Docket No. 50-397

June 4, 1991

Mr. G. C. Sorensen, Manager
Regulatory Programs
Washington Public Power Supply System
3000 George Washington Way
P.O. Box 968
Richland, Washington 99352

Dear Mr. Sorensen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. NPF-21 FOR THE WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 77311)

The Commission has issued the enclosed Amendment No. 94 to the Facility Operating License for the WPPSS Nuclear Project No. 2. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated August 2, 1990 (G02-90-132) and as amended by letters dated February 25 (G02-91-038), April 19 (G02-91-081), May 6 (G02-91-090), and May 20, 1991 (G02-91-104).

The amendment removes values for cycle specific parameter limits which change with each core reload from the Technical Specifications as discussed in Generic Letter 88-16, "Removal of Cycle Specific Parameter Limits from Technical Specifications," and transfers the cycle specific parameter limits to the Core Operating Limits Report (COLR). The COLR will be developed for each operating cycle.

During the review of your submittals, it became necessary to request additional information and clarification on a number of occasions. For example, you referenced a topical report which had not been reviewed and approved for use by the NRC at the time of your submittal. We would suggest that you carefully review your future submittals for completeness and technical accuracy.

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

Sincerely,

Original signed by:

Patricia L. Eng, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation



Enclosures:
1. Amendment No.94 to NPF-21
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. G. C. Sorensen
Washington Public Power Supply System

WPPSS Nuclear Project No. 2
(WNP-2)

cc:
Mr. J. W. Baker
WNP-2 Plant Manager
Washington Public Power Supply System
P.O. Box 968, MD 927M
Richland, Washington 99352

G. E. C. Doupe, Esq.
Washington Public Power Supply System
3000 George Washington Way
P. O. Box 968
Richland, Washington 99532

Mr. R. G. Waldo, Chairman
Energy Facility Site Evaluation Council
Mail Stop PY-11
Olympia, Washington 98504

Mr. Alan G. Hosler, Licensing Manager
Washington Public Power Supply System
P. O. Box 968, MD 956B
Richland, Washington 99352

Mr. A. Lee Oxsen, Acting
Managing Director for Operations
Washington Public Power Supply System
P. O. Box 968, MD 1023
Richland, Washington 99352

Mr. Gary D. Bouchey, Director
Licensing and Assurance
Washington Public Power Supply System
P. O. Box 968, MD 280
Richland, Washington 99352

Regional Administrator, Region V
U.S. Nuclear Regulatory Commission
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

Chairman
Benton County Board of Commissioners
P. O. Box 190
Prosser, Washington 99350-0190

Mr. R. C. Sorensen
U. S. Nuclear Regulatory Commission
P. O. Box 69
Richland, Washington 99352

Nicholas S. Reynolds, Esq.
Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

(10)

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

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. NPF-21

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

   A. The application for amendment by the Washington Public Power Supply System (licensees) dated August 2, 1990 and as amended by letters dated February 25, April 19, May 6, and May 20, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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;

   D. The issuance of this 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 amended by 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-21 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 94 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James E Dyer

James E. Dyer, Director
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 4, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 94

FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

| Remove Pages | Insert Pages |
|---|---|
| i | i |
| v | v |
| xviii | xviii |
| xix | xix |
| xx | xx |
| xx(a) | xx(a) |
| xxi | xxi |
| 1-1 | 1-1 |
| 1-2 | 1-2 |
| 3/4 2-1 | 3/4 2-1 |
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| 3/4 2-3 | |
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1.0 DEFINITIONS PAGE

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## 1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE BUNDLE EXPOSURE

1.2 The AVERAGE BUNDLE EXPOSURE is equal to the sum of the axially averaged exposure of all the fuel rods in the specified bundle divided by the number of fuel rods in the bundle.

AVERAGE PLANAR EXPOSURE

1.3 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.4 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.7 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

## CORE ALTERATION

1.8 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

## CORE OPERATING LIMITS REPORT

1.8A The CORE OPERATING LIMITS REPORT is the WNP-2 specific document that provides CORE OPERATING LIMITS for the current operating reload cycle. These cycle-specific CORE OPERATING LIMITS shall be determined for each reload cycle in accordance with Specification 6.9.3. Plant operation within these Operating Limits is addressed in individual specifications.

## CRITICAL POWER RATIO

1.9 The CRITICAL POWER RATIO (CPR) shall be that power in the assembly which is calculated by application of the appropriate critical power correlation to cause some point in the assembly to experience boiling transition divided by the actual assembly operating power.

## DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

## $\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

## EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

## 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 shall not exceed the limits specified in the Core Operating Limits Report.

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 specified in the Core Operating Limits Report, 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.

### SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the Core Operating Limits Report.

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.

## 3/4.2.2 APRM SETPOINTS

### LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale 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:

| TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|
| $S \leq (0.66W + 51\%)T$ | $S \leq (0.66W + 54\%)T$ |
| $S_{RB} \leq (0.66W + 42\%)T$ | $S_{RB} \leq (0.66W + 45\%)T$ |

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 108.5 million lbs/h.

