationship in Specification 3.2.2.

d. Satisfying the following relationship:

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \text{ for } P > 0.5$$

$$F_Q^C(7) \leq \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \text{ for } P \leq 0.5$$

Where $F_Q^C(Z)$ is obtained in Specification 4.2.2.2b above, $F_Q^{RTP}$ is the $F_Q$ limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation.
$F_Q^{RTP}$, K(Z), and W(Z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.8.1.6.

e. Measuring $F_Q(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

*During power escalation after each fuel loading, power level may be increased until equilibrium conditions at any power level greater than or equal to 50% of RATED THERMAL POWER have been achieved and a power distribution map obtained.

f. With measurements indicating

$$\text{maximum over} \left( \frac{F_Q^C(Z)}{K(Z)} \right)$$

has increased since the previous determination of $F_Q^C(Z)$ either of the following actions shall be taken:

1) Increase $F_Q^C(Z)$ by 2% and verify that this value satisfies the relationship in Specification 4.2.2.2d, or

2) $F_Q^C(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\text{maximum over} \left( \frac{F_Q^C(Z)}{K(Z)} \right) \text{ is not increasing.}$$

g. With the relationships specified in Specification 4.2.2.2d above not being satisfied:

1) Calculate the percent $F_Q(Z)$ exceeds its limits by the following expression:

$$\left\{ \left( \text{maximum over } Z \left[ \frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)} \right] \right) \right\} -1 \quad \times 100 \text{ for } P > 0.5$$

$$\left\{ \left( \text{maximum over } Z \left[ \frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)} \right] \right) \right\} -1 \quad \times 100 \text{ for } P \leqq 0.5, \text{ and}$$

2) The following action shall be taken.

Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2g.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits.

h. The limits specified in Specification 4.2.2.2c are applicable in all core plane regions, i.e., 0 - 100%, inclusive.

i. The limits specified in Specifications 4.2.2.2d, 4.2.2.2f, and 4.2.2.2g above are not applicable in the following core plane regions:

1) Lower core region from 0 to 15%, inclusive.
2) Upper core region from 85 to 100%, inclusive.

4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_{xy}$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER before exceeding 75% of RATED THERMAL POWER following each fuel loading.

b. Increasing the measured $F_{xy}$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

c. Comparing the $F_{xy}$ computed ($F_{xy}^C$) obtained in Specification 4.2.2.2b., above to:

1) The $F_{xy}$ limits for RATED THERMAL POWER ($F_{xy}^{RTP}$) for the appropriate measured core planes given in Specification 4.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1+PF_{xy}(1-P)],$$

Where $F_{xy}^L$ is the limit for fractional THERMAL POWER operation expressed as a function of $F_{xy}^{RTP}$, $PF_{xy}$ is the power factor multiplier for $F_{xy}$ specified in the COLR, and P is the fraction of RATED THERMAL POWER at which $F_{xy}$ was measured.

d. Remeasuring $F_{xy}$ according to the following schedule:

1) When $F_{xy}^C$ is greater than the $F_{xy}^{RTP}$ limit for the appropriate measured core plane but less than the $F_{xy}^L$ relationship, additional power distribution maps shall be taken and $F_{xy}^C$ compared to $F_{xy}^{RTP}$ and $F_{xy}^L$ either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which $F_{xy}^C$ was last determined, or

b) At least once per 31 Effective Full Power Days (EFPD), whichever occurs first.

2) When the $F_{xy}^{C}$ is less than or equal to the $F_{xy}^{RTP}$ limit for the appropriate measured core plane, additional power distribution maps shall be taken and $F_{xy}^{C}$ compared to $F_{xy}^{RTP}$ and $F_{xy}^{L}$ at least once per 31 EFPD.

e. The $F_{xy}$ limits used in the Constant Axial Offset Control analysis for RATED THERMAL POWER ($F_{xy}^{RTP}$) shall be specified for all core planes containing Bank "D" control rods and all unrodded core planes in the COLR per Specification 6.8.1.6;

f. The $F_{xy}$ limits of Specification 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2%, and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height [± 2.88 inches] about the bank demand position of the Bank "D" control rods.

g. With $F_{xy}^{C}$ exceeding $F_{xy}^{L}$ the effects of $F_{xy}$ on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than $F_{xy}$ determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

## 3/4.2.5 DNB PARAMETERS

### LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits:

a. Reactor Coolant System $T_{avg}$ (TI-0412, TI-0422, TI-0432, TI-0442), $\leq$ 592.5°F (UNIT 1) or 591°F (UNIT 2)

b. Pressurizer Pressure (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), $\geq$ 2199 psig* (UNIT 1) or 2224 psig* (UNIT 2)

c. Reactor Coolant System Flow (FI-0414, FI-0415, FI-0416, FI-0424, FI-0425, FI-0426, FI-0434, FI-0435, FI-0436, FI-0444, FI-0445, FI-0446) $\geq$ 393,136 gpm** (UNIT 1) or 396,198 gpm** (UNIT 2)

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System $T_{avg}$ and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours. RCS flow rate shall be monitored for degradation at least once per 12 hours. In the event of flow degradation, RCS flow rate shall be determined by precision heat balance within 7 days of detection of flow degradation.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at each fuel loading and at least once per 18 months.

4.2.5.3 After each fuel loading, the RCS flow rate shall be determined by precision heat balance within 7 days after exceeding 90% RATED THERMAL POWER (UNIT 1) or prior to operation above 75% RATED THERMAL POWER (UNIT 2). The RCS flow rate shall also be determined by precision heat balance at least once per 18 months. Within 7 days prior to performing the precision heat balance flow measurement, the instrumentation used for performing the precision heat balance shall be calibrated. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

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*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**Includes a 2.7% (UNIT 1) or 3.5% (UNIT 2) flow measurement uncertainty.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 7. Semi-Automatic Switchover to Containment Emergency Sump (Continued) | | | | | |
| b. RWST Level--Low-Low Coincident With Safety Injection (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A) | 3.5 | 0.71 | 1.67 | ≥275.3 in. from tank base (≥39.1% of instrument span) | ≥264.9 in. from tank base (≥37.4% of instrument span) |
| 8. Loss of Power to 4.16 kV ESF Bus | | | | | |
| a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage | N.A. | N.A. | N.A. | ≥2975 volts with a ≤ 0.8 second time delay. | ≥2912 volts with a ≤ 0.8 second time delay. |
| b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage | N.A. | N.A. | N.A. | ≥3746 volts with a <20 second time delay. | ≥3683 volts with a <20 second time delay. |
| 9. Engineered Safety Features Actuation System Interlocks | | | | | |
| a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A) | N.A. | N.A. | N.A. | ≤ 2000 psig (UNIT 1)<br>≤ 1970 psig (UNIT 2) | ≤ 2010 psig (UNIT 1)<br>≤ 1980 psig (UNIT 2) |
| b. Reactor Trip, P-4 | N.A. | N.A. | N.A. | N.A. | N.A. |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 10. Control Room Emergency Filtration System Actuation | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| c. Safety Injection | See Functional Unit 1. above for all Safety Injection Trip Setpoints and Allowable Values. | | | | |
| d. Intake Radiogas monitor (RE-12116, RE-12117) | N.A. | N.A. | N.A. | ≤ 3X background | N.A. |
| 11. Fuel Handling Building Post-Accident Ventilation Actuation (Common System) | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| b. Fuel Handling Building Exhaust Duct Radiation Signal (ARE-2532A&B, ARE-2533A&B) | N.A. | N.A. | N.A. | c | N.A. |
| c. Automatic Actuation Logic And Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS

### 3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. UNIT 1: A contained borated water volume of between 6555 (29.2% of instrument span) and 6909 gallons (70.7% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

UNIT 2: A contained borated water volume of between 6616 (36% of instrument span) and 6854 gallons (64% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

c. A boron concentration of between 1900 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B, PI-0967A&B).

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open (HV-8808A, B, C, D).

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume (67 gallons) by verifying the boron concentration of the accumulator solution; and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the isolation valve operator is open.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

## 3/4.5.3 ECCS SUBSYSTEMS - $T_{avg}$ LESS THAN 350°F

### SAFETY INJECTION PUMPS

### LIMITING CONDITION FOR OPERATION

3.5.3.2 All Safety Injection pumps shall be inoperable.

APPLICABILITY MODES 4, 5, and 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

### SURVEILLANCE REQUIREMENTS

4.5.3.2 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3, prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, and at least once per 31 days thereafter.

* An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

## 3/4.5.4 REFUELING WATER STORAGE TANK

### LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 631,478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A).

b. A boron concentration of between 2400 ppm and 2600 ppm of boron,

c. A minimum solution temperature of 44°F (UNIT 1) or 54°F (UNIT 2), and

d. A maximum solution temperature of 116°F (TI-10982)

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

APPLICABILITY: MODES 1, 2, 3 and 4.
ACTION:

a. With the RWST inoperable except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

### SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:
   1) Verifying the contained borated water volume in the tank, and
   2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F (UNIT 1) or 50°F (UNIT 2).

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

## 3/4.2 POWER DISTRIBUTION LIMITS - UNIT 1

### BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) meeting the DNB design criterion during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods; and

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

### 3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the $F_Q$ limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

AXIAL FLUX DIFFERENCE (Continued)

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the allowed ΔI power operating space for RAOC operation specified within the COLR and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limit on peak local power density is not exceeded, (2) the DNB design criterion is met, and (3) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod banks are sequenced with a constant tip-to-tip distance between banks as described in Specification 3.1.3.6;

Deleted for Unit 1 Only

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an $F_Q$ measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor $F_Q(Z)$ is met. W(Z) accounts for the effects of normal operation transients with the AFD band and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(Z) function for normal operation and the AFD band are provided in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to meet the DNB design criterion throughout each analyzed transient. The indicated $T_{avg}$ value of 592.5°F and the indicated pressurizer pressure value of 2199 psig correspond to analytical limits of 594.4°F and 2185 psig respectively, with allowance for measurement uncertainty.

