Docket

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

August 5, 1988

Docket No. 50-397

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

Dear Mr. Sorensen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 62 TO FACILITY OPERATING LICENSE NO. NPF-21 - WPPSS NUCLEAR PROJECT NO. 2 (TAC NO. 67538)

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 62 to Facility Operating License NPF-21 to the Washington Public Power Supply System for WPPSS Nuclear Project No. 2, located in Benton County near Richland, Washington. This amendment is in response to your letter dated March 7, 1988 as supplemented on May 13, 1988.

This amendment revises limiting conditions for operation and instrumentation setpoints in the technical specifications to allow the operation of WNP-2 up to a power level of 75% power with one recirculation loop operating to the design burnup of the reload fuel of 35,000 MWD/MT bundle outage. The amendment makes revisions to related sections of the technical specifications to improve clarity. The amendment adds a new section on power/flow instability and moves the specification addressing flux noise from the section on instrumentation (Section 3/4.3) to the section on power distribution limits (Section 3/4.2).

As stated in your supplemental letter of May 13, 1988 (G02-88-116) this is not an authorization to depart from the bounds of the assumptions of 10 CFR 51.52.

DF01
1/1

A copy of the related safety evaluation supporting Amendment No. 62 to Facility Operating License No. NPF-21 is enclosed. Enclosed for your information is a copy of the Notice of Issuance, that has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Robert B. Samworth

Robert B. Samworth, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 62 to Facility Operating License No. NPF-21
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

August 5, 1988

A copy of the related safety evaluation supporting Amendment No.   to Facility Operating License No. NPF-21 is enclosed. Enclosed for your information is a copy of the Notice of Issuance, that has been forwarded to the Office of the Federal Register for publication.

Sincerely,

original signed by

Robert B. Samworth, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 62 to Facility Operating License No. NPF-21
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

DISTRIBUTION
Docket File
NRC & Local PDRs
PD5 Reading
GMHolahan
RBSamworth
OGC-Rockville
DHagan
EJordan
BGrimes
TBarnhart (4)
Wanda Jones
EButcher
ACRS (10)
GPA/PA
ARM/LFMB

DRSP/PDV	DRSP/PDV	OGC	DRSP/PDV
JLee	RBSamworth:dr		GWKnighton
7/ /88	8/5/88	7/29/88	8/5/88

OFFICIAL RECORD COPY

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

WPPSS Nuclear Project No. 2
(WNP-2)

cc:
Nicholas S. Reynolds, Esq.
Bishop, Cook, Purcell
& Reynolds
1400 L Street NW
Washington, D.C. 20005-3502

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

Mr. Curtis Eschels, 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
Assistant Managing Director for Operations
Washington Public Power Supply System
P. O. Box 968, MD 1023
Richland, WA 99352

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

Mr. C. M. Powers
WNP-2 Plant Manager
Washington Public Power Supply System
P. O. Box MD 927M
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
Prosser, Washington 99350

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

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

WPPSS NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 62
License No. NPF-21

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

   A. The application for amendment filed by the Washington Public Power Supply System (the Supply System, also the licensee), dated March 7, 1988 as supplemented on May 13, 1988 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 regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this 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. 62 , 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

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 5, 1988

ENCLOSURE TO LICENSE AMENDMENT NO. 62

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.

REMOVE	INSERT
v	v
vi	vi
xii	xii
xiii	xiii
xx	xx
2-4	2-4
3/4 2-4	3/4 2-4
3/4 2-4c	3/4 2-4c
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
--	3/4 2-11
--	3/4 2-12
--	3/4 2-13
--	3/4 2-14
3/4 3-55	3/4 3-55
3/4 3-102	3/4 3-102
3/4 3-103	3/4 3-103
3/4 3-104	3/4 3-104
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2
B3/4 2-1	B3/4 2-1
B3/4 2-4	B3/4 2-4
--	B3/4 2-5
--	B3/4 2-6
B3/4 3-7	B3/4 3-7
B3/4 3-7a	--
B3/4 4-1	B3/4 4-1