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. T is always less than or equal to 1.

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-upscale 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 2 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 the MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale 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 MFLPD greater than or equal to FRTP.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

## 3/4.2.3 MINIMUM CRITICAL POWER RATIO

### LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be:

Greater than or equal to the applicable MCPR limit specified in the Core Operating Limits Report.

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

ACTION: With MCPR less than the applicable MCPR limit specified in the Core Operating Limits Report, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25 percent of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit specified in the Core Operating Limits Report.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15 percent 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 MCPR.

## 3/4.2.4 LINEAR HEAT GENERATION RATE

### LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the values specified in the Core Operating Limits Report.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit specified in the Core Operating Limits Report, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit specified in the Core Operating Limits Report:

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 on a LIMITING CONTROL ROD PATTERN for LHGR.

## 3/4.2.6 POWER/FLOW INSTABILITY

### LIMITING CONDITION FOR OPERATION

3.2.6 Operation with THERMAL POWER/core flow conditions which lay in Region A of Figure 3.2.6-1 is prohibited.

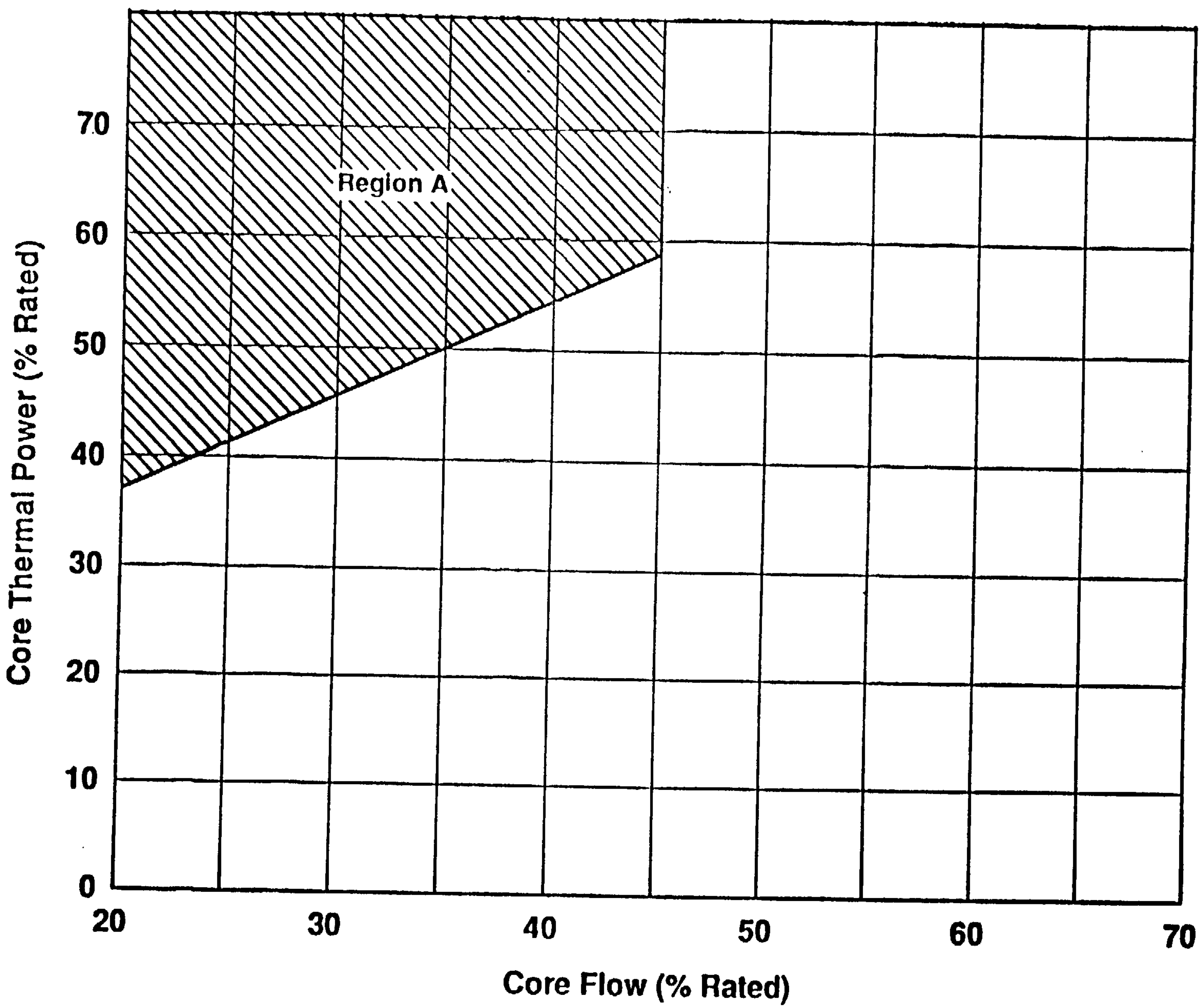
APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than 39% of RATED THERMAL POWER and core flow is less than or equal to 45% of rated core flow.