3/4.2.5 DNB PARAMETERS (Continued)

The measurement uncertainty for the RCS total flow is based upon performing a precision heat balance flow measurement above 90% RATED THERMAL POWER and using the results to correlate the flow indication channels with the measured flow. If a precision heat balance flow measurement is performed below 90% RATED THERMAL POWER, the effect on the measurement uncertainty shall be taken into account. Potential fouling of the feedwater venturis which might not be detected could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected feedwater venturi fouling is included in the measurement uncertainty. Any fouling which might bias the RCS flow rate measurement greater than 0.1% may be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance flow measurements, i.e., either the effect of the fouling shall be quantified and accounted for in the RCS flow rate measurement or the affected venturis shall be cleaned to eliminate the fouling. The indicated RCS flow value of 393,136 gpm corresponds to an analytical value of 382,800 gpm with allowance for measurement and indication uncertainties.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

## 3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the $F_Q$ limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits specified in the COLR while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod banks are sequenced with a constant tip-to-tip distance between banks as described in Specification 3.1.3.6;

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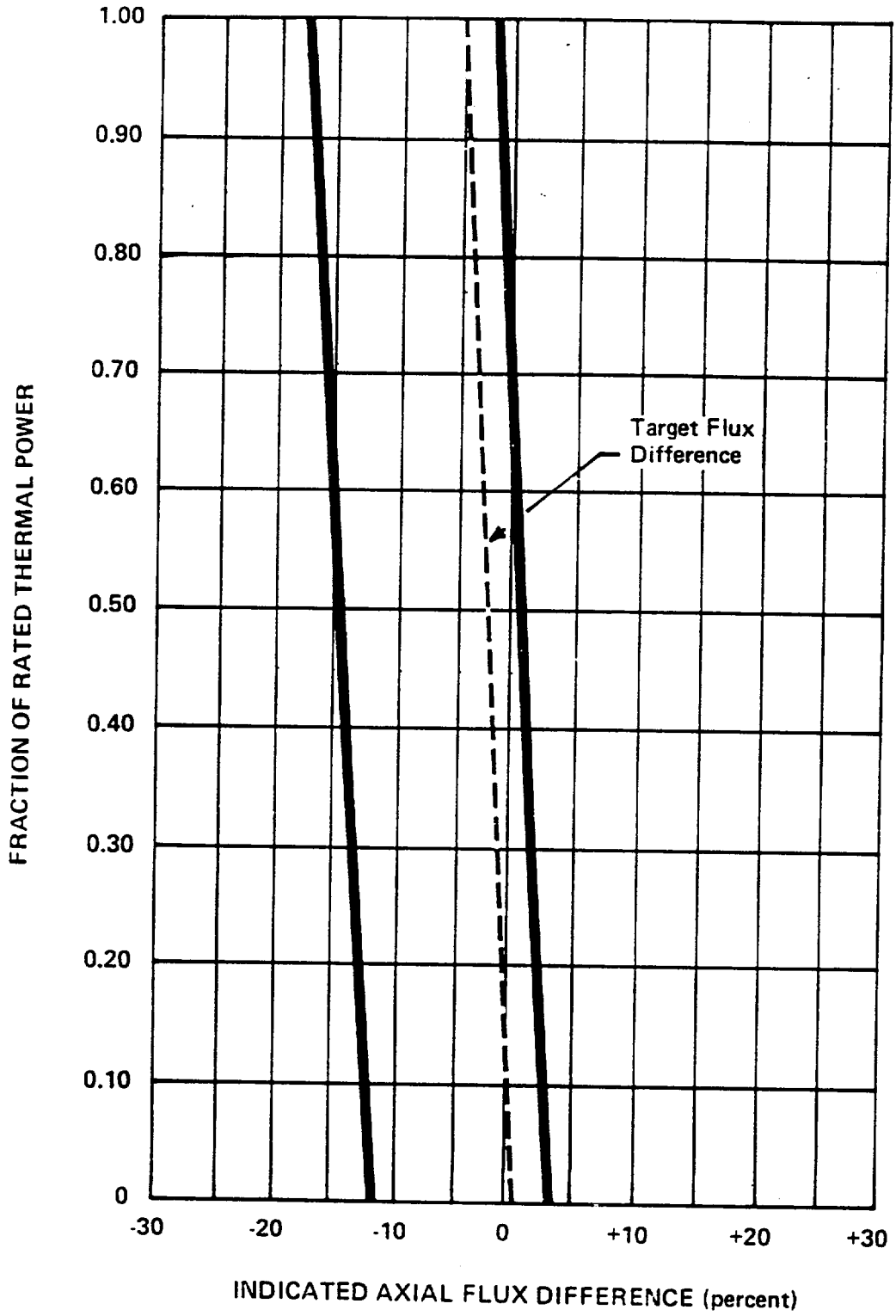


FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an $F_Q$ measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. This margin includes the following:

a. Design limit DNBR of 1.30 vs 1.28,

b. Grid Spacing ($K_s$) of 0.046 vs 0.059,

c. Thermal Diffusion Coefficient of 0.038 vs 0.059,

d. DNBR Multiplier of 0.86 vs 0.88, and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The $F_{xy}$ limit for RATED THERMAL POWER ($F_{xy}^{RTP}$) as specified in the COLR per Specification 6.8.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated $T_{avg}$ value of 591°F and the indicated pressurizer pressure value of 2224 psig correspond to analytical limits of 592.5°F and 2205 psig respectively, with allowance for measurement uncertainty.

3/4.2.5 DNB PARAMETERS (Continued)

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

## 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and meet the DNB design criterion during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two trains/loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR trains be OPERABLE. The locking closed of the required valves, except valves 1208-U4-176 and 1208-U4-177 for short periods of time to maintain chemistry control, in Mode 5 (with the loops not filled) precludes the possibility of uncontrolled boron dilution of the filled portion of the Reactor Coolant System. These actions prevent flow to the RCS of unborated water in excess of that analyzed. These limitations are consistent with the initial conditions assumed for the boron dilution accident in the safety analysis.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. In MODE 4 the starting of an RCP, when no other RCP is operating, is restricted to a range of temperatures that are consistent with the analysis assumptions used to demonstrate that the RHR design pressure is not exceeded when RHR relief valves are used for RCS overpressure protection.

## 3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. The minimum boron concentration must ensure the reactor core will remain subcritical during the accumulator injection period of a small-break LOCA.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

## 3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.9 or 3.3.3.10, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT - UNIT 1

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN LIMIT FOR MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN LIMITS FOR MODES 3, 4 and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z) and W(Z), for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

CORE OPERATING LIMITS REPORT (Continued) - UNIT 1

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).
(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for $F_Q$ Methodology).)

c. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION", February 1982 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

CORE OPERATING LIMITS REPORT - UNIT 2

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN LIMIT FOR MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN LIMITS FOR MODES 3, 4 and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits, and target band for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z), the Power Factor Multiplier and $F_{xy}^{RTP}$, for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

CORE OPERATING LIMITS REPORT (Continued) - UNIT 2

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary).
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

c. T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

d. NUREG-0800, Standard Review Plan, U. S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.
(Methodology for Specification 3.2.1 - Axial Flux Difference [Constant Axial Offset Control].)

e. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION", February 1982 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Facility Operating License No. NPF-68 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated November 29, 1990, as supplemented January 29, March 6, March 27, March 29, April 19, August 8, and August 19, 1991, 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 2.C.(2) of Facility Operating License No. NPF-68 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 44 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective beginning with the initial loading of VANTAGE-5 fuel into Vogtle Unit 2 Cycle 3.

FOR THE NUCLEAR REGULATORY COMMISSION

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: September 19, 1991

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Facility Operating License No. NPF-81 filed by the Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees) dated November 29, 1990, as supplemented January 29, March 6, March 27, March 29, April 19, August 8, and August 19, 1991, 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 2.C.(2) of Facility Operating License No. NPF-81 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 24 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. GPC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective beginning with the initial loading of VANTAGE-5 fuel into Vogtle Unit 2 Cycle 3.

FOR THE NUCLEAR REGULATORY COMMISSION

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: September 19, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 44

FACILITY OPERATING LICENSE NO. NPF-68

AND LICENSE AMENDMENT NO. 24

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NOS. 50-424 AND 50-425

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages effective beginning with initial loading of VANTAGE-5 fuel, Unit 2, Cycle 3. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

| Remove Pages | Insert Pages |
|---|---|
| III | III |
| V | V |
| XV/XVI* | XV/XVI* |
| XXIII | XXIII |
| 2-1, 2-2 | 2-1, 2-2 |
| 2-2a | -- |
| 2-3 - 2-19 | 2-3 - 2-11 |
| B 2-1 | B 2-1 |
| B 2-1a | -- |
| 3/4 1-11, 3/4 1-12 | 3/4 1-11, 3/4 1-12 |
| 3/4 1-13 | 3/4 1-13 |
| 3/4 1-19 | 3/4 1-19 |
| 3/4 2-1, 3/4 2-1a, 3/4 2-2 | 3/4 2-1, 3/4 2-2 |
| 3/4 2-4 | 3/4 2-4 |
| 3/4 2-5, 3/4 2-6 | 3/4 2-5, 3/4 2-6 |
| 3/4 2-7, 3/4 2-7a | 3/4 2-7 |
| 3/4 2-13 | 3/4 2-13 |
| 3/4 3-33 / 3/4 3-34* | 3/4 3-33 / 3/4 3-34* |
| 3/4 5-1 / 3/4 5-2* | 3/4 5-1 / 3/4 5-2* |
| 3/4 5-9* / 3/4 5-10 | 3/4 5-9* / 3/4 5-10 |
| B 3/4 2-1 - B 3/4 2-6 | B 3/4 2-1 - B 3/4 2-5 |
| B 3/4 2-7 - B 3/4 2-12 | -- |
| 6-21 - 6-21c | 6-21, 6-21a |

* Overleaf pages containing no changes

# INDEX

## SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS


| SECTION | PAGE |
|---|---|
| 2.1 SAFETY LIMITS | |
| 2.1.1 REACTOR CORE | 2-1 |
| 2.1.2 REACTOR COOLANT SYSTEM PRESSURE | 2-1 |
| FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT | 2-2 |
| 2.2 LIMITING SAFETY SYSTEM SETTINGS | |
| 2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS | 2-3 |
| TABLE 2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS | 2-4 |


## BASES


| SECTION | PAGE |
|---|---|
| 2.1 SAFETY LIMITS | |
| 2.1.1 REACTOR CORE | B 2-1 |
| 2.1.2 REACTOR COOLANT SYSTEM PRESSURE | B 2-2 |
| 2.2 LIMITING SAFETY SYSTEM SETTINGS | |
| 2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS | B 2-3 |