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

BASES

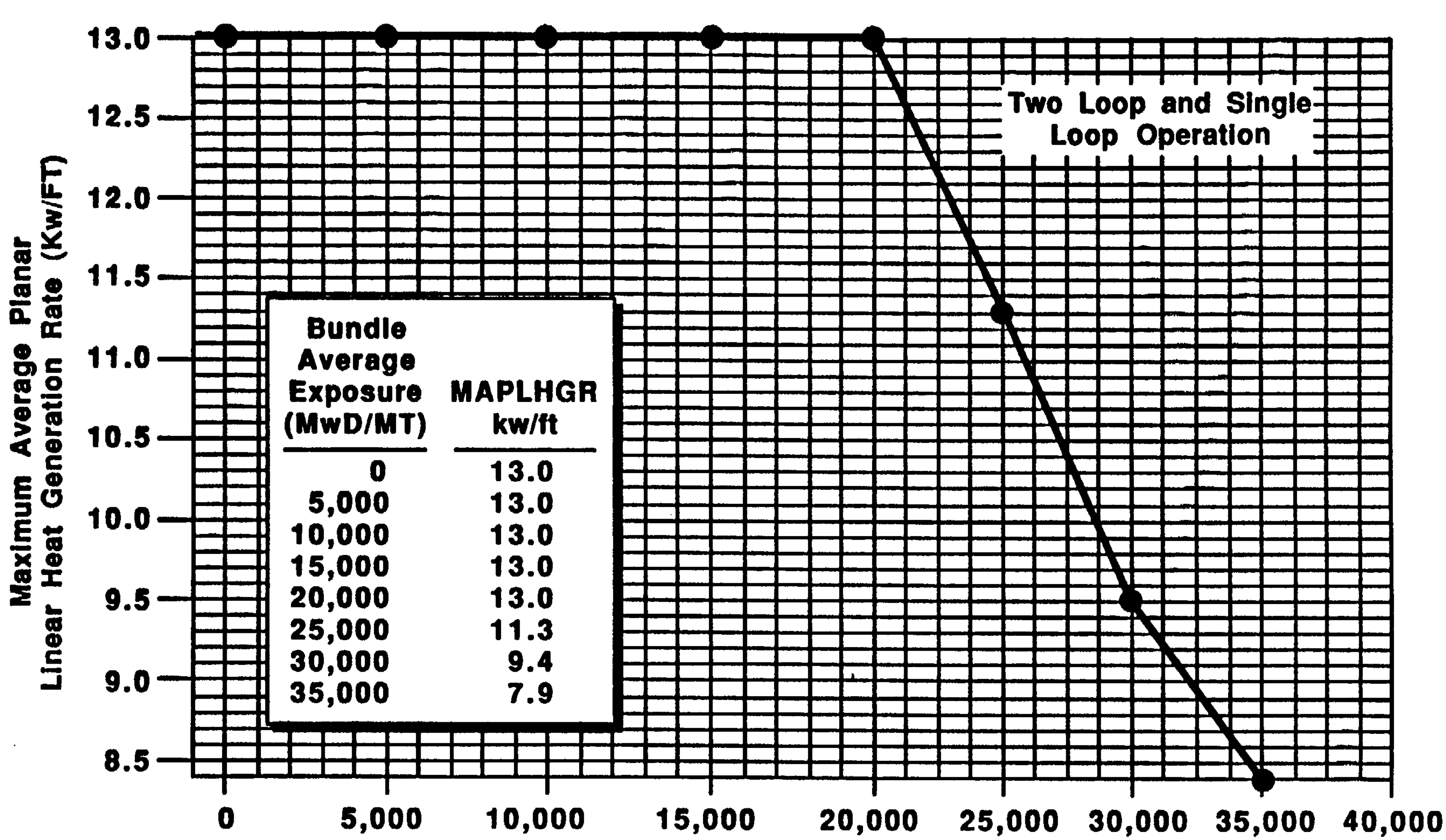
LIST OF FIGURES

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		
1) Flow Biased	≤ 0.66W + 51%, with a maximum of	≤ 0.66W + 54%, with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 10.0% closed	≤ 12.5% closed
6. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background

*See Bases Figure B 3/4 3-1.



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

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

Figure 3.2.1-3

THIS PAGE INTENTIONALLY LEFT BLANK

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:

a. Greater than or equal to the applicable MCPR limit determined from Table 3.2.3-1 during steady state operation at or above rated core flow in two loop operation, or when in single loop operation, or

b. Greater than or equal to the greater of the two values determined from Table 3.2.3-1 and Figure 3.2.3-1 during steady state operation at less than rated core flow when in two recirculation loop operation.

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 determined from Table 3.2.3-1 and Figure 3.2.3-1, 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 determined from Table 3.2.3-1 and Figure 3.2.3-1.

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.