ACTION:

With THERMAL POWER/core flow conditions which lay in Region A of Figure 3.2.6-1, then as soon as practical, but in all cases within 15 minutes, initiate a MANUAL SCRAM.

### SURVEILLANCE REQUIREMENTS

4.2.6 The THERMAL POWER/core flow conditions shall be verified to lay outside Region A of Figure 3.2.6-1 once per 24 hours when operating in the region of APPLICABILITY.




Operating Region Limits of Specification 3.2.6
Figure 3.2.6-1

## 3/4.2.7 STABILITY MONITORING - TWO LOOP OPERATION

### LIMITING CONDITION FOR OPERATION

3.2.7 The stability monitoring system shall be operable* and the decay ratio of the neutron signals shall be less than .75 when operating in the region of APPLICABILITY.

APPLICABILITY: OPERATIONAL CONDITION 1, with two recirculation loops in operation and THERMAL POWER/core flow conditions which lay in Region C of Figure 3.2.7-1.

ACTION:

a. With decay ratios of any two (2) neutron signals greater than .75 or with two (2) consecutive decay ratios on any single neutron signal greater than .75:

   As soon as practical, but in all cases within 15 minutes, initiate action to reduce the decay ratio by either decreasing THERMAL POWER with control rod insertion or increasing core flow with recirculation flow control valve manipulation. The starting or shifting of a recirculation pump for the purpose of decreasing decay ratio is specifically prohibited.

b. With the stability monitoring system inoperable and when operating in the region of APPLICABILITY:

   As soon as practical, but in all cases within 15 minutes, initiate action to exit the region of APPLICABILITY by either decreasing THERMAL POWER with control rod insertion or increasing core flow with recirculation flow control valve manipulation. The starting or shifting of a recirculation pump for the purpose of exiting the region of APPLICABILITY when the stability monitoring system is inoperable is specifically prohibited. Exit the region of APPLICABILITY within one (1) hour.

### SURVEILLANCE REQUIREMENTS

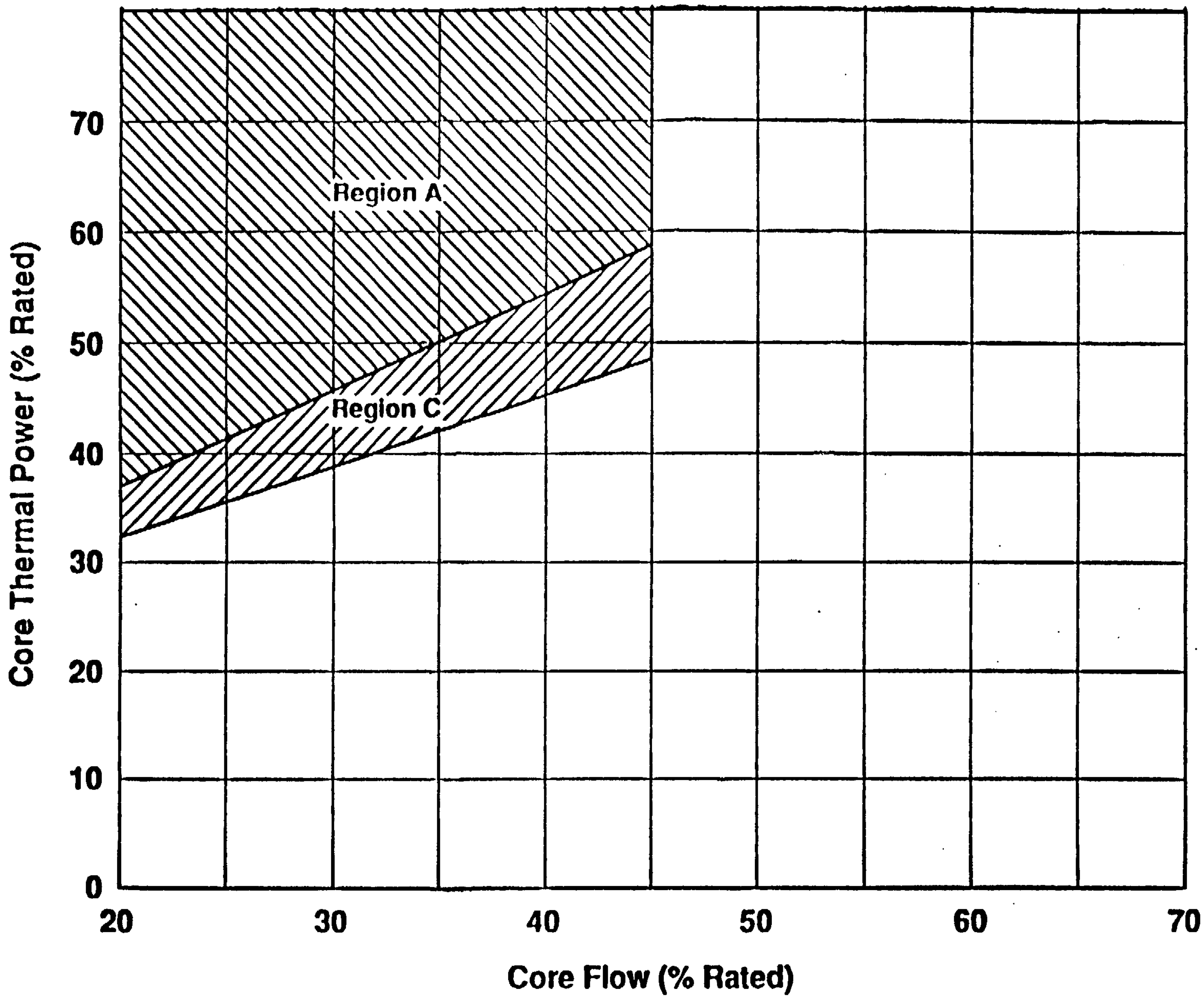
4.2.7.1 The provisions of Specification 4.0.4 are not applicable.

