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## 2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

with two recirculation loop operation and shall not be less than 1.08 with single recirculation loop operation

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

with two recirculation loop operation or less than 1.08 with single recirculation loop operation

With MCPR less than 1.07 and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure above 1325 psig, as measured in the reactor vessel steam dome, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

## TABLE 2.2.1-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1. Intermediate Range Monitor, Neutron Flux-High | ≤ 120/125 divisions of full scale | ≤ 122/125 divisions of full scale |
| 2. Average Power Range Monitor: | | |
| a. Neutron Flux-High, Setdown | ≤ 15% of RATED THERMAL POWER | ≤ 20% of RATED THERMAL POWER |
| ~~b. Flow Biased Simulated Thermal Power-High~~ | | |
| Insert "A" ~~1) Flow Biased~~ | ~~≤ 0.66 W+48%, with a maximum of~~ | ~~≤ 0.66 W+51%, with a maximum of~~ |
| ~~2) High Flow Clamped~~ | ~~≤ 111.0% of RATED THERMAL POWER~~ | ~~≤ 113.0% of RATED THERMAL POWER~~ |
| c. Neutron Flux-High | ≤ 118% of RATED THERMAL POWER | ≤ 120% of RATED THERMAL POWER |
| d. Inoperative | NA | NA |
| 3. Reactor Vessel Steam Dome Pressure - High | ≤ 1064.7 psig | ≤ 1079.7 psig |
| 4. Reactor Vessel Water Level - Low, Level 3 | ≥ 9.7 inches above instrument zero* | ≥ 8.7 inches above instrument zero |
| 5. Reactor Vessel Water Level-High, Level 8 | ≤ 51.0 inches above instrument zero* | ≤ 52.1 inches above instrument zero |
| 6. Main Steam Line Isolation Valve - Closure | ≤ 8% closed | ≤ 12% closed |
| 7. Main Steam Line Radiation - High | ≤ 3.0 x full power background | ≤ 3.6 x full power background |
| 8. Drywell Pressure - High | ≤ 1.68 psig | ≤ 1.88 psig |
| 9. Scram Discharge Volume Water Level - High | | |
| a. Level Transmitter - LISN601A and B | ≤ 49" | ≤ 52" |
| LISN601C and D | ≤ 49" | ≤ 51.7" |
| b. Float Switches - LSN013A and B | ≤ 48.76" | ≤ 53.50" |
| LSN013C and D | ≤ 46.88" | ≤ 49.00" |

*See [illegible] Figure B 3/4 3-1.

Insert "A"

b. Flow Biased Simulated Thermal Power-High

1) Two Recirculation Loop Operation

| | | |
|---|---|---|
| a) Flow Biased | $\leq$ 0.66 W+48%, with a maximum of | $\leq$ 0.66 W+51%, with a maximum of |
| b) High Flow Clamped | $\leq$ 111.0% of RATED THERMAL POWER | $\leq$ 113.0% of RATED THERMAL POWER |

2) Single Recirculation Loop Operation

| | | |
|---|---|---|
| a) Flow Biased | $<$ 0.66 W+42.7%, with a maximum of | $<$ 0.66 W+45.7%, with a maximum of |
| b) High Flow Clamped | $<$ 111.0% of RATED THERMAL POWER | $<$ 113.0% of RATED THERMAL POWER |

## 2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back Insert B approach is used to establish a Safety Limit such that the MCPR is not less than 1.07. MCPR greater than 1.07 represents a conservative margin relative Insert B to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

## 2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GE Critical Power correlation (Reference 1) is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28,000 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28,000 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

Insert "B"

for two recirculation loop operation and 1.08 for single recirculation loop operation.

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## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

#### LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 or 3.2.1-6, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

The limits of figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, and 3.2.1-6 shall be reduced to a value of 0.84 times the two recirculation loop operation limit when in single loop operation.

#### SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 and 3.2.1-6:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

POWER DISTRIBUTION LIMITS

## 3/4.2.2 APRM SETPOINTS

### LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-high scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint ($S_{RB}$) shall be established according to the following relationships:

Insert C →

| ~~TRIP SETPOINT~~ | ~~ALLOWABLE VALUE~~ |
|---|---|
| ~~$S \leq (0.66W + 48\%)T$~~ | ~~$S \leq (0.66W + 51\%)T$~~ |
| ~~$S_{RB} \leq (0.66W + 42\%)T$~~ | ~~$S_{RB} \leq (0.66W + 45\%)T$~~ |

where: S and $S_{RB}$ are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 84.5 million lbs/hr.
T = The ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD). T is applied only if less than or equal to 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-high scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or $S_{RB}$, as above determined, initiate corrective action within 15 minutes and adjust S and/or $S_{RB}$ to be consistent with the Trip Setpoint value * within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-high scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with CMFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

*With CMFLPD greater than the FRTP, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

Insert "C"

a. Two Recirculation Loop Operation

| Trip Setpoint | Allowable Value |
|---|---|
| $S \leq (0.66W + 48\%)T$ | $S \leq (0.66W + 51\%)T$ |
| $S_{RB} \leq (0.66W + 42\%)T$ | $S_{RB} \leq (0.66W + 45\%)T$ |

b. Single Recirculation Loop Operation

| Trip Setpoint | Allowable Value |
|---|---|
| $S \leq (0.66W + 42.7\%)T$ | $S \leq (0.66W + 45.7\%)T$ |
| $S_{RB} \leq (0.66W + 36.7\%)T$ | $S_{RB} \leq (0.66W + 39.7\%)T$ |

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. ROD PATTERN CONTROL SYSTEM | | |
| a. Low Power Setpoint | 27.5 ± 3% of RATED THERMAL POWER | 27.5 ± 7.5% of RATED THERMAL POWER |
| b. High Power Setpoint | 62.5 ± 3% of RATED THERMAL POWER | 62.5 ± 7.5% of RATED THERMAL POWER |
| 2. APRM | | |
| ~~a. Flow Biased Neutron Flux - Upscale~~ Insert D → | ~~≤ 0.66 W + 42%*~~ | ~~≤ 0.66 W + 45%*~~ |
| b. Inoperative | NA | NA |
| c. Downscale | ≥5% of RATED THERMAL POWER | ≥ 3% of RATED THERMAL POWER |
| d. Neutron Flux - Upscale Startup | ≤ 12% of RATED THERMAL POWER | ≤ 14% of RATED THERMAL POWER |
| 3. SOURCE RANGE MONITORS | | |
| a. Detector not full in | NA | NA |
| b. Upscale | ≤ $1 \times 10^5$ cps | ≤ $1.6 \times 10^5$ cps |
| c. Inoperative | NA | NA |
| d. Downscale | ≥ 0.7 cps | ≥ 0.5 cps** |
| 4. INTERMEDIATE RANGE MONITORS | | |
| a. Detector not full in | NA | NA |
| b. Upscale | ≤ 108/125 division of full scale | ≤ 110/125 division of full scale |
| c. Inoperative | NA | NA |
| d. Downscale | ≥ 5/125 division of full scale | ≥ 3/125 division of full scale |
| 5. SCRAM DISCHARGE VOLUME | | |
| a. Water Level-High - LISN602A | ≤ 18.00" | ≤ 21.12" |
| LISN602B | ≤ 18.00" | ≤ 21.60" |
| 6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW | | |
| a. Upscale | ≤ 108% of rated flow | ≤ 111% of rated flow |

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

**Provided signal to noise ratio is ≥ 2, otherwise setpoint of 3 cps and allowable 1.8 cps.

Insert "D"

a. Flow Biased Neutron Flux Upscale

| | | |
|---|---|---|
| 1) Two Recirculation Loop Operation | $\leq 0.66W + 42\%$* | $\leq 0.66W + 45\%$* |
| 2) Single Recirculation Loop Operation | $\leq 0.66W + 36.7\%$* | $\leq 0.66W + 39.7\%$* |

## 3/4.4 REACTOR COOLANT SYSTEM

### 3/4.4.1 RECIRCULATION SYSTEM

### RECIRCULATION LOOPS

### LIMITING CONDITION FOR OPERATION

~~3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:~~

~~a. Total core flow greater than or equal to 45% of rated core flow, or~~

~~b. THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1.~~

Replace with Insert E

~~APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.~~

~~ACTION~~

~~a. With one reactor coolant system recirculation loop not in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours.~~

~~b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.~~

~~c. With two reactor coolant system recirculation loops in operation and total core flow less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:~~