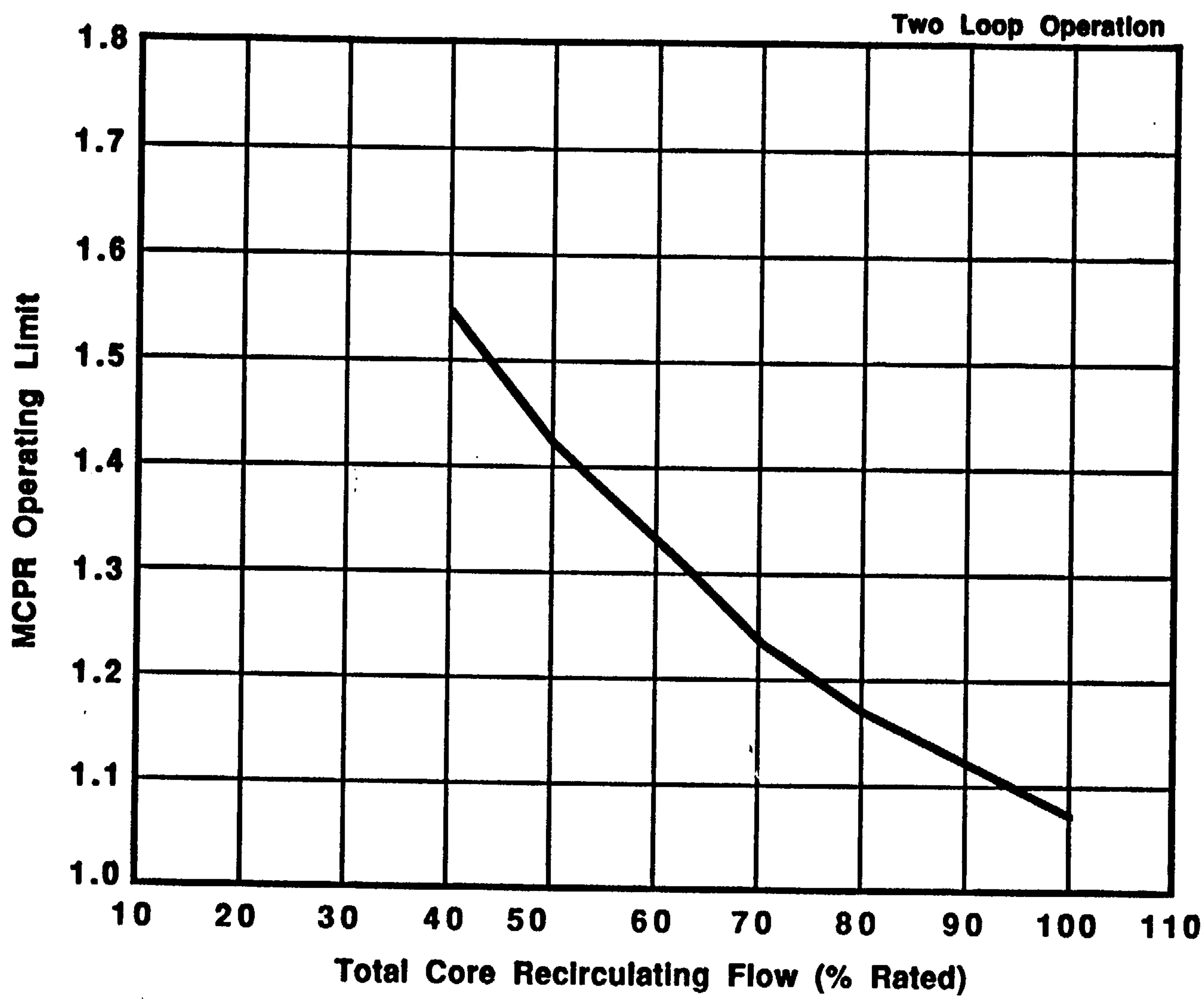
Table 3.2.3-1
MCPR OPERATING LIMITS

	Cycle Exposure	Equipment Status	MCPR Operating Limit Up to 106% Core Flow	
			GE Fuel	ANF Fuel
1.	0 MWD/MTU - 3750 MWD/MTU	*	1.40	1.28
2.	3750 MWD/MTU - EOC MWD/MTU	Normal scram times**	1.40	1.31
3.	3750 MWD/MTU - EOC MWD/MTU	Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.50	1.38
4.	3750 MWD/MTU - EOC MWD/MTU	RPT inoperable Normal scram times	1.50	1.37
5.	3750 MWD/MTU - EOC MWD/MTU	RPT inoperable Control rod insertion bounded by Tech. Spec. limits (3.1.3.4 - p 3/4 1-8)	1.55	1.43
6.	0 MWD/MTU - EOC MWD/MTU	Single loop operation RPT operable Normal scram times**	1.40	1.37

*In this portion of the fuel cycle, operation with the given MCPR operating limits is allowed for both normal and Tech. Spec. scram times and for both RPT operable and inoperable.

**These MCPR values are based on the ANF Reload Safety Analysis performed using the control rod insertion times shown below (defined as normal scram). In the event that surveillance 4.1.3.2 shows these scram insertion times have been exceeded, the plant thermal limits associated with normal scram times default to the values associated with Tech. Spec. scram times (3.1.3.4-p 3/4 1-8), and the scram insertion times must meet the requirements of Tech. Spec. 3.1.3.4.

Position Inserted From Fully Withdrawn	Slowest measured average control rod insertion times to specified notches for all operable control rods for each group of 4 control rods arranged in a a two-by-two array (seconds)
Notch 45	.404
Notch 39	.660
Notch 25	1.504
Notch 5	2.624



Reduced Flow MCPR Operating Limit
Figure 3.2.3-1

3/4.2.6 POWER/FLOW INSTABILITY

LIMITING CONDITION FOR OPERATION

3.2.6 Operation with THERMAL POWER/core flow conditions which lay in the crosshatched region of Figure 3.2.6-1 is prohibited.