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS


| SECTION | | PAGE |
|---|---|---|
| 3/4.2 POWER DISTRIBUTION LIMITS | | |
| 3/4.2.1 | AXIAL FLUX DIFFERENCE | 3/4 2-1 |
| 3/4.2.2 | HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$ | 3/4 2-3 |
| 3/4.2.3 | NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$ | 3/4 2-8 |
| 3/4.2.4 | QUADRANT POWER TILT RATIO | 3/4 2-10 |
| 3/4.2.5 | DNB PARAMETERS | 3/4 2-13 |
| 3/4.3 INSTRUMENTATION | | |
| 3/4.3.1 | REACTOR TRIP SYSTEM INSTRUMENTATION | 3/4 3-1 |
| TABLE 3.3-1 | REACTOR TRIP SYSTEM INSTRUMENTATION | 3/4 3-2 |
| TABLE 4.3-1 | REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS | 3/4 3-9 |
| 3/4.3.2 | ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION | 3/4 3-15 |
| TABLE 3.3-2 | ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION | 3/4 3-17 |
| TABLE 3.3-3 | ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS | 3/4 3-28 |
| TABLE 4.3-2 | ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS | 3/4 3-36 |
| 3/4.3.3 | MONITORING INSTRUMENTATION | |
| | Radiation Monitoring For Plant Operations | 3/4 3-45 |
| TABLE 3.3-4 | RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS | 3/4 3-46 |
| TABLE 4.3-3 | RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS | 3/4 3-48 |
| | Movable Incore Detectors | 3/4 3-49 |
| | Seismic Instrumentation (Common System) | 3/4 3-50 |

INDEX

BASES


| SECTION | PAGE |
|---|---|
| 3/4.0 APPLICABILITY | B 3/4 0-1 |
| 3/4.1 REACTIVITY CONTROL SYSTEMS | |
| 3/4.1.1 BORATION CONTROL | B 3/4 1-1 |
| 3/4.1.2 BORATION SYSTEMS | B 3/4 1-2 |
| 3/4.1.3 MOVABLE CONTROL ASSEMBLIES | B 3/4 1-3 |
| 3/4.2 POWER DISTRIBUTION LIMITS | B 3/4 2-1 |
| 3/4.2.1 AXIAL FLUX DIFFERENCE | B 3/4 2-1 |
| 3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$ | B 3/4 2-2 |
| 3/4.2.4 QUADRANT POWER TILT RATIO | B 3/4 2-4 |
| 3/4.2.5 DNB PARAMETERS | B 3/4 2-4 |
| 3/4.3 INSTRUMENTATION | |
| 3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION | B 3/4 3-1 |
| 3/4.3.3 MONITORING INSTRUMENTATION | B 3/4 3-3 |
| 3/4.3.4 TURBINE OVERSPEED PROTECTION | B 3/4 3-6 |




Correction letter of 10-15-91

# INDEX

BASES


| SECTION | PAGE |
|---|---|
| 3/4.4 REACTOR COOLANT SYSTEM | |
| 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION | B 3/4 4-1 |
| 3/4.4.2 SAFETY VALVES | B 3/4 4-2 |
| 3/4.4.3 PRESSURIZER | B 3/4 4-2 |
| 3/4.4.4 RELIEF VALVES | B 3/4 4-3 |
| 3/4.4.5 STEAM GENERATORS | B 3/4 4-3 |
| 3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE | B 3/4 4-4 |
| 3/4.4.7 CHEMISTRY | B 3/4 4-5 |
| 3/4.4.8 SPECIFIC ACTIVITY | B 3/4 4-5 |
| 3/4.4.9 PRESSURE/TEMPERATURE LIMITS | B 3/4 4-7 |
| TABLE B 3/4.4-1a UNIT 1 REACTOR VESSEL TOUGHNESS | B 3/4 4-9 |
| TABLE B 3/4.4-1b UNIT 2 REACTOR VESSEL TOUGHNESS | B 3/4 4-9a |
| FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE | B 3/4 4-11 |
| FIGURE B 3/4.4-2 EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF $RT_{NDT}$ FOR REACTOR VESSELS EXPOSED TO RADIATION AT 550°F | B 3/4 4-12 |
| 3/4.4.10 STRUCTURAL INTEGRITY | B 3/4 4-16 |
| 3/4.4.11 REACTOR COOLANT SYSTEM VENTS | B 3/4 4-17 |
| 3/4.5 EMERGENCY CORE COOLING SYSTEMS | |
| 3/4.5.1 ACCUMULATORS | B 3/4 5-1 |
| 3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS | B 3/4 5-1 |
| 3/4.5.4 REFUELING WATER STORAGE TANK | B 3/4 5-2 |

INDEX

ADMINISTRATIVE CONTROLS

SECTION

6.4.2 SAFETY REVIEW BOARD (SRB)
Function..... 6-9
Composition..... 6-10
Alternates..... 6-10
Consultants..... 6-10
Meeting Frequency..... 6-10
Quorum..... 6-10
Review..... 6-11
Audits..... 6-11
Records..... 6-12

6.5 REPORTABLE EVENT ACTION..... 6-13

6.6 SAFETY LIMIT VIOLATION..... 6-13

6.7 PROCEDURES AND PROGRAMS..... 6-13

6.8 REPORTING REQUIREMENTS

6.8.1 ROUTINE REPORTS..... 6-17
Startup Report..... 6-17
Annual Reports..... 6-17
Annual Radiological Environmental Surveillance Report..... 6-18
Semiannual Radioactive Effluent Release Report..... 6-19
Monthly Operating Reports..... 6-21
Core Operating Limits Report ..... 6-21

6.8.2 SPECIAL REPORTS..... 6-21a

6.9 RECORD RETENTION..... 6-22

## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS*

### 2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER (NI-0041, NI-0042, NI-0043, NI-0044), pressurizer pressure (PI-0455A, B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), and the highest operating loop coolant temperature ($T_{avg}$) (TI-0412, TI-0422, TI-0432, TI-0442) shall not exceed the limits shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.6.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure (PI-0408, PI-0418, PI-0428, PI-0438) shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.6.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.6.1.

*Where specific instrument numbers are provided in parentheses they are for information only, and apply to each unit unless specifically noted (to assist in identifying associated instrument channels or loops) and are not intended to limit the requirements to the specific instruments associated with the number.

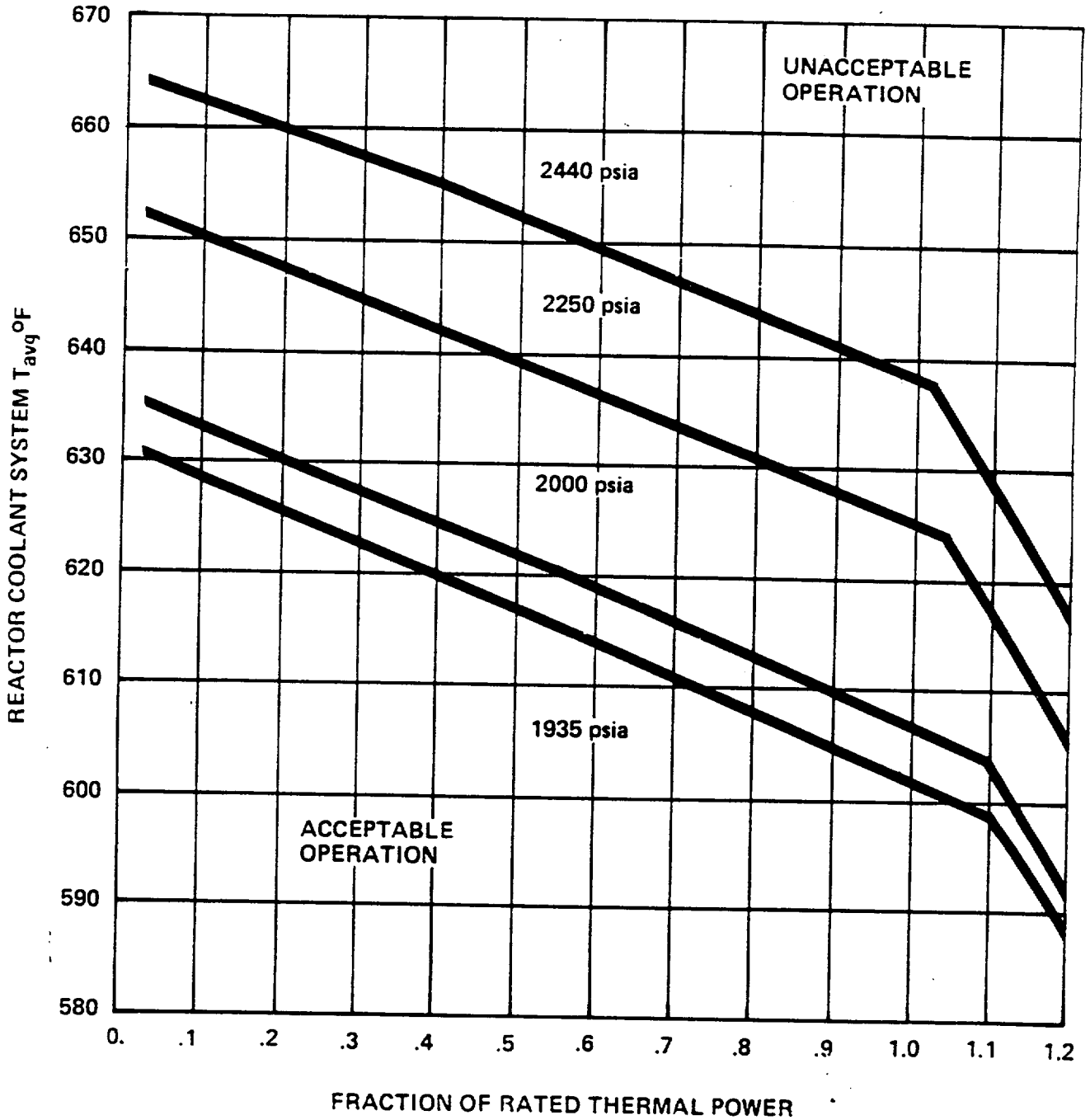


FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | N.A. | N.A. | N.A. | N.A. | N.A. |
| 2. Power Range, Neutron Flux (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | | | | | |
| a. High Setpoint | 7.5 | 4.56 | 0 | ≤109% of RTP# | ≤111.3% of RTP# |
| b. Low Setpoint | 8.3 | 4.56 | 0 | ≤25% of RTP# | ≤27.3% of RTP# |
| 3. Power Range, Neutron Flux, High Positive Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | 1.6 | 0.50 | 0 | ≤5% of RTP# with a time constant ≥2 seconds | ≤6.3% of RTP# with a time constant ≥2 seconds |
| 4. Power Range, Neutron Flux, High Negative Rate (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | 1.6 | 0.50 | 0 | ≤5% of RTP# with a time constant ≥2 seconds | ≤6.3% of RTP# with a time constant ≥2 seconds |
| 5. Intermediate Range, Neutron Flux (NI-0035B, NI-0036B) | 17.0 | 8.41 | 0 | ≤25% of RTP# | ≤31.1% of RTP# |
| 6. Source Range, Neutron Flux (NI-0031B, NI-0032B) | 17.0 | 10.01 | 0 | $\leq 10^5$ cps | $\leq 1.4 \times 10^5$ cps |
| 7. Overtemperature ΔT (TDI-411C, TDI-421C, TDI-431C, TDI-441C) | 10.7 | 7.04 | 1.96 + 1.17 | See Note 1 | See Note 2 |
| 8. Overpower ΔT (TDI-411B, TDI-421B, TDI-431B, TDI-441B) | 4.3 | 1.54 | 1.96 | See Note 3 | See Note 4 |