~~1. Determine the APRM and LPRM** noise levels (Surveillance 4.4.1.1.2):~~

~~a) At least once per 8 hours, and~~

~~b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.~~

~~2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.~~

---

~~*See Special Test Exception 3.10.4.~~

~~**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.~~

Insert "E"

3.4.1.1 The reactor coolant system recirculation loops shall be in operation and in Region I as specified in Figure 3.4.1.1-1 with either:

a. Two recirculation loops operating with limits and setpoints per Specifications 2.1.2, 2.2.1, 3.2.1, 3.2.2, 3.3.6, or

b. A single loop operating with:

   1. Volumetric recirculation loop flow rate less than or equal to 33,000 gpm, and

   2. The recirculation loop flow control system in the loop Manual (Position Control) Mode, and

   3. THERMAL POWER less than or equal to 70% of RATED THERMAL POWER

   4. Limits and setpoints for single recirculation loop operation per Specifications 2.1.2, 2.2.1, 3.2.1, 3.2.2, and 3.3.6, and

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*

ACTION

a. During single loop operation, with volumetric recirculation loop flow rate greater than 33,000 gpm, immediately initiate corrective action to reduce flow to less than or equal to 33,000 gpm within 1 hour.

b. During single loop operation, with the recirculation flow control system not in the Loop Manual mode, immediately initiate corrective action to place the recirculation flow control system in the Loop Manual mode within 1 hour.

c. During single loop operation, with THERMAL POWER greater than 70% of RATED THERMAL POWER, immediately initiate corrective action to reduce THERMAL POWER to less than or equal to 70% of RATED THERMAL POWER within 1 hour.

---

* See Special Exception 3.10.4

d. Within 4 hours upon entry into single loop operation, verify that the operating limits in Specification 3.2.1 have been appropriately adjusted for single loop operation.

e. Within 12 hours upon entry into single loop operation, verify that the setpoints in Specifications 2.2.1, 3.2.2 and 3.2.6 are within appropriate limits.

f. During single loop operation with either THERMAL POWER $\leq$ 30% of RATED THERMAL POWER or recirculation loop flow in the operating loop is $\leq$ 50% of rated recirculation loop flow and temperature differences exceeding the limits in Surveillance Requirement 4.4.1.1.4, suspend THERMAL POWER or recirculation loop flow increases. *

g. With one or two reactor coolant system recirculation loops in operation and total core flow greater than 39% and less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Region II of Figure 3.4.1.1-1:

   1. Determine the APRM and LPRM** noise levels (Surveillance 4.4.1.1.2):

      a) At least once per 8 hours, and

      b) Within 30 minutes after completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

   2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels within the required limits within 2 hours by increasing core flow to greater than or equal to 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Region II of Figure 3.4.1.1-1.

h. With one or two reactor coolant system recirculation loops in operation and total core flow less than 39% of rated core flow and THERMAL POWER greater than the limit specified in Region III of Figure 3.4.1.1-1, immediately within 15 minutes initiate corrective action to increase core flow to greater than or equal to 39% of rated core flow or reduce THERMAL POWER to less than the limit specified in Region III of Figure 3.4.1.1-1 within 4 hours.

---

* With one recirculation loop not in operation and isolated, the differential temperature requirements of Surveillance Requirement 4.4.1.1.4b and c are not applicable, and the provisions of Specification 3.0.4 are not applicable with respect to Surveillance Requirement 4.4.1.1.4b and c.

** Detector levels A and C of one LPRM string in the center of the core should be monitored.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

~~4.4.1.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by~~:

~~a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic control unit, and~~

~~b. Verifying that the average rate of control valve movement is:~~

~~1. Less than or equal to 11% of stroke per second opening, and~~

~~2. Less than or equal to 11% of stroke per second closing~~.

~~4.4.1.1.2 Establish a baseline APRM and LPRM* neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage~~.

Replace with
Insert F

~~*Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored~~.

Insert "F"

4.4.1.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic control unit, and

b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing

4.4.1.1.2 Establish a baseline APRM and LPRM* neutron flux noise valve within the regions for which monitoring is required (Specification 3.4.1.1 ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

4.4.1.1.3 Initially, within 1 hour upon entry into single loop operation and once per 12 hours thereafter, verify that:

a. THERMAL POWER is less than or equal to 70% of RATED THERMAL POWER, and

b. The recirculation flow control system is in the Loop Manual (Position Control) mode, and

c. The volumetric recirculation flow rate is less than or equal to 33,000 gpm.