APPLICABILITY: OPERATIONAL CONDITION 1

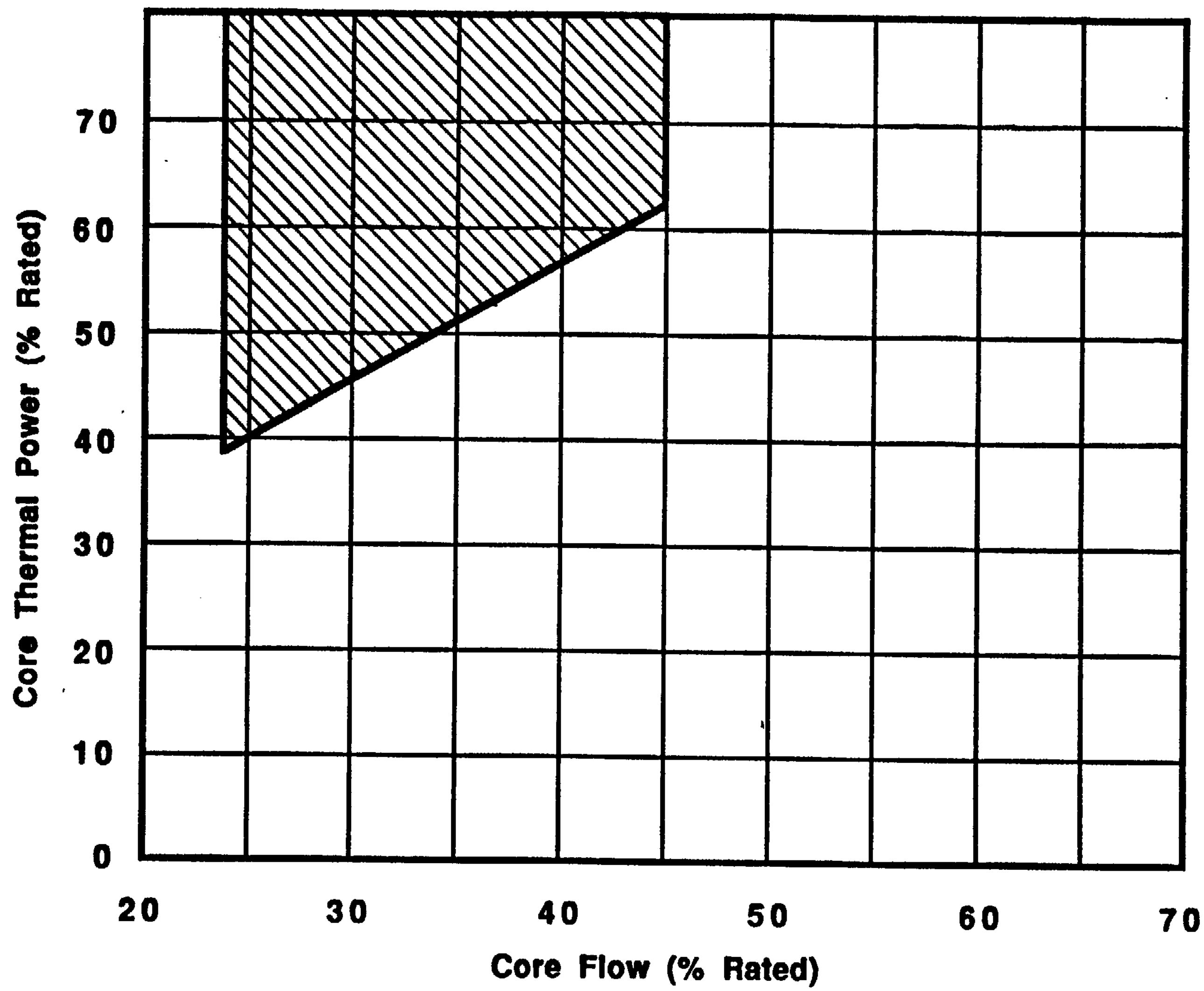
When THERMAL POWER is greater than 39% of RATED THERMAL POWER and core flow is less than or equal to 45% of rated core flow.

ACTION:

With THERMAL POWER/core flow conditions which lay in the crosshatched region of Figure 3.2.6-1, initiate corrective action within 15 minutes to establish a THERMAL POWER/core flow condition which lays outside the crosshatched region within 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The THERMAL POWER/core flow conditions shall be verified to lay outside the crosshatched region of Figure 3.2.6-1 once per 24 hours.



Operating Region Limits of Specification 3.2.6
Figure 3.2.6-1

3/4.2.7 NEUTRON FLUX NOISE MONITORING

LIMITING CONDITION FOR OPERATION

3.2.7 The APRM and LPRM neutron flux noise levels shall not exceed three (3) times their established baseline values when operating in the region of APPLICABILITY.

APPLICABILITY: OPERATIONAL CONDITION 1 with THERMAL POWER/core flow in Region B of Figure 3.2.7-1, with two reactor coolant system recirculation loops in operation and total core flow less than 45% of rated total core flow, or with one reactor coolant system recirculation loop not in operation.

ACTION:

a. If baseline APRM and LPRM neutron flux noise levels have not been established for the appropriate reactor coolant system condition (one or two loop operation) since the most recent CORE ALTERATION, then:

   Within 2 hours exit the region of APPLICABILITY. Establish baseline APRM and LPRM neutron flux noise levels prior to re-entering Region B of Figure 3.2.7-1.

b. If baseline APRM and LPRM neutron flux noise levels have been established for the appropriate reactor coolant system condition (one or two loop operation) since the most recent CORE ALTERATION, then:

   With the APRM or LPRM neutron flux noise levels greater than three (3) times their established noise levels, initiate corrective action within 15 minutes to restore the noise levels to within the required limits within 2 hours or reduce THERMAL POWER to below the region of APPLICABILITY within the next 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.7.1 The provisions of Specification 4.0.4 are not applicable.