4.2.7.2 The stability monitoring system shall be demonstrated operable* within one (1) hour prior to entry into the region of APPLICABILITY.

4.2.7.3 Decay ratio and peak-to-peak noise values calculated by the stability monitoring system shall be monitored when operating in the region of APPLICABILITY.

---

*Verify that the stability monitoring system data acquisition and calculational modules are functioning, and that displayed values of signal decay ratio and peak-to-peak noise are being updated. Detector levels A and C (or B and D) of one LPRM string in each of the nine core regions (a total of 18 LPRM detectors) shall be monitored. A minimum of four (4) APRMs shall also be monitored.




Operating Region Limits of Specification 3.2.7
Figure 3.2.7-1

## 3/4.2 POWER DISTRIBUTION LIMITS

## 3/4.2.8 STABILITY MONITORING - SINGLE LOOP OPERATION

### LIMITING CONDITION FOR OPERATION

3.2.8 The stability monitoring system shall be operable* and the decay ratio of the neutron signals shall be less than .75 when operating in the region of APPLICABILITY.

APPLICABILITY: OPERATIONAL CONDITION 1, with one recirculation loop in operation and THERMAL POWER/core flow conditions which lay in Region C of Figure 3.2.8-1.

ACTION:

a. With decay ratios of any two (2) neutron signals greater than .75 or with two (2) consecutive decay ratios on any single neutron signal greater than .75:

   As soon as practical, but in all cases within 15 minutes, initiate action to reduce the decay ratio by either decreasing THERMAL POWER with control rod insertion or increasing core flow with recirculation flow control valve manipulation. The starting or shifting of a recirculation pump for the purpose of decreasing decay ratio is specifically prohibited.

b. With the stability monitoring system inoperable and when operating in the region of APPLICABILITY:

   As soon as practical, but in all cases within 15 minutes, initiate action to exit the region of APPLICABILITY by decreasing THERMAL POWER with control rod insertion. Exit the region of APPLICABILITY within one (1) hour.