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, and either THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is less than or equal to 50% of rated recirculation loop flow, within 15 minutes prior to an increase in THERMAL POWER or recirculation loop flow, verify that the following differential temperature requirements are met:

---

* Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of core should be monitored.

a. $\leq 100^{\circ}F$ between reactor vessel steam space coolant and bottom head drain line coolant, and

b. $\leq 50^{\circ}F$ between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel**, and

c. $\leq 50^{\circ}F$ between the reactor coolant within the loop not in operation and the operating loop.**

---

** With one recirculation loop not in operation and isolated, the differential temperature requirements of Surveillance Requirement 4.4.1.1.4b and c are not applicable and the provision of Surveillance Requirement 4.0.4 are not applicable with respect to Surveillance Requirement 4.4.1.1.4b and c.

Replace with new Figure
Insert H

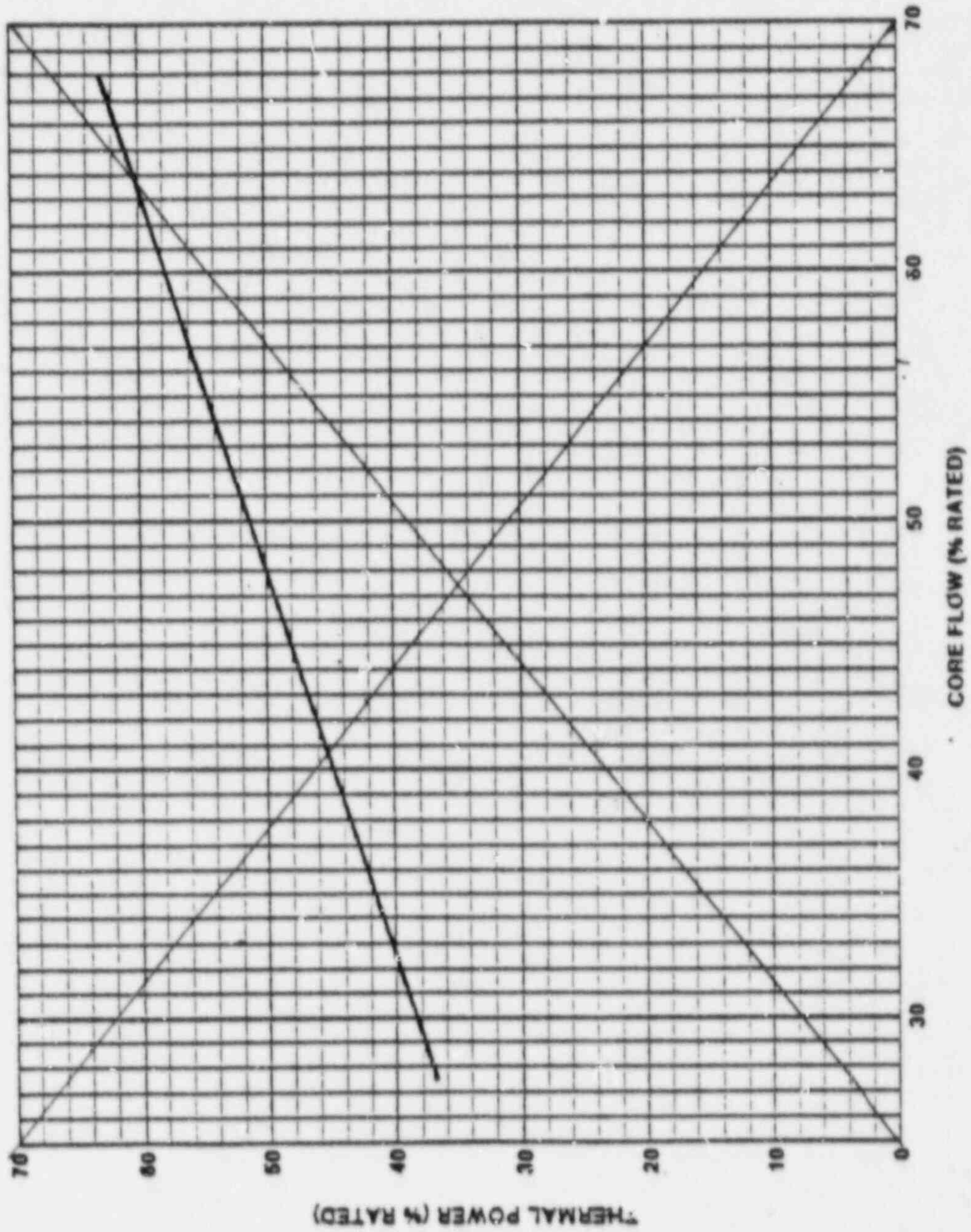


FIGURE 3.4.1.1-1
THERMAL POWER VERSUS CORE FLOW

INSERT H

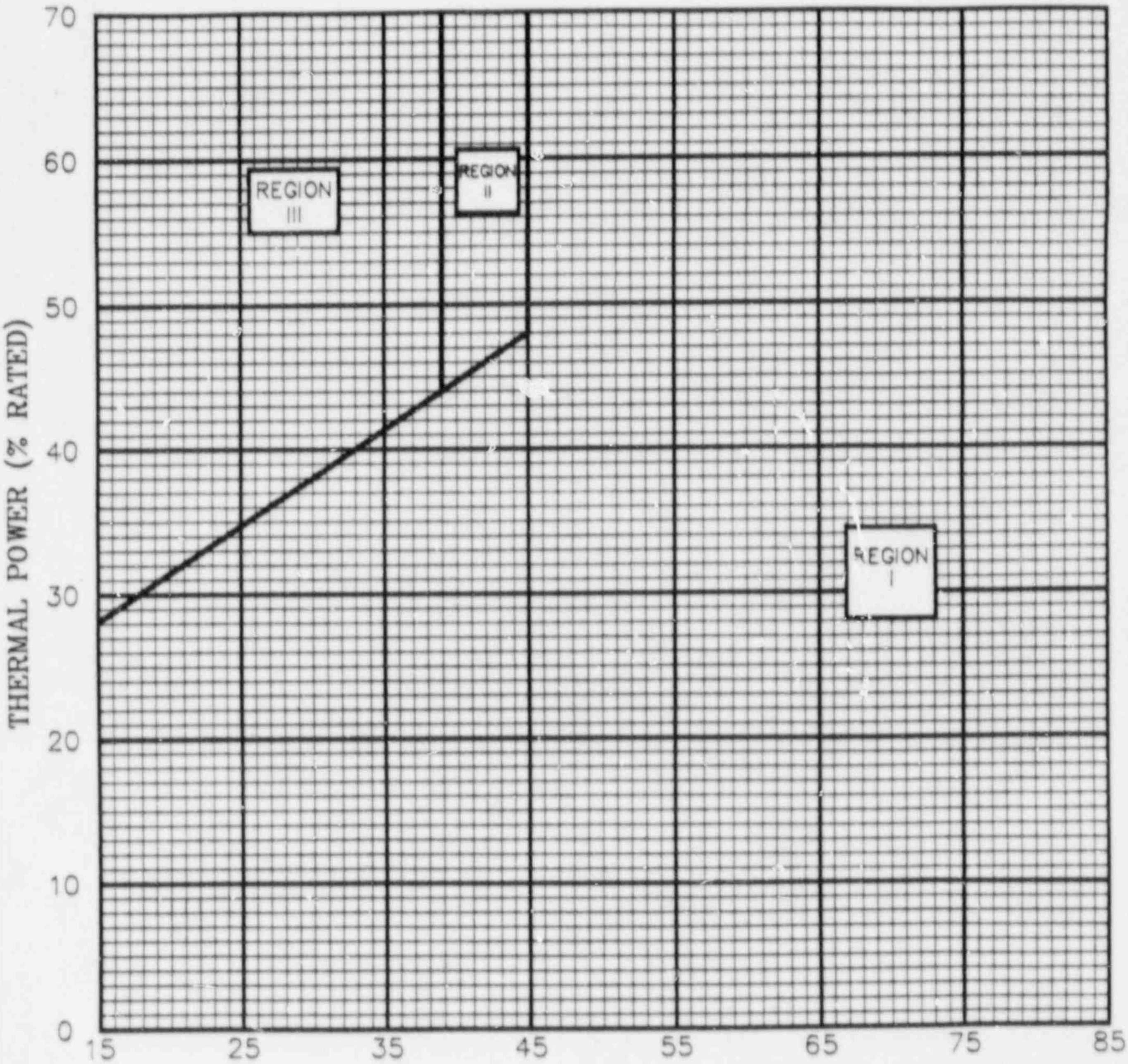


FIGURE 3.4.1.1-1
THERMAL POWER VERSUS CORE FLOW

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

During two Recirulation loop operation each

4.4.1.2.1 ~~Each~~ of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER, and at least once per 24 hours while greater than 25% of RATED THERMAL POWER, by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when both recirculation loop indicated flows are in compliance with Specification 3.4.1.3.