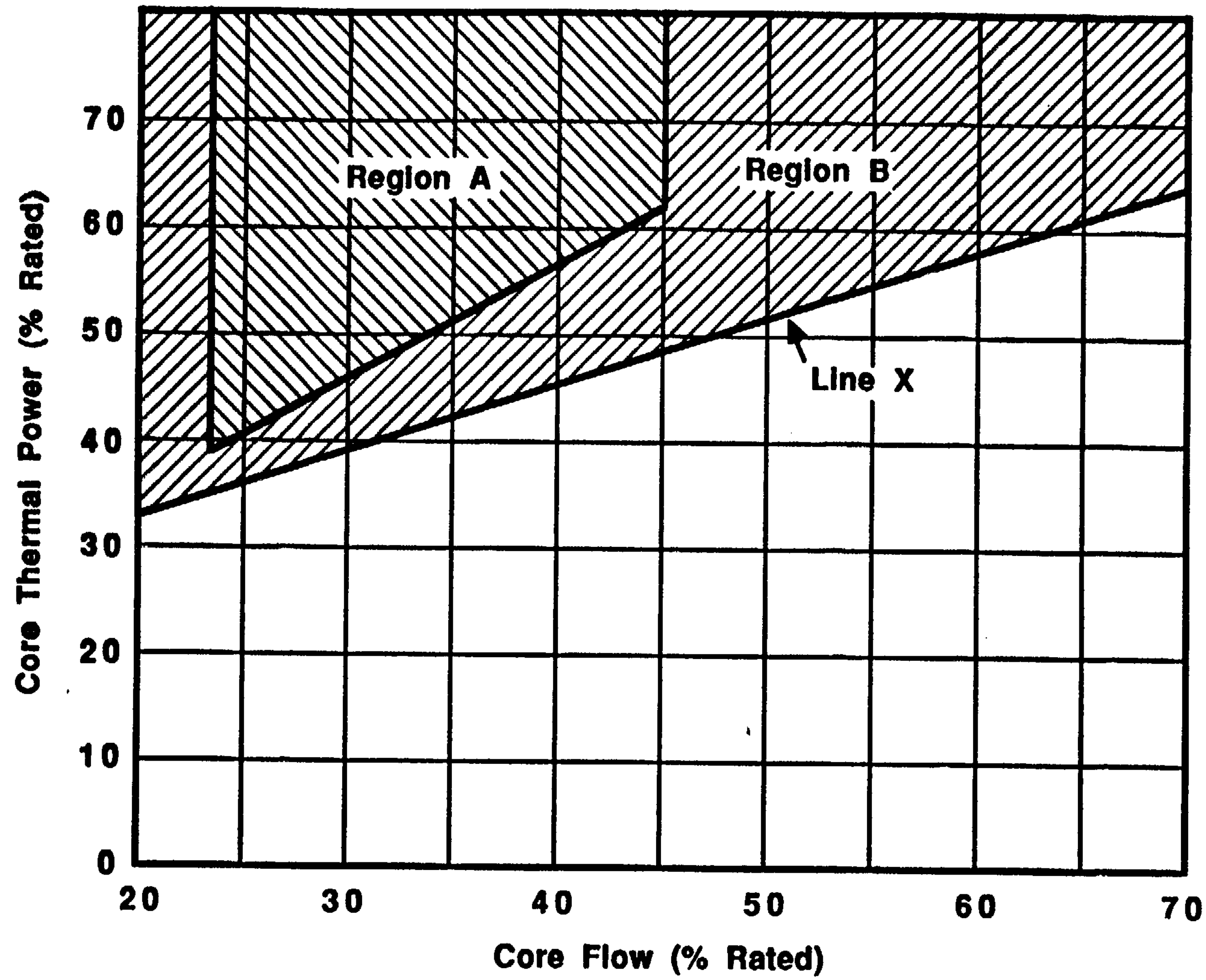
4.2.7.2 The APRM and LPRM neutron flux noise levels shall be determined to be less than or equal to three (3) times their established baseline values:

a. At least once per 8 hours, and

b. Within 30 minutes after completion of a THERMAL POWER increase of greater than or equal to 5% of rated THERMAL POWER.

---

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



Operating Region Limits of Specification 3.2.7
Figure 3.2.7-1

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66 W + 40%	≤ 0.66 W + 43%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux Upscale	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*
b. Inoperative	N.A.	N.A.
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	N.A.	N.A.
b. Upscale	$< 1 \times 10^5$ cps	$< 1.6 \times 10^5$ cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 0.7 cps	≥ 0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	< 108/125 divisions of full scale	< 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	< 527 ft 2 in. elevation	< 527 ft 4 in. elevation
b. Scram Trip Bypass	N.A.	N.A.
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108/125 divisions of full scale	≤ 111/125 divisions of full scale
b. Inoperative	N.A.	N.A.
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation

*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.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

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. Within 15 minutes:

      a. Verify that core flow is greater than or equal to 39% of rated core flow or that THERMAL POWER/core flow conditions lay below the line in Figure 3.4.1.1-1. With core flow less than 39% of rated core flow and THERMAL POWER/core flow conditions above the line in Figure 3.4.1.1-1, initiate action to reduce THERMAL POWER to below the line in Figure 3.4.1.1-1 or increase core flow to greater than or equal to 39% of rated core flow within the next 4 hours.

      b. Verify that the requirements of LCO 3.2.7 are met, or comply with the associated ACTION statement within the specified time limits.

   2. 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 by 0.01 to 1.07 per Specification 2.1.2, and,

      c) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) for General Electric fuel limit to a value of 0.84 times the two recirculation loop operation limit per Specification 3.2.1, 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 $\leq$ 25%*** of RATED THERMAL POWER or the recirculation loop flow in the operating loop is $\leq$ 10%*** of rated loop flow.

f) Reduce recirculation loop flow in the operating loop until the core plate ΔP noise does not deviate from the established core plate ΔP noise patterns by more than 100%.