### SURVEILLANCE REQUIREMENTS

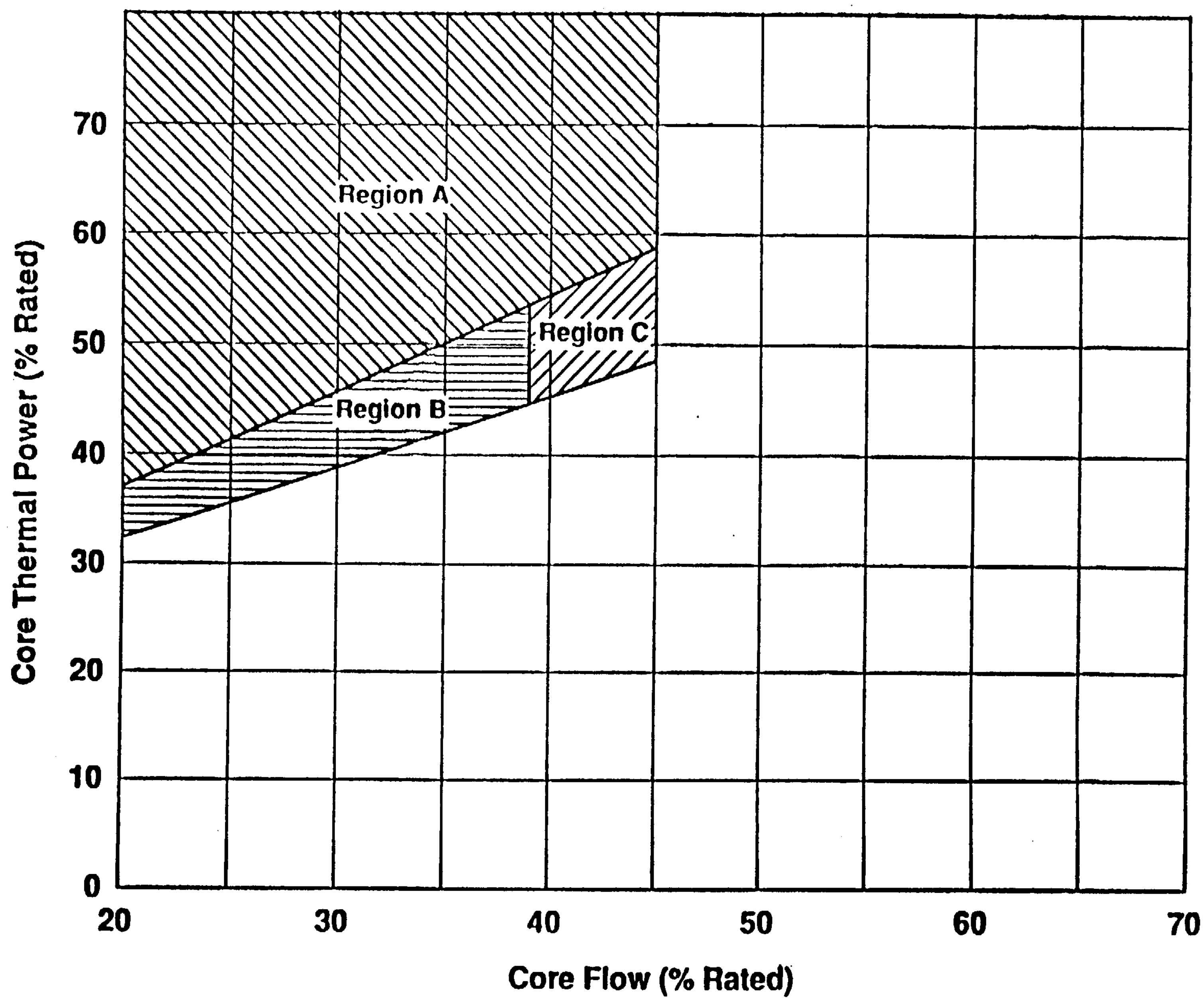
4.2.8.1 The provisions of Specification 4.0.4 are not applicable.

4.2.8.2 The stability monitoring system shall be demonstrated operable* within one (1) hour prior to entry into the region of APPLICABILITY.

4.2.8.3 Decay ratio and peak-to-peak noise values calculated by the stability monitoring system shall be monitored when operating in the region of APPLICABILITY.

---

*Verify that the stability monitoring system data acquisition and calculational modules are functioning, and that displayed values of signal decay ratio and peak-to-peak noise are being updated. Detector levels A and C (or B and D) of one LPRM string in each of the nine core regions (a total of 18 LPRM detectors) shall be monitored. A minimum of four (4) APRMs shall also be monitored.




Operating Region Limits of Specification 3.2.8
Figure 3.2.8-1

## BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

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

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. For GE fuel, the peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady-state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for GE fuel is this LHGR of the highest powered rod divided by its local peaking factor which results in a calculated LOCA PCT much less than 2200°F. The Technical Specification APLHGR for ANF fuel is specified to assure the PCT following a postulated LOCA will not exceed the 2200°F limit. The limiting value for APLHGR is specified in the Core Operating Limits Report.

The calculational procedure used to establish the APLHGR specified in the Core Operating Limits Report is based on a loss-of-coolant accident analysis. The analysis was performed using calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. These models are referenced in Specification 6.9.3.

## 3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analysis. In addition, the APRM setpoints must be adjusted for both two recirculation loop operation and single recirculation loop operation to ensure that the MCPR does not become less than the fuel cladding safety limit or that $\geq$ 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 MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