a. The indicated recirculation loop flow differs by more than 10% from the established~~*~~ flow control valve position-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established~~*~~ total core flow value derived from recirculation loop flow measurements.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established~~*~~ patterns by more than 10%.

d. The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

Insert 1

~~*To be determined during the startup test program.~~

Insert "I"

4.4.1.2.2 During single recirculation loop operation, each of the required jet pumps in the operating recirculation loop shall be demonstrated OPERABLE at least once per 24 hours while greater than 25% of RATED THERMAL POWER, by determining recirculation loop flow in the operating loop, total core flow and diffuser-to-lower plenum differential pressure for each jet pump in the operating loop and verifying that no two of the following conditions occur:

a. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established* single recirculation flow control valve position - loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established* jet pump flow/recirculation pump flow characteristic for the operating loop.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established* single recirculation loop patterns by more than 10%.

d. The provisions of specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

---

* To be determined during initial use of single loop operation. Surveillance Requirements of 4.4.1.2 are not required to allow determination of characteristic curves.

REACTOR COOLANT SYSTEM

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* during two recirculation loop operation.

ACTION:

With recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Shutdown one of the recirculation loops ~~Declare the recirculation loop with the lower flow not in operation~~ and take the ACTION required by Specification 3.4.1.1.**

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

** The Provisions of Specification 3.0.4 are not Applicable.

## 3/4.1.3 CONTROL RODS

The specifications of this section (1) ensure that the minimum SHUTDOWN MARGIN is maintained and the control rod insertion times are consistent with those used in the safety analyses, and (2) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem. Therefore, with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period that is long enough to permit determining the cause of the inoperability yet prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those not fully inserted are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shut down for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than ~~1.07~~ the fuel cladding safety limit during the limiting power transient analyzed in Section 15.0 of the FSAR. This analysis shows that the negative reactivity rates, resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than ~~1.07~~ the fuel cladding safety limit. The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and, therefore, the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE-05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

Insert J

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-high scram trip setpoint and the flow biased neutron flux-upscale control rod block trip setpoints of the APRM instruments must be adjusted ~~to ensure that the MCPR does not become less than 1.07~~ or that > 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification, when the combination of THERMAL POWER and CMFLPD indicates a peak power distribution, to ensure that an LHGR transient would not be increased in degraded conditions.

for both two recirculation loop operation and single recirculation loop operation to ensure that MCPR does not become less than the fuel cladding safety limit

Insert "J"

For plant operation with a single recirculation loop, the MAPLHGR limits of figures 3.2.1-1 through 3.2.1-6 are multiplied by 0.84. The constant factor 0.84 is derived from LOCA analyses initiated from single recirculation loop operation to account for earlier boiling transition at the limiting fuel mode compared to the standard LOCA evaluations.

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## Bases Table B 3.2.1-1

## SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters;

Core THERMAL POWER ..................... 3015 Mwt* which corresponds to 105% of rated steam flow

Vessel Steam Output .................... $13.08 \times 10^6$ lbm/hr which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure.............. 1060 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks $2.2 \text{ ft}^2$.

b. Small Breaks $0.09 \text{ ft}^2$.

Fuel Parameters:

| FUEL TYPE | FUEL ASSEMBLY GEOMETRY | PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft) | DESIGN AXIAL PEAKING FACTOR | INITIAL MINIMUM CRITICAL POWER RATIO |
|---|---|---|---|---|
| Initial Core | 8 x 8 | 13.4 | 1.4 | 1.17 ** |

A more detailed listing of input of each model and its source is presented in Section II of NEDE 20566[1] and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes an assembly power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

** For single recirculation loop operation, loss of nucleate boiling is assumed at 0.01 after LOCA regardless of initial MCPR.