3. The provisions of Specification 3.0.4 are not applicable.

4. 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 $\leq$ 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.

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 shown in Figures 3.2.1-1 and 3.2.1-2 for two recirculation loop operation and Figures 3.2.1-4 and 3.2.1-5 for single loop operation. Figure 3.2.1-3 applies to both single and two loop operation.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, and 3.2.1-5 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 described in Reference 1 or XN-NF-80-19, Volumes 2, 2A, 2B and 2C, Rev. 1.

MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

3/4.2.6 POWER/FLOW INSTABILITY

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., power shape, bundle power, and bundle flow).

In February, 1984, GE issued SIL 380 addressing boiling instability and supplying several recommendations. In this SIL, the power/flow map was divided into several regions of varying concern. It also discussed the objectives and philosophy of "detect and suppress," coining the phrase.

The ANF topical report for COTRAN (XN-NF-691P) discusses boiling instability. The SER written on this topical (dated May 10, 1984) interprets the topical to require that the detect-and-suppress surveillance be used in regions which have code calculated decay ratios .75 or greater and that operation is forbidden in regions having calculated decay ratios of .9 and greater.

The NRC Generic Letter 86-02 addressed both GE and ANF (then EXXON) stability calculation methodology and stated that due to uncertainties, General Design Criterias 10 and 12 could not be met using analytic procedures on a BWR 5 design. The letter espoused GE SIL 380 and stated that General Design Criterias 10 and 12 could be met by imposing the SIL 380 recommendations in operating regions of potential instability. The NRC concluded that regions of potential instability constituted calculated decay ratios of .8 and greater by the GE methodology and .75 and greater by the EXXON methodology.

POWER/FLOW INSTABILITY (Continued)

Predicated on the SIL 380 endorsement, WNP-2 has divided the power/flow map on the following boundary lines:

1. 80% rod line
2 45% core flow line
3. APRM rod block line minus 3% power
4. Natural Circulation flow line
5. Minimum Forced Circulation for normal recirculation lineup.

This division conforms to the SIL 380 recommendations with a 3% power penalty on the APRM rod block line. For LCO 3.2.6, the region of concern is bounded by the APRM rod block line, minus 3% power, the natural circulation flow line, and the 45% core flow line. Calculated decay ratios between the two flow lines and on the APRM rod block line minus 3% must be less than .9. Operation in the region between the two flow lines and above the rod block line minus 3% is forbidden due to the potential for boiling instabilities.

For the ease of annual licensing submittals, a 3% margin from the rod block line is taken to avail the opportunity to submit with no Technical Specification changes under the provisions of 10 CFR 50.59. This 3% provides margin to assure that vendor stability calculations can easily support the allowable operating region. For calculational ease the power boundary is linearized between two points, (24% Flow, 39% Power) and (45% Flow, 62% Power).

3/4.2.7 NEUTRON FLUX NOISE MONITORING

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 patterns, 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 0.75 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).

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 1-12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual

NEUTRON FLUX NOISE MONITORING (Continued)

recirculation loop operation. 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 5-10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) 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.

In the case of single loop operation (SLO), the normal neutron flux noise may increase more rapidly when reverse flow occurs in the inactive jet pumps. This justifies a smaller flow range under high flow SLO conditions. Baseline data should be taken at flow intervals which correspond to less than a 50% increase in APRM neutron flux noise level. If baseline data are not specifically available for SLO, then baseline data with two recirculation loops in operation can be conservatively applied to SLO since for the same core flow SLO will exhibit higher neutron flux noise levels than operation with two loops. However, because of reverse flow characteristics of SLO, the core flow/drive flow relationship is different than the two loop relationship and therefore the baseline data for SLO should be based on the active loop recirculation drive flow, and not the core flow. Because of the uncertainties involved in SLO at high reverse flows, baseline data should be taken at or below the power specified in Figure 3.4.1.1-1. This will result in approximately a 25% conservative baseline value if compared to baseline data taken near the rated rod line and will therefore not result in an overly restrictive baseline value, while providing sufficient margin to cover uncertainties associated with SLO.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

3/4.3.9 FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater system/main turbine trip system actuation instrumentation is provided to initiate the feedwater system/main turbine trip system in the event of reactor vessel water level equal to or greater than the level 8 setpoint associated with a feedwater controller failure.

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and been found to be acceptable provided the unit is operated in accordance with the single recirculation loop operation Technical Specifications herein.

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.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. Where the recirculation loop flow mismatch limits cannot be maintained during two recirculation loop operation, continued operation is permitted in the single recirculation loop operation mode.

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. 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 was greater than 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve capacity is designed to limit the primary system pressure, including transients, in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1971, Nuclear Power Plant components (up to and including Summer 1971 Addenda). The Code allows a peak pressure of 110% of design pressure (1250 (design) X 1.10 = 1375 psig maximum) under upset conditions. In addition, the Code specifications require that the lowest valve setpoint be at or below design pressure and the highest valve setpoint be set so that total accumulated pressure does not exceed 110% of the design pressure.