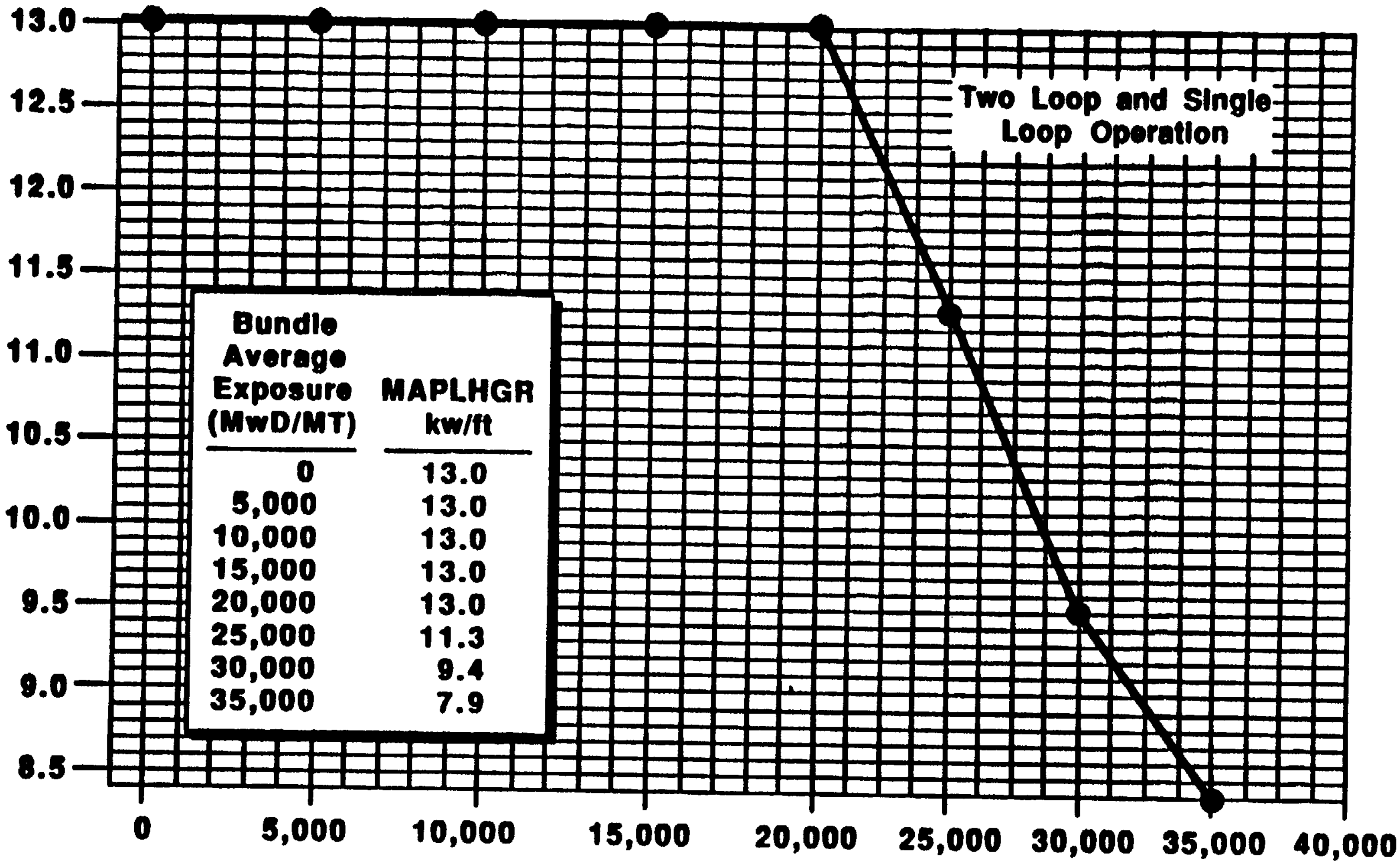
The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the 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 setting 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, the required minimum operating limit MCPR of Specification 3.2.3 is specified in the Core Operating Limits Report.

The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to an ANF core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and nonpressurization events are referenced in Specification 6.9.3. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the flow dependent $MCPR_f$ specified in the Core Operating Limits Report is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required $MCPR_f$ is the maximum of the rated flow MCPR and the reduced flow MCPR both specified in the Core Operating Limits Report. $MCPR_f$ assures that the Safety Limit MCPR will not be violated. $MCPR_f$ is only calculated for the manual flow control mode. Automatic flow control operation is not permitted.

Lead Fuel Assemblies (LFA's) from Advanced Nuclear Fuels (ANF), General Electric (GE) and ABB Atom (ABB) reside in the reactor core. Analyses performed by the three vendors indicate that the transient CPR changes for the LFA's are greater than the CPR change calculated for the dominant ANF 8x8 fuel, due primarily to the shorter thermal time constants of the smaller diameter rods. All vendors state that their LFA's have inherently higher thermal margins (by design) than the dominant 8x8 fuel. Each vendor chose to address the CPR limit in a slightly different fashion. These methods are discussed as follows.




| Bundle Average Exposure (MwD/MT) | MAPLHGR kw/ft |
|---|---|
| 0 | 13.0 |
| 5,000 | 13.0 |
| 10,000 | 13.0 |
| 15,000 | 13.0 |
| 20,000 | 13.0 |
| 25,000 | 11.3 |
| 30,000 | 9.4 |
| 35,000 | 7.9 |

ANF 8x8 Reload Fuel
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Bundle Average Exposure

Figure 3.2.1-3

## 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.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Verify that the requirements of LCO 3.2.6 and LCO 3.2.8 are met, or comply with the associated ACTION statements.

2. Verify that THERMAL POWER/core flow conditions lay outside Region B of Figure 3.4.1.1-1.

   With THERMAL POWER/core flow conditions which lay in Region B of Figure 3.4.1.1-1, as soon as practical, but in all cases within 15 minutes, initiate action to exit Region B by either decreasing THERMAL POWER with control rod insertion or increasing core flow with flow control valve manipulation. Within 1 hour exit Region B. The starting or shifting of a recirculation pump for the purpose of exiting Region B is specifically prohibited.

3. Within 4 hours:

   a) Place the recirculation flow control system in the Local Manual (Position Control) mode, and

   b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit per Specification 2.1.2, and,

   c) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for General Electric fuel limit to the single recirculation loop operation limit specified in the Core Operating Limits Report, and

   d) Reduce the volumetric flow rate of the operating recirculation loop to $\leq$ 41,725** gpm.