#RTP = RATED THERMAL POWER





Correction letter of 10-15-91

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 9. Pressurizer Pressure-Low (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A) | 3.1 | 0.71 | 1.67 | ≥1960 psig** | ≥1950 psig |
| 10. Pressurizer Pressure-High (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A) | 3.1 | 0.71 | 1.67 | ≤2385 psig | ≤2395 psig |
| 11. Pressurizer Water Level-High (LI-0459A, LI-0460A, LI-0461) | 8.0 | 2.18 | 1.67 | ≤92% of instrument span | ≤93.9% of instrument span |
| 12. Reactor Coolant Flow-Low (LOOP1 FI-0414 FI-0415 FI-0416; LOOP2 FI-0424 FI-0425 FI-0426; LOOP3 FI-0434 FI-0435 FI-0436; LOOP4 FI-0444 FI-0445 FI-0446) | 2.5 | 1.87 | 0.60 | ≥90% of loop design flow* | ≥89.4% of loop design flow* |
| 13. Steam Generator Water Level Low-Low (LOOP1 LI-0517 LI-0518 LI-0519 LI-0551; LOOP2 LI-0527 LI-0528 LI-0529 LI-0552; LOOP3 LI-0537 LI-0538 LI-0539 LI-0553; LOOP4 LI-0547 LI-0548 LI-0549 LI-0554) | 18.5 (21.8)*** | 17.18 (18.21)*** | 1.67 | ≥18.5% (37.8)*** of narrow range instrument span | ≥17.8% (35.9)*** of narrow range instrument span |
| 14. Undervoltage - Reactor Coolant Pumps | 6.0 | 0.58 | 0 | ≥9600 volts (70% bus voltage) | ≥9481 volts (69% bus voltage) |
| 15. Underfrequency - Reactor Coolant Pumps | 3.3 | 0.50 | 0 | ≥57.3 Hz | ≥57.1 Hz |

*Loop design flow = 95,700 gpm

**Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

***The value stated inside the parenthesis is for instrument that has the lower tap at elevation 333"; the value stated outside the parenthesis is for instrumentation that has the lower tap at elevation 438".





Correction letter of 10-15-91

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 16. Turbine Trip | | | | | |
| a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163) | N.A. | N.A. | N.A. | ≥580 psig | ≥500 psig |
| b. Turbine Stop Valve Closure | N.A. | N.A. | N.A. | ≥96.7% open | ≥96.7% open |
| 17. Safety Injection Input from ESF | N.A. | N.A. | N.A. | N.A. | N.A. |
| 18. Reactor Trip System Interlocks | | | | | |
| a. Intermediate Range Neutron Flux, P-6 (NI-0035B, NI-0036B) | N.A. | N.A. | N.A. | $\geq 1 \times 10^{-10}$ amp | $\geq 6 \times 10^{-11}$ amp |
| b. Low Power Reactor Trips Block, P-7 | | | | | |
| 1) P-10 input (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A. | ≤10% of RTP# | ≤12.3% of RTP# |
| 2) P-13 input (PI-0505, PI-0506) | N.A. | N.A. | N.A. | <10% RTP# Turbine Impulse Pressure Equivalent | <12.3% RTP# Turbine Impulse Pressure Equivalent |
| c. Power Range Neutron Flux, P-8 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A. | ≤48% of RTP# | ≤50.3% of RTP# |

#RTP = RATED THERMAL POWER





Correction letter of 10-15-91

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| d. Power Range Neutron Flux, P-9 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A. | ≤50% of RTP# | ≤52.3% of RTP# |
| e. Power Range Neutron Flux, P-10 (NI-0041B&C, NI-0042B&C, NI-0043B&C, NI-0044B&C) | N.A. | N.A. | N.A | ≥10% of RTP# | ≥7.7% of RTP# |
| f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506) | N.A. | N.A. | N.A. | ≤10% RTP# Turbine Impulse Pressure Equivalent | ≤12.3% RTP# Turbine Impulse Pressure Equivalent |
| 19. Reactor Trip Breakers | N.A. | N.A. | N.A | N.A. | N.A. |
| 20. Automatic Trip and Interlock Logic | N.A. | N.A. | N.A. | N.A. | N.A. |

#RTP = RATED THERMAL POWER





Correction letter of 10-15-91

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE $\Delta T$

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | Measured $\Delta T$ by RTD Manifold Instrumentation |
| $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ | = | Lead-lag compensator on measured $\Delta T$; |
| $\tau_1$, $\tau_2$ | = | Time constants utilized in lead-lag compensator for $\Delta T$, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s; |
| $\frac{1}{1 + \tau_3 S}$ | = | Lag compensator on measured $\Delta T$; |
| $\tau_3$ | = | Time constants utilized in the lag compensator for $\Delta T$, $\tau_3 = 0$ s; |
| $\Delta T_0$ | = | Indicated $\Delta T$ at RATED THERMAL POWER; |
| $K_1$ | $\leq$ | 1.12; |
| $K_2$ | = | 0.0224/°F; |
| $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ | = | The function generated by the lead-lag compensator for $T_{avg}$ dynamic compensation; |
| $\tau_4$, $\tau_5$ | = | Time constants utilized in the lead-lag compensator for $T_{avg}$, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s; |
| $T$ | = | Average temperature, °F; |
| $\frac{1}{1 + \tau_6 S}$ | = | Lag compensator on measured $T_{avg}$; |
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6 = 0$ s; |





Correction letter of 10-15-91

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

$T'$ $\leq$ 588.4°F Nominal $T_{avg}$ operating temperature;

$K_3$ = 0.00115/psig;

$P$ = Pressurizer pressure, psig;

$P'$ = 2235 psig (Nominal RCS operating pressure);

$S$ = Laplace transform variable, $s^{-1}$;

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -32.0% and + 11.0%, $f_1(\Delta I) = 0$, where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -32.0%, the ΔT Trip Setpoint shall be automatically reduced by 3.25% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 11.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.97% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.1% of ΔT span.





Correction letter of 10-15-91

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \frac{(\tau_7 S)}{(1 + \tau_7 S)} \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | Measured ΔT by RTD manifold instrumentation; |
| $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ | = | Lead-lag compensator on measured ΔT; |
| $\tau_1$, $\tau_2$ | = | Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ S, $\tau_2 \leq 3$ s; |
| $\frac{1}{1 + \tau_3 S}$ | = | Lag compensator on measured ΔT; |
| $\tau_3$ | = | Time constants utilized in the lag compensator for ΔT, $\tau_3$=0s; |
| $\Delta T_0$ | = | Indicated ΔT at RATED THERMAL POWER; |
| $K_4$ | $\leq$ | 1.08, |
| $K_5$ | $\geq$ | 0.02/°F for increasing average temperature and $\geq$ 0 for decreasing average temperature, |
| $\frac{\tau_7 S}{1 + \tau_7 S}$ | = | The function generated by the rate-lag compensator for $T_{avg}$ dynamic compensation, |
| $\tau_7$ | = | Time constants utilized in the rate-lag compensator for $T_{avg}$, $\tau_7 \geq 10$ s, |
| $\frac{1}{1 + \tau_6 S}$ | = | Lag compensator on measured $T_{avg}$; |





Corrections letter of 10-15-91

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

| | | |
|---|---|---|
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6 = 0$ s; |
| $K_6$ | $\geq$ | 0.0020/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$, |
| T | = | Average Temperature, °F; |
| T" | = | Indicated $T_{avg}$ at RATED THERMAL POWER (Calibration temperature for $\Delta T$ instrumentation, $\leq$ 588.4°F, |
| S | = | Laplace transform variable, $s^{-1}$; and |
| $f_2(\Delta I)$ | = | 0 for all $\Delta I$. |

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.9% of $\Delta T$ span.

## 2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through correlations which have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNBR thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show reactor core safety limits for a range of THERMAL POWER, REACTOR COOLANT SYSTEM pressure and average temperature which satisfy the following criteria:

A. The average enthalpy at the vessel exit is less than the enthalpy of saturated liquid (far left line segment in each curve).

B. The minimum DNBR satisfies the DNB design criterion (all the other line segments in each curve). The VANTAGE 5 fuel is analyzed using the WRB-2 correlation with design limit DNBR values of 1.24 and 1.23 for the typical cell and thimble cells, respectively. The LOPAR fuel is analyzed using the WRB-1 correlation with design limit DNBR values of 1.23 and 1.22 for the typical and thimble cells, respectively.

C. The hot channel exit quality is not greater than the upper limit of the quality range (including the effect of uncertainties) of the DNB correlations. This is not a limiting criterion for this plant).

## BORATED WATER SOURCE - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 9504 gallons (19% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 99404 gallons (9% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm, and

3) A minimum solution temperature of 44°F (TI-10982).

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

### SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is <72°F, verify the boric acid storage tank solution temperature is ≥ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when it is the source of borated water and the outside air temperature is less than 40°F.

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage Tank with:

1) A minimum contained borated water volume of 36674 gallons (81% of instrument span) (LI-102A, LI-104A),

2) A boron concentration between 7000 ppm and 7700 ppm, and

3) A minimum solution temperature of 65°F (TI-0103).