The safety valve sizing evaluation assumes credit for operation of the scram protective system which may be tripped by one of two sources; i.e., a direct position switch or neutron flux signal. The direct scram signal is derived from position switches mounted on the main steamline isolation valves (MSIV's) or the turbine stop valve, or from pressure switches mounted on the dump valve of the turbine control valve hydraulic actuation system. The position switches are actuated when the respective valves are closing, and following 10% travel of full stroke. The pressure switches are actuated when a fast closure of the control valves is initiated. Further, no credit is taken for power operation of the pressure relieving devices. Credit is only taken for

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.62 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

WPPSS NUCLEAR PROJECT NO.2

DOCKET NO. 50-397

1.0 INTRODUCTION

By letter dated March 7, 1988 (G02-88-053) and supplement dated May 13, 1988 (G02-88-116) the Washington Public Power Supply System proposed certain changes to the Technical Specifications for WNP-2. (The license submitted a letter, dated April 12, 1988, containing corrections to the March 7, 1988 letter.) The objective of the proposed amendment is to allow the operation of WNP-2 up to a power level of 75% with one recirculation loop operating to the design burnup of the reload fuel of 35,000 MWD/MT bundle average. The current technical specifications limit the operation to a power level of 72% with one recirculation loop operating, and further, prohibit single loop operation beyond a fuel burnup of 15,000 MWD/MT bundle average.

On the same date the licensee filed an amendment application and supporting documentation for the license conditions applicable during the fourth fuel cycle of operation (Ref. 1).

The fuel burnup and power level limitations on single loop operation in the current technical specifications are the bounds of the analyses of single loop operation previously submitted by the licensee and reviewed by the staff. The licensee submitted three reports (Refs 2, 3 and 4) containing additional safety analyses in support of the March 7, 1988 amendment application.

Sections 2.2.1, 3/4.2.1, 3/4.2.2, 3/4.2.3, 3/4.3.6, 3/4.4.1 and associated bases for these sections are proposed to be revised to incorporate new limits for single loop operation.

The licensee also proposed reorganization of one section of the technical specifictions. Currently Section 3/4.3.10, "Neutron Flux Monitoring Instrumentation," contains a limiting condition of operation for APRM and LPRM neutron flux noise levels. The licensee argues that this is not an instrumentation issue and has proposed to move this requirement to section 3/4.2, "Power Distribution Limits." The licensee also noted that the action statement of existing technical specification 3/4.3.10 did not have a corresponding LCO to get into the action statement. To correct this the licensee proposed adding a technical specification addressing flow/power instability.

By letter dated May 13, 1988 (G02-88-116) the licensee clarified that although the peak bundle burnup limits extend to 35,000 MWD/MT the discharged fuel will conform to the assumptions of 10 CFR 51.52.

A separate license amendment application (Ref. 1) for Cycle 4 fuel reloading was based on a more conservative analysis which would allow the licensee to refuel in the future without needing further license amendments. The staff reviewed and approved that application and issued its safety evaluation as part of Amendment 59 to License No. NPF-21. That application included more conservative MCPR operating limits (Table 3.2.3-1) for single loop operation. However those proposed single loop limits were not included in the revisions issued with Amendment 59 because the review of this current application for single loop operation had not been completed. The more conservative single loop MCPR limits requested for the Cycle 4 reload are included in Table 3.2.3-1 at this time.

2.0 EVALUATION

2.1 Single Loop Operation

WNP-2 is currently permitted to operate with one recirculation loop out-of-service at power levels up to 72% of rated power and for burnups up to 15,000 MWD/MTU on the ANF reload fuel. The initial core GE fuel is permitted to operate to burnups of 44,000 MWD/MTU with single loop operation (SLO). ANF has performed analyses to evaluate operation of WNP-2 during SLO at power levels up to 75% power and for burnups up to 35,000 MWD/MTU on the ANF reload fuel (Refs. 2, 3). These analyses were performed for the most limiting transients and accidents and for the LOCA at the maximum extended power state during SLO using approved methodology. Consideration of the higher uncertainties for core flow, radial power and axial power during SLO results in an increase in the MCPR safety limit of 0.01 (to 1.07), the same value supported by the GE analyses for the initial core GE fuel with a single recirculation loop in operation. However, the two loop MCPR limits at SLO flow conditions bound the required MCPR limits for SLO conditions including the higher safety limit MCPR. The results of the ANF analyses support the use of a single bounding MCPR operating limit of 1.35 and two loop ANF fuel MAPLHGR limits for SLO for current and future cycles. The only exception to this is, in the event the two loop CRWE event operating limit exceeds the current value of 1.26, the single loop CRWE results should be reviewed to ensure they remain conservative. The licensee has committed to this review (Ref 4). For GE fuel, the reduction of the MAPLHGR limit to a value of 0.84 times the two recirculation loop operation limit still remains. The staff finds the proposed conditions for SLO acceptable.

WNP-2 will be entering its fourth cycle of operation and is approaching an equilibrium cycle. Analysis results between Cycles 3 and 4 show little change. As a result, WNP-2 has chosen to add some small CPR penalties for margin to envelope future anticipated analysis results (Ref. 5). The intent is to be able to submit future reload applications which require no Technical Specification changes, thereby allowing application per the provisions of 10 CFR 50.59. Including these self-imposed penalties for single loop operation increases the MCPR operating limit for GE fuel to 1.40 and for ANF fuel to 1.37. The staff reviewed these more conservative limits as part of its review of the application for Cycle 4 fuel reloading and found the limits acceptable.