*See Special Test Exception 3.10.4.

**This value represents the actual volumetric recirculation loop flow which produces 100% core flow at 100% THERMAL POWER. This value was determined during the Startup Test Program.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

e) Perform Surveillance Requirement 4.4.1.1.2 if THERMAL POWER is ≤ 25%*** of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 10%*** of rated loop flow.

4. The provisions of Specification 3.0.4 are not applicable.

5. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 With one reactor coolant system recirculation loop not in operation, at least once per 8 hours verify that:

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

b. The volumetric flow rate of the operating loop is ≤ 41,725 gpm.**

**This value represents the actual volumetric recirculation loop flow which produces 100% core flow at 100% THERMAL POWER. This value was determined during the Startup Test Program.

***Final values were determined during Startup Testing based upon actual THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.

ANNUAL REPORTS (Continued)

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions; and

b. Documentation of all challenges to main steam line safety/relief valves.

c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Cleanup system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience 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.

REPORTABLE OCCURRENCES

6.9.1.7 DELETED

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 DELETED

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

## THIRTY DAY WRITTEN REPORTS

6.9.1.9 DELETED

## ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.10 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

## SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.11 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Radioactive Effluent Release Report shall also include once a year an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

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*In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

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

CORE OPERATING LIMITS REPORT

6.9.3.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for Specifications 3.2.1 and 3.4.1.

b. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.3.

c. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4.

and shall be documented in the CORE OPERATING LIMITS REPORT.

CORE OPERATING LIMITS REPORT (Continued)

6.9.3.2 The analytical methods used to determine the core operating limits shall be those topical reports and those revisions and/or supplements of the topical report previously reviewed and approved by the NRC, which describe the methodology applicable to the current cycle. For WNP-2 the topical reports are:

1. ANF-1125(P)(A), and Supplements 1 and 2, "ANFB Critical Power Correlation," April 1990
2. Letter, R. C. Jones (NRC) to R. A. Copeland (ANF), "NRC Approval of ANFB Additive Constants for ANF 9x9-9X BWR Fuel," dated November 14, 1990
3. XN-NF-524(P)(A), Revision 2 and Supplements 1 and 2, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," November 1990
4. ANF-913(P)(A), Volume 1, Revision 1 and Volume 1, Supplements 2, 3 and 4, "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis," August 1990
5. ANF-CC-33(P)(A), Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option," January 1991
6. XN-NF-80-19(P)(A), Volume 1, Supplements 3 and 4, "Exxon Nuclear Methodology for Boiling Water Reactors," November 1990
7. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," June 1986
8. XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," January 1987
9. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactor Reload Fuel," September 1986
10. ANF-89-014(P), "Generic Mechanical Design for ANF 9x9-IX and 9x9-9X BWR Reload Fuel," May 1989
11. ANF-89-014(P), Supplement 1, "Generic Mechanical Design of ANF 9x9-IX and 9x9-9X BWR Reload Fuel," June 1990
12. Letter, A.C. Thadani (NRC) to R. Copeland (ANF) "Acceptance for Referencing of Topical Report ANF-89-014(P), Rev. 0, Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel," April 22, 1991
13. XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology," November 1983
14. NEDE-24011-P-A-6, "General Electric Standard Application for Reactor Fuel," April 1983

CORE OPERATING LIMITS REPORT (Continued)

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

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

## 6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. NPF-21

# WASHINGTON PUBLIC POWER SUPPLY SYSTEM

# NUCLEAR PROJECT NO. 2

# DOCKET NO. 50-397

## 1.0 INTRODUCTION

By letter dated August 2, 1990 and as amended by letters dated February 25, April 19, May 6, and May 20, 1991, Washington Public Power Supply System (WPPSS) proposed certain changes to the Technical Specifications for WPPSS Nuclear Project No. 2 (WNP-2). The February 25, April 19, May 6, and May 20, 1991, letters provided clarifying information that did not change the scope of the proposed amendment and did not affect the initial proposed no significant hazards consideration determination.

The amendment removes values for cycle specific parameter limits which change with each core reload from the Technical Specifications as discussed in Generic Letter 88-16, "Removal of Cycle Specific Parameter Limits from Technical Specifications," and transfers the cycle specific parameter limits to the Core Operating Limits Report (COLR). The COLR will be developed for each operating cycle.

## 2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definition section of the TS was modified to include the Core Operating Limits Report which establishes cycle/reload-specific parameter limits on a unit-specific basis in accordance with an NRC approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of individual cycle-specific parameter limits with a reference to the COLR.

  (a) Specifications 3.2.1 and 3.4.1, Surveillance Requirement 4.2.1 and Bases 3/4.2.1

  The Average Planar Linear Heat Generation Rate (APLHGR) Limits, associated surveillance requirements and bases are to be specified in the COLR.