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 631478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A),

2) A boron concentration between 2400 ppm and 2600 ppm,

3) A minimum solution temperature of 44°F,

4) A maximum solution temperature of 116°F (TI-10982), and

5) RWST Sludge Mixing Pump Isolation Valves capable of closing on RWST low-level.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the Boric Acid Storage Tank inoperable and being used as one of the above required borated water sources, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT for MODE 5 at 200°F; restore the Boric Acid Storage Tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20903) is $\leq$ 72°F, verify the boric acid storage tank solution temperature is $\geq$ 65°F.

b. At least once per 24 hours by verifying the RWST temperature (TI-10982) when the outside air temperature is less than 40°F.

c. At least once per 18 months by verifying that the Sludge Mixing Pump Isolation Valves automatically close upon RWST low-level test signal.

## ROD DROP TIME

### LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg}$ (TI-0412, TI-0422, TI-0432, TI-0442) greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

### SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated (NI-0041B, NI-0042B, NI-0043B, NI-0044B) AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 ABOVE 50 PERCENT RATED THERMAL POWER.*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR,

   1. Either restore the indicated AFD to within the limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux* - High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2) At least once per hour until the AFD Monitor Alarm is updated after restoration to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside its limits.

*See Special Test Exceptions Specification 3.10.2.

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SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Determining the computed heat flux hot channel factor $F_Q^C(Z)$, as follows:

Increase the measured $F_Q(Z)$ obtained from the power distribution map by 3% to account for manufacturing tolerances and further increase the value by 5% to account for measurement uncertainties.

c. Verifying that $F_Q^C(Z)$, obtained in Specification 4.2.2.2b. above, satisfies the relationship in Specification 3.2.2.

d. Satisfying the following relationship:

$$F_Q^C(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)} \text{ for } P > 0.5$$

$$F_Q^C(7) \leq \frac{F_Q^{RTP} \times K(Z)}{0.5 \times W(Z)} \text{ for } P \leq 0.5$$

Where $F_Q^C(Z)$ is obtained in Specification 4.2.2.2b above, $F_Q^{RTP}$ is the $F_Q$ limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and W(Z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation.
$F_Q^{RTP}$, K(Z), and W(Z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.8.1.6.

e. Measuring $F_Q(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

*During power escalation after each fuel loading, power level may be increased until equilibrium conditions at any power level greater than or equal to 50% of RATED THERMAL POWER have been achieved and a power distribution map obtained.

f. With measurements indicating

$$\begin{matrix}\text{maximum} \\ \text{over}\end{matrix} \left( \frac{F_Q^C(Z)}{K(Z)} \right)$$

has increased since the previous determination of $F_Q^C(Z)$ either of the following actions shall be taken:

1) Increase $F_Q^C(Z)$ by 2% and verify that this value satisfies the relationship in Specification 4.2.2.d, or

2) $F_Q^C(Z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\begin{matrix}\text{maximum} \\ \text{over}\end{matrix} \left( \frac{F_Q^C(Z)}{K(Z)} \right) \text{ is not increasing.}$$

g. With the relationships specified in Specification 4.2.2.2d above not being satisfied:

1) Calculate the percent $F_Q(Z)$ exceeds its limits by the following expression:

$$\left\{ \left( \begin{matrix}\text{maximum} \\ \text{over } Z\end{matrix} \left[ \frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{P}} \times K(Z) \right] \right) - 1 \right\} \times 100 \text{ for } P > 0.5$$

$$\left\{ \left( \begin{matrix}\text{maximum} \\ \text{over } Z\end{matrix} \left[ \frac{F_Q^C(Z) \times W(Z)}{\frac{F_Q^{RTP}}{0.5}} \times K(Z) \right] \right) - 1 \right\} \times 100 \text{ for } P \leqq 0.5, \text{ and}$$

2) The following action shall be taken.

Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the CORE OPERATING LIMITS REPORT by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2g.1. Within 8 hours, reset the AFD alarm setpoints to these modified limits.

SURVEILLANCE REQUIREMENTS (Continued)

h. The limits specified in Specification 4.2.2.2c are applicable in all core plane regions, i.e., 0 - 100%, inclusive.

i. The limits specified in Specifications 4.2.2.2d, 4.2.2.2f, and 4.2.2.2g above are not applicable in the following core plane regions:

1) Lower core region from 0 to 15%, inclusive.
2) Upper core region from 85 to 100%, inclusive.

4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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## 3/4.2.5 DNB PARAMETERS

### LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits:

a. Reactor Coolant System $T_{avg}$ (TI-0412, TI-0422, TI-0432, TI-0442), $\leq$ 592.5°F

b. Pressurizer Pressure (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A), $\geq$ 2199 psig*

c. Reactor Coolant System Flow (FI-0414, FI-0415, FI-0416, FI-0424, FI-0425, FI-0426, FI-0434, FI-0435, FI-0436, FI-0444, FI-0445, FI-0446) $\geq$ 393,136 gpm**

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.5.1 Reactor Coolant System $T_{avg}$ and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours. RCS flow rate shall be monitored for degradation at least once per 12 hours. In the event of flow degradation, RCS flow rate shall be determined by precision heat balance within 7 days of detection of flow degradation.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at each fuel loading and at least once per 18 months.

4.2.5.3 After each fuel loading, the RCS flow rate shall be determined by precision heat balance within 7 days after exceeding 90% RATED THERMAL POWER. The RCS flow rate shall also be determined by precision heat balance at least once per 18 months. Within 7 days prior to performing the precision heat balance flow measurement, the instrumentation used for performing the precision heat balance shall be calibrated. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

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*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**Includes a 2.7% flow measurement uncertainty.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 7. Semi-Automatic Switchover to Containment Emergency Sump (Continued) | | | | | |
| b. RWST Level--Low-Low Coincident With Safety Injection (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A) | 3.5 | 0.71 | 1.67 | >275.3 in. from tank base (≥39.1% of instrument span) | >264.9 in. from tank base (≥37.4% of instrument span) |
| 8. Loss of Power to 4.16 kV ESF Bus | | | | | |
| a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage | N.A. | N.A. | N.A. | ≥2975 volts with a < 0.8 second time delay. | ≥2912 volts with a < 0.8 second time delay. |
| b. 4.16 kV ESF Bus Undervoltage-Degraded Voltage | N.A. | N.A. | N.A. | ≥3746 volts with a <20 second time delay. | ≥3683 volts with a <20 second time delay. |
| 9. Engineered Safety Features Actuation System Interlocks | | | | | |
| a. Pressurizer Pressure, P-11 (PI-0455A,B&C, PI-0456 & PI-0456A, PI-0457 & PI-0457A, PI-0458 & PI-0458A) | N.A. | N.A. | N.A. | ≤ 2000 psig | ≤ 2010 psig |
| b. Reactor Trip, P-4 | N.A. | N.A. | N.A. | N.A. | N.A. |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | Z | SENSOR ERROR (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 10. Control Room Emergency Filtration System Actuation | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |
| c. Safety Injection | See Functional Unit 1. above for all Safety Injection Trip Setpoints and Allowable Values. | | | | |
| d. Intake Radiogas monitor (RE-12116, RE-12117) | N.A. | N.A. | N.A. | ≤ 3X background | N.A. |
| 11. Fuel Handling Building Post-Accident Ventilation Actuation (Common System) | | | | | |
| a. Manual Initiation | N.A. | N.A. | N.A. | N.A. | N.A. |
| b. Fuel Handling Building Exhaust Duct Radiation Signal (ARE-2532A&B, ARE-2533A&B) | N.A. | N.A. | N.A. | c | N.A. |
| c. Automatic Actuation Logic And Actuation Relays | N.A. | N.A. | N.A. | N.A. | N.A. |

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS

### 3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 6555 (29.2% of instrument span) and 6909 gallons (70.7% of instrument span) (LI-0950, LI-0951, LI-0952, LI-0953, LI-0954, LI-0955, LI-0956, LI-0957),

c. A boron concentration of between 1900 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 617 and 678 psig. (PI-0960A&B, PI-0961A&B, PI-0962A&B, PI-0963A&B, PI-0964A&B, PI-0965A&B, PI-0966A&B, PI-0967A&B).

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open (HV-8808A, B, C, D).

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume (67 gallons) by verifying the boron concentration of the accumulator solution; and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the isolation valve operator is open.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

## 3/4.5.3 ECCS SUBSYSTEMS - $T_{avg}$ LESS THAN 350°F

### SAFETY INJECTION PUMPS

### LIMITING CONDITION FOR OPERATION

3.5.3.2 All Safety Injection pumps shall be inoperable.

APPLICABILITY MODES 4, 5, and 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

### SURVEILLANCE REQUIREMENTS

4.5.3.2 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3, prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, and at least once per 31 days thereafter.

* An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

## 3/4.5.4 REFUELING WATER STORAGE TANK

### LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 631,478 gallons (86% of instrument span) (LI-0990A&B, LI-0991A&B, LI-0992A, LI-0993A).

b. A boron concentration of between 2400 ppm and 2600 ppm of boron,

c. A minimum solution temperature of 44°F, and

d. A maximum solution temperature of 116°F (TI-10982)

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

APPLICABILITY: MODES 1, 2, 3 and 4.
ACTION:

a. With the RWST inoperable except for the Sludge Mixing Pump Isolation Valves, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the sludge mixing system by either closing the manual isolation valves or deenergizing the OPERABLE solenoid pilot valve within 6 hours and maintain closed.

### SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:
   1) Verifying the contained borated water volume in the tank, and
   2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F.

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

## 3/4.2 POWER DISTRIBUTION LIMITS

## BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) meeting the DNB design criterion during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods; and

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

### 3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the $F_Q$ limit specified in the CORE OPERATING LIMITS REPORT (COLR) times K(Z) is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

AXIAL FLUX DIFFERENCE (Continued)

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the allowed $\Delta I$ power operating space for RAOC operation specified within the COLR and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limit on peak local power density is not exceeded, (2) the DNB design criterion is met, and (3) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod banks are sequenced with a constant tip-to-tip distance between banks as described in Specification 3.1.3.6;

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an $F_Q$ measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor $F_Q(Z)$ is met. W(Z) accounts for the effects of normal operation transients with the AFD band and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(Z) function for normal operation and the AFD band are provided in the CORE OPERATING LIMITS REPORT per Specification 6.8.1.6.

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to meet the DNB design criterion throughout each analyzed transient. The indicated $T_{avg}$ value of 592.5°F and the indicated pressurizer pressure value of 2199 psig correspond to analytical limits of 594.4°F and 2185 psig respectively, with allowance for measurement uncertainty.