2.2 Reorganization of Technical Specifications

Specification 3/4.3.10 Neutron Flux Monitoring Instrumentation states that the APRM and LPRM neutron flux noise levels shall not exceed three times their established baseline values when operating in the allowable region of Figure 3.3.10-1. The action statement prescribes steps the licensee must take when the neutron noise flux noise levels do not meet the limiting condition for operation. The action statement further prescribes steps the licensee must take when reactor power and core flow fall outside of the allowable region of the figure. The licensee contends that this LCO and associated action statements constitute power distribution limits rather than instrumentation limits. It is proposed to move this to part 3.4.2 of the technical specifications and to include an LCO corresponding to each of the two parts of the action statement. The staff finds that since the limits are not changed there is no safety significance to this revision and it is, therefore, acceptable.

2.3 Technical Specification Changes

To accomplish the proposed changes for SLO limits, incorporation of a modified power/flow operating map, nuclear limits, and Bases and reference updates, the following Technical Specification changes have been requested:

(1) Specification 2.2.1: The APRM flow biased simulated thermal power (high) trip setpoint and allowable value for SLO have been removed from Table 2.2.1-1. This is acceptable because the two loop limits at SLO flow conditions have been shown to bound the required limits for SLO conditions and no distribution need be made in the reactor protection system instrumentation setpoints.

(2) Specification 3/4.2.1: Figure 3.2.1-3 has been revised to include SLO. These MAPLHGR limits are not impacted by the small enrichment change associated with ANF fuel loaded for Cycle 4. In addition, for SLO these limits also apply to ANF fuel consistent with the flow dependent MCPR curve (1.35 at 50% of rated flow). The change is, therefore, acceptable.

(3) Specification 3/4.2.2: Separate values of the APRM flow biased simulated thermal power-upscale scram trip setpoint and flow biased neutron flux-upscale control rod block trip setpoint for SLO have been removed. This is acceptable for the same reasons as stated in (1) above.

(4) Specification 3/4.2.3: Table 3.2.3-1 and Figure 3.2.3-1 have been revised to reflect Cycle 4 MCPR operating limits. These new limits are based on the Cycle 4 reload safety analysis which has been evaluated and approved in Section 2 and are, therefore, acceptable.

(5) Specification 3/4.2.6: A new Section on Power/Flow Instability has been added. This was done to clarify the intent of action statement in existing technical specification 3.3.10. The power/flow operating map which was part of figure 3.3.10-1 is now Figure 3.2.6-1 and has been modified to bound future stability calculations. Based on the staff's evaluation of the stability analyses and surveillance requirements for Cycle 4, the proposed changes are acceptable.

(6) Specification 3/4.2.7: A new Section on Neutron Flux Noise Monitoring has been created by moving existing requirements from Section 3/4.3.10. This provides assurance that neutron flux limit cycle oscillations will be detected and suppressed by monitoring APRM and LPRM neutron flux noise levels while operating in this region and is, therefore, acceptable. The associated changes to the wording in the LCO and Applicability sections, the Action section, and the Surveillance Requirements are also acceptable.

(7) Specification 3/4.3.6: The RBM and APRM instrumentation trip setpoints for two loop operation and SLO have been combined. As stated earlier, this is acceptable.

(8) Specification 3/4.3.10: This specification has been relocated to the Power Distribution Limits section of the Technical Specifications as described in item (6) above. Since the intent of this LCO is to monitor neutron flux noise levels to detect the approach of an unstable region of operation and has little to do with instrument calibration, it is more appropriately located in the Power Distribution Limits section immediately adjacent to related LCO 3/4.2.6, Power/Flow Stability. The relocation is, therefore, acceptable.

(9) Specification 3/4.4.1: Actions required to be initiated within 15 minutes have been removed from within the 4 hour criteria and localized. The Action requirement to reduce the scram and rod block trip setpoints when in SLO is no longer necessary since the Cycle 4 analysis evelopes operation up to 75% thermal power with single loop core flows and drive flows. In addition, the reduction in MAPLHGR limit during SLO is changed to only be necessary for the GE fuel as mentioned in Section 2.7. These changes are consistent with the Cycle 4 reload analysis and are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, the NRC prepared an environmental assessment. A notice of environmental assessment and finding of no significant impact was published in the Federal Register on August 2, 1988 (53 FR 29092).

4.0 CONTACT WITH STATE OFFICIAL

The State of Washington was provided a copy of the amendment application and did not have any comment.

5.0 CONCLUSION

The staff has reviewed the reports submitted for single loop operation using ANF methodology and analysis. Based on this review, the staff concludes that the safety of operating with a single recirculation loop out of service for an extended period of time has been adequately demonstrated. The Technical Specification changes submitted reflect the use of of acceptable methodology and the operating limits associated with those changes. The proposed operation of WNP-2 is therefore, acceptable.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Laurence I. Kopp

Dated: August 5, 1988

REFERENCES

1. Letter from G. C. Sorensen (WPPSS) to NRC, Request for Amendment to Technical Specifications - Reload License Amendment (Cycle 4), March 7, 1988 (G02-88-054).

2. ANF-87-119, "WNP-2 Single Loop Operation Analysis," September 1987.

3. ANF-87-118, "WNP-2 LOCA Analysis for Single Loop Operation," September 1987.

4. WPPSS-EANF-115, "WNP-2 Single Loop Operation Summary Report," February 1988.

5. WPPSS-EANF-119, "WNP-2 Cycle 4 Reload Summary Report," February 1988.