(b) Specification 3.2.3, Surveillance Requirement 4.2.3.1 and Bases 3/4.2.3

The Minimum Critical Power Ratio (MCPR) operating limits, associated surveillance requirements and bases are to be specified in the COLR.

(c) Specification 3.2.4

The Linear Heat Generation Rate (LHGR) limits for this specification are to be specified in the COLR.

(3) Specification 6.9.3 was added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the values of these limits be established using an NRC approved methodology and be consistent with all applicable limits of this safety analysis. The approved methodology is the following:

(a) ANF-1125(P)(A), and Supplements 1 and 2, "ANFB Critical Power Correlation," April 1990.

(b) Letter, R. C. Jones (NRC) to R. A. Copeland (ANF), "NRC Approval of ANFB Additive Constants for ANF 9x9-9X BWR Fuel," dated November 14, 1990.

(c) XN-NF-524(P)(A), Revision 2 and Supplements 1 and 2, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," November 1990.

(d) ANF-913(P)(A), Volume 1, Revision 1 and Volume 1, Supplements 2, 3, and 4, "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis," August 1990.

(e) ANF-CC-33(P)(A), Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option," January 1991.

(f) XN-NF-80-19(P)(A), Volume 1, Supplements 3 and 4, "Exxon Nuclear Methodology for Boiling Water Reactors," November 1990.

(g) XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," June 1986.

(h) XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," January 1987.

(i) XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactor Reload Fuel," September 1986.

(j) ANF-89-014(P), "Generic Mechanical Design for ANF 9x9-IX and 9x9-9X BWR Reload Fuel," May 1989.

(k) ANF-89-014(P), Supplement 1, "Generic Mechanical Design of ANF 9x9-IX and 9x9-9X BWR Reload Fuel," June 1990.

(l) Letter, A.C. Thadani (NRC) to R. Copeland (ANF), "Acceptance for Referencing of Topical Report ANF-89-014(P), Rev. 0, Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel," dated April 22, 1991.

(m) XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology," November 1983.

(n) NEDE-24011-P-A-6, "General Electric Standard Application for Reactor Fuel," April 1983.

Item (l) is the SER attached to a letter dated April 22, 1991 from A. C. Thadani (NRC) to R. Copeland (ANF) on the subject "Acceptance for Referencing of Topical Report ANF-89-014(P), Rev. 0, Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9x9-1X and 9x9-9X BWR Reload Fuel." Items (j) and (k) were reviewed and approved as a part of the SER referenced in Item (l) above.

In our SER of ANF-89-014(P), Revision 0, Supplement 1, we stated that a plant-specific analysis of a combined seismic and LOCA loading event on the effect of fuel bundle uplift is required for each reload application of ANF 9x9-IX and 9x9-9X fuel. A lifted bundle may move laterally and interfere with control blade insertion. In addition, the licensee should evaluate the impact of a mixed core situation of 8x8 and 9x9 fuel bundles on the combined seismic/LOCA event. The primary concern in a mixed core situation for the seismic/LOCA event is the different pressure drops between the 8x8 and 9x9 fuel bundles.

In a letter dated May 6, 1991 (G02-91-090), from G. C. Sorensen (WPPSS) to NRC, "Nuclear Plant No. 2, Operating License NPF-21 Answer to NRC Question on Seismic/LOCA Response of 9x9-9X Fuel," the licensee performed a plant-specific analysis to address the fuel bundle uplift for the coming reload cycle. The licensee determined that the bounding condition for the fuel bundle uplift during seismic and LOCA loading that covers the mixed core condition is for a full core of 9x9-9X fuel assemblies because a full core of 9x9-9X fuel has the largest hydraulic resistance, i.e., largest pressure drop. For a full core of 9x9-9X fuel assemblies, the licensee compared the liftoff energy between 8x8 and 9x9 fuel, the results showed that the full 9x9-9X core remains seated during a combined seismic and LOCA loading and thus no fuel assembly liftoff is observed. We agree with the licensee analysis and conclude that the concern of fuel uplift in a mixed core of 8x8 and 9x9-9X or a full core of 9x9-9X during seismic and LOCA loading has been adequately addressed by the licensee.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to the NRC, prior to operation with the new parameter limits.

On the basis of the review of the above item, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC approved methodology, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

## 3.0 STATE CONSULTATION

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

## 4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change a surveillance requirement. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 5.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: T. Huang

Date: June 4, 1991

## 6.0 REFERENCE

1. Letter (G02-90-132) from G. C. Sorensen (WPPSS) to NRC, dated August 2, 1990.

2. Letter (G02-91-038) from G. C. Sorensen (WPPSS) to NRC, dated February 25, 1991.

3. Letter (G02-91-081) from G. C. Sorensen (WPPSS) to NRC, dated April 19, 1991.

4. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

5. Letter (G02-91-090) from G. C. Sorensen (WPPSS) to NRC, dated May 6, 1991.

6. Letter (G02-91-104) from G. C. Sorensen (WPPSS) to NRC, dated May 20, 1991.