3/4.2.5 DNB PARAMETERS (Continued)

The measurement uncertainty for the RCS total flow is based upon performing a precision heat balance flow measurement above 90% RATED THERMAL POWER and using the results to correlate the flow indication channels with the measured flow. If a precision heat balance flow measurement is performed below 90% RATED THERMAL POWER, the effect on the measurement uncertainty shall be taken into account. Potential fouling of the feedwater venturis which might not be detected could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected feedwater venturi fouling is included in the measurement uncertainty. Any fouling which might bias the RCS flow rate measurement greater than 0.1% may be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance flow measurements, i.e., either the effect of the fouling shall be quantified and accounted for in the RCS flow rate measurement or the affected venturis shall be cleaned to eliminate the fouling. The indicated RCS flow value of 393,136 gpm corresponds to an analytical value of 382,800 gpm with allowance for measurement and indication uncertainties.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate degradation on a 12 hour basis. A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.9 or 3.3.3.10, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.8.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.8.1.6 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

a. SHUTDOWN MARGIN LIMIT FOR MODES 1 and 2 for Specification 3/4.1.1.1,

b. SHUTDOWN MARGIN LIMITS FOR MODES 3, 4 and 5 for Specification 3/4.1.1.2,

c. Moderator temperature coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

d. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

e. Control Rod Insertion Limits for Specification 3/4.1.3.6,

f. Axial Flux Difference Limits for Specification 3/4.2.1,

g. Heat Flux Hot Channel Factor, K(Z) and W(Z), for Specification 3/4.2.2,

h. Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously approved by the NRC in:

CORE OPERATING LIMITS REPORT (Continued)

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
(Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).
(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for $F_Q$ Methodology).)

c. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION", February 1982 (W Proprietary).
(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 43 AND 44 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NOS. 23 AND 24 TO FACILITY OPERATING LICENSE NPF-81

GEORGIA POWER COMPANY, ET AL.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

## 1.0 INTRODUCTION

By letter dated November 29, 1990 (Ref. 1), Georgia Power Company, et al. (GPC or the licensee), requested amendments to Facility Operating Licenses NPF-68 and NPF-81 for Vogtle Electric Generating Plant (Vogtle or the facility) Units 1 and 2, respectively. Supplemental information was provided by letters dated January 29, 1991 (Ref. 2), March 6, 1991 (Ref. 3), March 27, 1991 (Ref. 4), March 29, 1991 (Ref. 5), April 19, 1991 (Ref. 6), August 8, 1991 (Ref. 7), and August 19, 1991 (Ref. 8). The August 19, 1991 letter provided clarifying information that did not change the action previously noticed or affect the initial proposed no significant hazards consideration determination. The proposed amendments would revise the Technical Specifications (TSs) to provide for a long-term fuel management strategy and enhanced operational flexibility. This fuel management strategy is based upon the use of reload fuel assemblies of the Westinghouse VANTAGE-5 design and would implement high energy 18-month fuel cycles with high capacity factors, low leakage patterns, and extended fuel burnup. Operating characteristics for the new fuel would be based upon analyses performed by Westinghouse with improved computer codes and methodologies.

Certain TS changes proposed by the licensee in its submittals are outside the scope of this Safety Evaluation.

## 2.0 EVALUATION

### 2.1 VANTAGE-5 Fuel Load

Currently, Vogtle Units 1 and 2 utilize the Westinghouse low-parasitic (LOPAR) fuel design for core reloads. The licensee proposed to use fresh VANTAGE-5 fuel assemblies in each future reload until a full core loading of VANTAGE-5 fuel is achieved. The licensee indicated that transition to full VANTAGE-5 fuel in the core would be completed by the third loading of VANTAGE-5 fuel for each Vogtle unit.

Since Vogtle uses combined Units 1 and 2 TSs, the licensee proposed to accomplish the changes to the TSs in two phases. Phase 1 would include TS changes involving Vogtle Unit 1 Cycle 4 initial loading of VANTAGE-5 fuel which is scheduled to begin in late September 1991. In Phase 1, the Unit 2 would continue to operate with the existing TSs. Phase 2 would include TS changes to reflect initial loading of VANTAGE-5 fuel into Vogtle Unit 2 Cycle 3 which is scheduled to begin in February 1992. Accordingly, the proposed amendments would contain two sets, Phase 1 and Phase 2, of TS change pages to be effective for Unit 1 Cycle 4 and Unit 2 Cycle 3, respectively.

The licensee's submittal includes a safety evaluation for the reload transition from the present LOPAR fueled core to an all VANTAGE-5 fueled core and provides the results of the fuel design and nuclear, thermal-hydraulic, and accident analyses. The NRC staff has reviewed these reports and the results of its review are presented in the following sections of this Safety Evaluation.

2.1.1 Fuel Design

Vogtle Units 1 and 2 are currently operating with Westinghouse 17x17 LOPAR fuel. Beginning with Unit 1 Cycle 4 and Unit 2 Cycle 3, reload fuel will consist of the Westinghouse VANTAGE-5 fuel design (Ref. 9). As a result, the future core loadings would range from LOPAR and VANTAGE-5 transition cores to eventually an all VANTAGE-5 fueled core. WCAP-10444 (Ref. 9) documents the VANTAGE-5 fuel design features, which include smaller diameter fuel rods, mid-span Zircaloy grids, intermediate flow mixer (IFM) grids, natural uranium oxide axial blankets (optional on the first transition core), integral fuel burnable absorbers (IFBAs), extended fuel burnup, and reconstitutable top nozzles (RTN). These design features have been generically approved by the NRC.

The NRC review of WCAP-10444 (Ref. 9) concluded that it was an acceptable reference to support plant-specific application of VANTAGE-5 subject to certain specified conditions. The licensee has addressed these conditions to support the use of VANTAGE-5 fuel assemblies in Vogtle Units 1 and 2, and the NRC staff evaluations of these conditions are included in the appropriate sections of this Safety Evaluation.

The licensee evaluated the fuel rod design for VANTAGE-5 fuel in Units 1 and 2 using NRC approved fuel performance models (Refs. 10 and 11) and the extended burnup design methods, which allow peak rod burnup to 60,000 MWD/MTU (Ref. 12). The extended burnup design methods have also been previously approved by the NRC with the exception of the statistical convolution method for evaluation of initial fuel rod to nozzle growth gap as a result of fuel rod irradiation effects. To determine the initial fuel rod to nozzle growth gap, the licensee used the method of worst-case fabrication tolerances in accordance with Section M of Reference 12 (mechanistic method) instead of the statistical convolution method. This method has been previously approved by the NRC. Therefore, the evaluation of initial fuel rod to nozzle growth gap is acceptable.

To evaluate the structural integrity of the fuel assembly when subjected to seismic and loss-of-coolant accident (LOCA) forces, the licensee used time-history numerical techniques based on the Vogtle plant-specific safe shutdown earthquake (SSE). The results show that neither the fully loaded VANTAGE-5 core nor the transition core would buckle due to combined impact forces of a seismic and LOCA event and that, therefore, the core coolable geometry would be maintained. These results meet the NRC requirements of Appendix A, Standard Review Plan (SRP) Section 4.2 (Ref. 13) for structural integrity.

Based on the above, the NRC staff concludes that the fuel design for the proposed fuel rod types in the Vogtle cores meet the NRC requirements of Appendix A, SRP Section 4.2. Therefore, the proposed fuel design for Vogtle Units 1 and 2 is acceptable.

2.1.2 Nuclear Design

The licensee's analytical methods and core models used to compute nuclear design parameters characterizing the Vogtle transition core are described in References 9, 14, 15, and 16. These methods and models have been used in other Westinghouse reactors and have been previously approved by the NRC for generic use. The licensee's analyses show that the differences in effects of the VANTAGE-5 fuel on the Vogtle physics parameters as compared to the LOPAR fuel are small. The licensee also previously performed a spent fuel pool criticality analysis to allow the storage of VANTAGE-5 fuel assemblies.

The licensee has proposed to revise the TSs to: a) increase the limit for maximum heat flux hot channel factor ($F_Q$), b) increase the limit for full power nuclear enthalpy rise hot channel factor (F delta h), c) implement Relaxed Axial Offset Control (RAOC) beginning with Cycle 4 of Unit 1 and Cycle 3 of Unit 2, and d) relocate Axial Flux Difference (AFD) operating bands for RAOC operation in the Core Operating Limits Report (COLR) instead of in the TSs.

The proposed increase in the maximum heat flux hot channel factor ($F_Q$) limit for LOCA would be from the current value of 2.3 to 2.5. The non-LOCA safety analyses assume a maximum $F_Q$ of 2.55. The full power nuclear enthalpy rise hot channel factor (F delta h) would be increased from the current LOPAR value of 1.55 to 1.57 for LOPAR and to 1.65 for VANTAGE-5 fuel for LOCA calculations. The non-LOCA calculations applicable to the VANTAGE-5 core assume a full power F delta h value of 1.70. The licensee performed an evaluation of the increased limits on peaking factors to demonstrate that the design limits on peak local power density and minimum departure from nucleate boiling ratio (DNBR) are not exceeded during normal operation and anticipated operational occurrences (AOOs), and that in the event of a LOCA, the peak clad temperature will not exceed the 2200°F emergency core cooling system (ECCS) acceptance criteria specified in 10 CFR 50.46. The NRC staff's review and acceptance of these changes is discussed in Section 2.1.4 of this Safety Evaluation.

For control of axial power distribution, the licensee proposed to use the RAOC method to define the allowed operating space as depicted on a graph of AFD versus thermal power (Ref. 15). This methodology allows direct surveillance of $F_Q$ rather than the radial component of the total peaking factor $F_{xy}$, and therefore, provides a more direct method of demonstrating compliance with the Limiting Condition for Operation. The models for demonstrating the RAOC method of control has been previously approved by the NRC (Ref. 10).

Relocation of the AFD band to the COLR has been approved by the NRC staff in previous TS Amendments 32 (Unit 1) and 12 (Unit 2) dated July 30, 1990.

Based on its review, the NRC staff finds that the licensee's evaluation of the nuclear design for the Vogtle reactors is based on the NRC approved methodology. Therefore, the proposed nuclear design is acceptable.

2.1.3 Thermal-Hydraulic Design

The licensee's DNB methodology for the transition core is in accordance with References 17 and 18 which have been previously approved by the NRC (Ref. 19). Using this methodology, the licensee analyzed the transition core as if the entire core consisted of one assembly type, i.e., full LOPAR or full VANTAGE-5. The resultant DNBR values were then reduced by application of appropriate transition core penalty factors to account for localized flow redistribution in a mixed core from the VANTAGE-5 assembly into the LOPAR assembly, and for fuel rod bow penalty. To obtain DNB uncertainty factors, the licensee used the Revised Thermal Design Procedure (RTDP) methodology (Ref. 20) for a statistical combination of variations in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and DNB correlation predictions. Based on the DNB uncertainty factors, RTDP design limit DNBR values were determined such that the DNB design criterion is met, i.e, there remains at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod during normal operation, operational transients, and during AOOs. The licensee's safety analyses used input parameters at their nominal values since the parameter uncertainties were considered in determining the RTDP design limit DNBR values.