## 3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07 and an analysis of abnormal operational transients. For any abnormal operating transient analysis, with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip settings given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR of 1.07, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and is presented in Figure 3.2.3-1. Insert "K" The power-flow map of Figure B 3/4 2.3-1 shows typical regions of plant operation.

The evaluation of a given transient begins with the system initial parameters identified in Reference 2 that are input to a GE core dynamic behavior transient computer program. The codes used to evaluate transients are described in Reference 2. The principal result of this evaluation is the reduction in MCPR caused by transient.

The purpose of the $MCPR_f$ and $MCPR_p$ of Figures 3.2.3-1 and 3.2.3-2 is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_f$s were calculated such that, for the maximum core flow rate and the corresponding THERMAL POWER along the 105%-of-rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105%-of-rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

Insert "K"

Analysis of transients occurring during single recirculation loop operation indicates that the maximum operating limit MCPR will be bounded by the limits in Specification 3.2.3.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Replace with Insert "L"

~~Operation with one reactor coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, and such operation has been determined to be acceptable~~.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design basis accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation. Insert "M" Recirculation loop flow mismatch limits are in compliance with ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. Insert "N"

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Insert "O" ~~Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference were greater than 100°F~~.

The objective of GE BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, at the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.6 was chosen, as the basis for determining the generic region for surveillance, to account for the plant-to-plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a thermal power greater than that specified in Figure 3.4.1.1-1 (Reference 1).

Plant-specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case the degree of conservatism can be reduced since plant-to-plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

## 3/4.4 REACTOR COOLANT SYSTEM

### BASES

RECIRCULATION SYSTEM (Continued)

Neutron flux noise limits are also established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1 to 12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of GE BWR fuel and are found to be of negligible consequence (Reference 2). In addition, stability tests at operating BWRs have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles of 5 to 10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Typically, neutron flux noise levels show a gradual increase in absolute magnitude as core flow is increased (constant control rod pattern) with two reactor recirculation loops in operation. Therefore, the baseline neutron flux noise level obtained at a specific core flow can be applied over a range of core flows. To maintain a reasonable variation between the low flow and high flow ends of the flow range, the range over which a specific baseline is applied should not exceed 20% of rated core flow with two recirculation loops in operation. Data from tests and operating plants indicate that a range of 20% of rated core flow will result in approximately a 50% increase in neutron flux noise level during operation with two recirculation loops. Baseline data should be taken near the maximum rod line at which the majority of operation will occur. However, baseline data taken at lower rod lines (i.e., lower power) will result in a conservative value since the neutron flux noise level is proportional to the power level at a given core flow.

References

(1) "BWR Core Thermal-Hydraulic Stability," Service Information Letter 380, Revision 1, February 1984.

(2) G. A. Watford, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria," December 1982 (NEDE 22277-P).

Insert "P"

### 3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) is to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig, in accordance with the ASME Code. A total of 9 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowabl values for the worst case upset transient. Any combination of 4 SRVs operating in the relief mode and 5 SRVs operating in the safety mode is acceptable.

## Insert "L"

The impact of single recirculation loop operation upon plant safety has been assessed and single recirculation loop operation is permitted if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2; APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively; MAPLHGR limits are decreased by the factor given in Specification 3.2.1 (Reference 3). MCPR operating limits are adjusted per specification 3/4.2.3, for both single and two recirculation loop operation.

Additionally, surveillances on the volumetric flow rate of the operating recirculation loop is imposed to exclude the possibility of excessive core internal vibration. The surveillance on differential temperatures below 30% THERMAL POWER or 50% rated recirculation loop flow is to mitigate the undue thermal stress on the vessel nozzles, recirculation pump and vessel bottom head during extended operation in the single recirculation loop mode.

## Insert "M"

During single loop operation the jet pump operability surveillances are only performed for the jet pumps in the operating recirculation loop, as the loads on the inactive jet pumps are expected to be very low due to the low flow in the reverse direction through the jet pumps.

## Insert "N"

In the case where the mismatch limits cannot be maintained during two recirculation loop operation, continued operation is permitted in a single recirculation loop operation mode.

Insert "O"

Sudden equalization of a temperature difference $100^{\circ}F$ between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

Insert "P"

(3) "Single-Loop Operation Analysis for River Bend Station, Unit 1," NEDO-31441, May 1987

Enclosure 3

Single-Loop Operation Analysis for
River Bend Station, Unit 1 May 1987
(NEDO-31441)