The licensee's DNBR analyses of the VANTAGE-5 and LOPAR fuel incorporated WRB-1 DNB correlation for the 17x17 LOPAR fuel assemblies (Ref. 21) and WRB-2 DNB correlation for the VANTAGE-5 fuel (Ref. 9). Both the WRB-1 and WRB-2 DNB correlations have a correlation limit of 1.17 and are applicable in the heated rod spans above the first mixing vane grid. For calculating DNBR values in the heated region below the first mixing vane grid, the licensee's analyses incorporated W-3 DNB correlation which does not take credit for mixing vane grids. A W-3 DNB correlation limit of 1.30 was used for system pressures in the range of 500 to 1000 psia. For system pressures greater than 1000 psia, a W-3 DNB correlation limit of 1.45 was used (Ref. 22). The improved THINC-IV PWR design modeling method (Ref. 23) was used by the licensee for the DNBR analyses of the VANTAGE-5 and LOPAR fuel.

Reference 24 shows that the rod bow penalty for the 17x17 LOPAR fuel and for the lower assembly spans in the VANTAGE-5 assemblies is less than 1.5%. Reference 25 contains a calculation of the transition core penalty using NRC approved methodology. This transition core penalty accounted for localized flow distribution in a mixed core consisting of VANTAGE-5 and LOPAR fuel assemblies. The DNBR penalty, which is a function of the number of VANTAGE-5 fuel assemblies in the core, is less than 15% for Unit 1 Cycle 4 and Unit 2 Cycle 3 operation.

The licensee's analyses show that the design limit DNBR values for 17x17 LOPAR fuel are 1.23 for typical cells and 1.22 for thimble cells. For VANTAGE-5 fuel, the design limit DNBR values are 1.24 for typical cells and 1.23 for thimble cells. The minimum VANTAGE-5 safety analyses DNBR limits are 1.51 for typical cells and 1.49 for thimble cells.

Based on its review of the thermal-hydraulic evaluation of Vogtle Units 1 and 2 with VANTAGE-5 fuel, the NRC staff finds that the 17x17 LOPAR and VANTAGE-5 fuel assemblies are hydraulically compatible. Approved methodology was used and all thermal-hydraulic design criteria were satisfied. Sufficient DNBR margin exists in the safety limit DNBR to cover rod bow and transition core penalties. Therefore, the staff concludes that the thermal-hydraulic design of the Units 1 and 2 transition cores, as well as the all VANTAGE-5 cores, are acceptable. Maintenance of adequate DNBR margin to cover DNBR penalties will be confirmed by the licensee on a cycle-specific basis during the reload safety evaluation process.

### 2.1.4 Transient and Accident Analyses

To determine the impact of the transition from Westinghouse LOPAR fuel to Westinghouse VANTAGE-5 fuel on the plant safety analyses, the licensee reviewed the changes in the design parameters on a cycle-by-cycle basis due to the transition and on a full VANTAGE-5 core basis. These changes were compared to the design parameters included in the existing plant safety analyses for the current core configuration. The licensee reanalyzed those transient events which are sensitive to the identified parameter changes. Reanalyses were also performed when the identified parameter changes were found to be significantly different. In cases where the parameter changes were insignificant or where the transient events are insensitive to the identified parameters, the licensee evaluated the magnitude of the effect and substantiated the validity of the existing plant safety analyses.

The licensee's safety analyses assumed a nominal core thermal power of 3565 MWt, thermal design flow of 374,400 gpm, 10% uniform steam generator tube plugging in each steam generator and increased core bypass flow due to optional thimble plug removal. These assumed values in the licensee's analyses conservatively bound the current rated core thermal power level of 3411 MWT and the licensing thermal design flow of 382,800 gpm. Although the licensee used increased power peaking factors in the safety analyses, the licensee did not propose TS changes for power increases. Moreover, these amendments do not authorize any increase in power nor any increase in the steam generator tube plugging limits.

To define the initial conditions for those transients which have DNB as a limiting criterion and are initiated at or near full power conditions, the licensee implemented the RTDP methodology. The licensee's safety analyses included temperature, pressure, power, and flow uncertainties which are required to be considered when using the RTDP methods. The calculated uncertainties were based on plant procedures for instrument calibration, heat balance calculations, and RCS flow measurement. For conditions where the RTDP methodology is not applicable, the licensee's analyses incorporated the Standard Thermal Design Procedure (STDP). In the STDP method, the licensee's DNBR analyses incorporated conservative input parameters to give the lowest minimum DNBR including DNBR margin to cover DNBR penalties.

The NRC staff has reviewed each of the accidents for which the licensee has performed reanalysis or reevaluation. The results, which include transition core effects, show changes in the consequences of transients and accidents previously analyzed. However, the results remain within the required acceptance criteria. Specifically, for non-LOCA events, during normal operation and AOOs, there is at least a 95% probability at a 95% confidence level (95/95 probability/confidence) that DNB will not occur on the limiting fuel rod. During these operational modes, there is also a 95/95 probability/confidence that the peak kw/ft fuel rods will not exceed the melting temperature of $UO_2$, taken as 4900°F (unirradiated) and 4800°F at end of life. For these events, peak RCS pressure does not exceed 110% of the 2250 psia design pressure. Peak RCS pressure does not exceed 120% of the design pressure during accidents. The maximum average fuel pellet enthalpy remains less than 200 cal/gm for all control rod ejection events, which meets the NRC criterion of less than 280 cal/gm.

The methodologies used in the reanalyses have been previously found acceptable by the NRC. However, the licensee proposed to change the TS Bases to reflect the use of leak-before-break methodology which would allow for control rod insertion when verifying that the reactor will remain subcritical during the long term cooling phase following a large break LOCA. Specifically, the licensee proposed to change TS Bases 3/4.5.4 to state that "... 3) the reactor will remain subcritical in the cold condition following a large break LOCA assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump, post-LOCA with all control rods assumed to be out (Unit 2) and all control rods inserted except for the two most reactive control assemblies (Unit 1)." The NRC does not currently support the use of the leak-before-break methodology with respect to either short-term or long-term effects of the FSAR Chapter 15 events. Accordingly, this proposed change to the TS Bases 3/4.5.4 is not acceptable. Thus, RWST boron concentration should be relied upon to maintain subcriticality at cold conditions during post LOCA conditions, rather than control rod insertion.

The NRC staff reviewed the radiological consequences of the accidents reanalyzed by the licensee because of the increase in fuel burnup to 60,000 MWD/MTU associated with the use of VANTAGE-5 fuel and other proposed changes. We find that these accidents remain bounded by the current analyses except for the fuel handling accident and the steam generator tube rupture accidents. The fuel handling accident and the steam generator tube rupture accident were reanalyzed by the licensee. The reanalyses showed that the doses were less than the NRC acceptance limits, namely 75 rem to the thyroid and 6 rem to the whole body for the fuel handling accident and 30 rem to the thyroid and 2.5 rem to the whole body for the steam generator tube rupture accident.

To develop specific peaking factor limits, the licensee performed a large break LOCA analysis for Vogtle Units 1 and 2 applicable to a full core of VANTAGE-5 fuel assemblies. The analyses were performed using the approved Westinghouse 1981 ECCS evaluation model (Refs. 27 and 28) for the spectrum of cold leg breaks. The analysis assumed operation of both maximum containment engineered safety features and minimum ECCS equipment (the loss of one complete train of ECCS components). This combination results in the lowest containment pressure and minimum delivered flow available to the RCS. The analyses applied a transition core penalty of 50°F to the peak clad temperature (PCT) of the VANTAGE-5 assembly, a 10°F increase due to containment mini-purge isolation, and a 11°F increase due to the RCS operating average temperature uncertainty band. The worst case PCT is 2108°F. Therefore, the results demonstrated that the PCT acceptance criterion of 2200°F is met. The results also demonstrate that the criteria of 10 CFR 50.46 related to clad oxidation and maximum hydrogen generation continue to be met. In addition, the core would remain amenable to cooling during and after the LOCA and would be maintained in a shutdown condition with borated water and without credit for control rod insertion.

The licensee's small break (less than one square foot) LOCA analysis used the NRC approved Westinghouse ECCS small break evaluation model and the NOTRUMP code (Ref. 29). The analysis assumed a full core of VANTAGE-5 fuel to determine PCT for a spectrum of cold leg breaks. The most limiting single failure was loss of an emergency power train, which results in loss of one complete train of ECCS components. The minimum delivered flow available to the RCS was based on this single failure. The results demonstrate that one centrifugal pump and one high head safety injection pump, together with the accumulators, provide sufficient core flooding to meet the acceptance criteria of 10 CFR 50.46.

## 3.0 TECHNICAL SPECIFICATION CHANGES

This section describes the NRC staff's safety evaluation of each of the specific TS changes proposed by the licensee for Vogtle Units 1 and 2. As described in Section 2.1 of this Safety Evaluation, the proposed changes to the TSs would be accomplished in two phases. The bases for the staff's acceptability of the proposed TS changes are either provided by evaluation in this section or by appropriate references to Section 2 of this Safety Evaluation.

### 3.1 Changes Associated with New Fuel and Methodologies

#### 3.1.1 Reactor Cores Safety Limits, Reactor Trip System Instrumentation Setpoints, DNB Parameters

(a) The Reactor Core Safety Limits of TS Figure 2.1-1 would be revised to reflect the use of the revised DNB methods.

This is acceptable as discussed in Section 2.1.3 of this Safety Evaluation since the licensee's DNB analyses are based on approved methodologies and provided acceptable results.

(b) In TS Table 2.2-1, "Reactor Trip System Instrumentation Setpoints," Overtemperature Delta T and Overpower Delta T reactor trip system instrumentation Setpoints would be revised. Specifically, changes would be made to the total allowance, Z value, sensor error, $K_1$, $K_2$, $K_3$, $K_4$, $K_6$, nominal $T_{avg}$, and the $f_1$ (Delta-I) function.

The implementation of VANTAGE-5 fuel and the inclusion of more conservative uncertainty values in temperature and pressure cause the DNB core limits to change. These core limit changes and implementation of RAOC result in Overtemperature Delta T and Overpower Delta T reactor trip setpoint changes. These setpoint changes are reflected in the Vogtle safety analyses, which resulted in acceptable consequences as discussed in Section 2.1.4 of this Safety Evaluation. These changes are, therefore, acceptable.

(c) TS Bases 2.1.1, 3/4.2, 3/4.2.2, 3/4.2.3, 3/4.2.5, and 3/4.4.1 would be revised to reflect the use of the revised methods and correlations. TS 3/4.2.5 would add that the measurement uncertainty of RCS flow includes an allowance of 0.1 % to account for feedwater venturi fouling.

The DNB-related parameters (RCS $T_{avg}$, pressurizer pressure, and RCS flow) and flow measurement uncertainty specified in TS 3.2.5 would be modified.

The DNB parameters would be modified due to the revised instrument uncertainties reported in References 25 and 26 and the revised DNB methods and correlations. The revised RCS average temperature uncertainty allowance is ±6°F as compared to the previous ±4°F. The revised pressurizer pressure uncertainty is ±50 psi as compared to the previous ±30 psi. The flow measurement uncertainty was reduced to ±2.7% from ±3.5% and includes a 0.1% allowance to account for feedwater venturi fouling (Ref. 7). The licensee indicated in their August 8, 1991 letter that it plans to use the main control board flow indicators for reading RCS flow and the flow measurement uncertainty of 2.7% accounts for uncertainties involved in reading RCS flow using the main control board instead of the plant process computers. Consistent with the uncertainty values, the indicated RCS average temperature value is increased to 592.5°F from 591°F, the pressurizer pressure is decreased to 2199 psig from 2224 psig. By letter dated August 19, 1991, the licensee revised the RCS flow limit to 393,136 gpm which is the actual value corresponding to 2.7% flow measurement uncertainty. These uncertainty values were used by the licensee in the RTDP and they conservatively bound the values derived in the licensee's analyses.

Based on the review of these DNB parameters, the NRC staff finds that the DNB parameters were determined by the licensee using methodologies which have been generically approved by the NRC. As discussed in Section 2.1.4 of this Safety Evaluation, the revised limits for the reanalyzed transient events are consistent with the original plant safety analyses assumptions and the DNBR design criterion continues to be met. Therefore, the proposed changes are acceptable.

### 3.1.2 Axial Flux Difference and Peaking Factor Surveillance

(a) The AFD specified in TS 3.2.1 would be modified to reflect use of RAOC for the control of axial power distribution.

Currently, Vogtle Units 1 and 2 use the constant axial offset control (CAOC) procedure for the control of axial power distribution. The RAOC procedure which defines the allowed operating space of AFD versus thermal power has been approved by the NRC as discussed in Section 2.1.2 of this Safety Evaluation. The proposed TS Sections are consistent with the existing Westinghouse Standard TS and, therefore, the changes are acceptable.

(b) The surveillance requirements of TS 3/4.2.2 would be revised to reflect appropriate $F_Q$ surveillance.

As previously stated, the procedure for controlling axial power distribution will be RAOC. The proposed changes are appropriate since RAOC allows direct surveillance of the heat flux hot channel factor $F_Q$. The changes are also consistent with the Westinghouse Standard TS. Therefore, the proposed changes are acceptable.

(c) In TS 3.2.2 Action Statement A, the phrase "Overpower Delta-T Trip Setpoints have been reduced at least 1%" would be revised to read "Overpower Delta-T Trip Setpoints (Value of $K_4$) have been reduced at least 1% (in Delta-T span)."

This is an administrative change to indicate the component of the "Overpower Delta-T Trip Setpoints" which have been reduced by 1%. This action statement is consistent with the TS Table 2.2-1 "Reactor Trip System Instrumentation Setpoints." Therefore, the change is acceptable.

(d) TS 6.8.1.6 would be revised to reflect the use of RAOC methodology by substituting reference to WCAP-10216-P-A in lieu of the present reference for axial offset control.

This Section of the TS identifies analytical methods, previously approved by the NRC, that are used to determine core operating limits specified in the COLR. Since RAOC is proposed to be used, it is appropriate that the NRC approved report which describes the RAOC methodology (WCAP-10216-P-A) be referenced and that reference to the previous methodology be deleted. This change is, therefore, acceptable.

3.1.3 Increase in Shutdown and Control Rod Drop Time

TS 3/4.1.3.4 would be revised to reflect an allowable shutdown and control rod drop time of 2.7 seconds instead of the present 2.2 seconds.

This increase in the control rod drop time on the Vogtle safety analyses has been considered in the licensee analyses and the results are shown to be acceptable. The actual drop times will be verified to be less than the values used in the analysis through measurement prior to reactor criticality. Therefore, the change is acceptable.

3.2 Widened Accumulator Water Level Range

TS 3.5.1 would be revised to widen the range of the water volume within the ECCS accumulators. The range would be changed from a minimum of 36% span (6616 gallons) to a minimum of 29.2% span (6555 gallons) and from a maximum of 64% span (6854 gallons) to a maximum of 70.7% span (6909 gallons).

The large break LOCA, which accounts for variations in accumulator water from the nominal level, were reanalyzed by the licensee. The results confirm that acceptable peak clad temperatures and other ECCS acceptance criteria continue to be met with the modified accumulator water level range. The analyses also show that the variation in accumulator water volume would have an insignificant effect on the containment sump level and boron concentration. As discussed in Section 2.1.4 of this Safety Evaluation, we find the analyses and its results acceptable. The proposed change meets 10 CFR 50.46 and is, therefore, acceptable.

3.3 P-11 Setpoint

TS Table 3.3-3 presently specifies a P-11 setpoint of 1970 psig and an allowable value of 1980 psig. This P-11 setpoint would be changed to 2000 psig with an allowable value of 2010 psig.

The effect of this proposed change would be to increase the band between the pressure for which safety injection (SI) is allowed to be blocked (P-11 setpoint) and the setpoint for SI actuation on low pressurizer pressure (1870 psig).

During planned depressurizations, the SI signal is blocked in order to prevent an inadvertent SI actuation. The P-11 interlock assures that the block of the SI signal is removed when pressure is returned to the normal operating range by defeating the SI block when pressurizer pressure is above the P-11 setpoint.

The proposed 30 psi increase in the difference between the P-11 setpoint and the SI actuation setpoint will reduce the probability of an inadvertent SI signal during planned depressurization. The revised band is sufficient for this purpose. The setpoint for the SI signal remains unaffected. None of the safety analyses in the Vogtle FSAR use the P-11 setpoint. There are no effects on the safety analyses as a result of this change. Therefore, the proposed change is acceptable.

3.4 RCS Flow Measurement and Its Associated Uncertainty

(a) TS 4.2.5.3 would be revised to replace the phrase "prior to 75% RTP" with the phrase "within 7 days after exceeding 90% RTP (Unit 1) or prior to operation above 75% RTP (Unit 2)."

(b) In TS 3.2.5, the flow uncertainty in the footnote would be changed from "3.5%" to "2.7% (Unit 1) or 3.5% (Unit 2)." The associated flow would be changed from "396,198 gpm" to "393,136 gpm (Unit 1) or 396,198 gpm (Unit 2)."

The above described changes would become effective with the initial use of VANTAGE-5 fuel on Vogtle Unit 1 Cycle 4. With the initial use of VANTAGE-5 fuel on Unit 2 Cycle 3, the "(Unit 1) or prior to operation above 75% RTP (Unit 2)" in TS 4.2.5.3 and "(Unit 1) or 396,198 gpm (Unit 2)" in TS 3.2.5 would be deleted.

(c) TS Bases 3/4.2.5 would be supplemented to describe the bases for the uncertainty used for the measurement of RCS flow.

The licensee has implemented RTDP (described in the WCAPs 12460 and 12462) to determine uncertainty in RCS flow measurement. The RTDP methodology has been previously approved by the NRC. These WCAPs, which were submitted to the NRC by letter dated March 6, 1991, had indicated that the RCS flow measurement uncertainty was based on performing precision heat balance measurement above 70% RTP. However, by letter dated August 8, 1991, the licensee provided errata sheets to these WCAPs indicating that the flow measurement uncertainty was based on precision heat balance measurement above 90% RTP instead of 70% RTP.

The NRC staff finds that conducting a precision heat balance measurement above 90% RTP is consistent with the licensee's analyses and represents common industry practices. The change is, therefore, acceptable.

3.5 <u>RWST Minimum Temperature Change</u>

(a) TS 3/4.1.2.5, 3/4.1.2.6, and 3/4.5.4 would be revised to reduce the RWST minimum solution temperature from 54°F to 44°F. The associated surveillance limit, which establishes a minimum outside air temperature at which the tank solution temperature must be verified daily, would be reduced from 50°F to 40°F.

The inadvertent operation of the ECCS during power operation, small break LOCA, and steam generator tube rupture accidents are sensitive to minimum RWST solution temperature. These events have been reanalyzed by the licensee as part of the VANTAGE-5 fuel transition using the NRC approved methodology. As discussed in Section 2.2 of this Safety Evaluation, the analyses demonstrate that the consequences remain within the staff's acceptance criteria. The temperature reduction and the proposed TS changes are, therefore, acceptable.

(b) TS 3/4.5.4 would be revised to delete "break flow equal to or greater than 3.0 square feet."

This revision would clarify the erroneous implication of this phrase that a large break LOCA is defined as being greater than 3.0 square feet. This is an editorial change affecting the text only and does not affect the associated analyses. It is, therefore, acceptable.

(c) TS Bases 3/4.5.4 Refueling Water Storage Tank which states that "The limits on RWST minimum volume and boron concentration ensure that ... 3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow equal to or greater than 3.0 square feet) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump, post-LOCA with all control rods assumed to be out."

As discussed in Section 2.1.4 of this Safety Evaluation, the NRC staff does not currently support the use of the leak-before-break methodology with respect to either short-term or long-term effects of FSAR Chapter 15 events. Accordingly, this proposed change to the TS Bases 3/4.5.4 is not acceptable and is not included in the amendments. Thus, RWST boron concentration should be relied upon to maintain subcriticality at cold conditions during post LOCA conditions without credit for control rod insertion.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments. The State official had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amdendments 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 findings (56 FR 20037, 56 FR 24211, and 56 FR 41147). 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.

Principal Contributors: L. Kopp, SRXB
F. Orr, SRXB
L. Raghavan, PDII-3

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