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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
License No. DPR-59

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

   A. The application for amendment by the Power Authority of the State of New York (the licensee) dated October 11, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

   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. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 98 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George W Rivenbark for

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 6, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 98

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

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1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective:

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications:

A. Reactor Pressure $>$ 785 psig and Core Flow $>$ 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit, hereafter called the Safety Limit. An MCPR Limit of 1.08 shall apply during single-loop operation.

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications:

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. IRM - The IRM flux scram setting shall be set at $\leq$120/125 of full scale.

1.1 (cont'd)

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 785 psig)

When the reactor pressure is $\leq$ 785 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 (cont'd)

A.1.b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

APRM - The APRM flux scram setting shall be $\leq$ 15 percent of rated neutron flux with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

c. APRM Flux Scram Trip Settings (Run Mode)

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be:

$S \leq 0.66\ W + 54\%$ for two loop operation

or:

$S \leq (0.66\ W + 54\% - 0.66\Delta W)$ for single loop operation

where:

S = Setting in percent of rated thermal power (2436 MWT)

W = Recirculation flow in percent of rated

ΔW= Difference between two loop and single loop effective drive flow at the same core flow. ($\Delta W = 0$ for two loop operation. ΔW for single loop operation is to be determined upon implementation of single loop operation.)

2.1 (cont'd)

For no combination of recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

1.1 (cont'd)

D. Reactor Water Level (Hot or Cold Shutdown Conditions)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 18 inches above the Top of Active Fuel when it is seated in the core.

2.1 (cont'd)

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%)(FRP/MFLPD)$$

for two loop operation or,

$$S \leq (0.66\ W + 54\% - 0.66\Delta W)(FRP/MFLPD)$$

for single loop operation.

Where:

FRP = fraction of rated thermal power (2436 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

(2) Fixed High Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

$$S \leq 120\%\ \text{Power}$$

1.1 (cont'd)

2.1 (cont'd)

A.1.d <u>APRM Rod Block Setting</u>

The APRM Rod block trip setting shall be:

$S \leq (0.66\ W + 42\%)$
for two loop operation or,

$S \leq (0.66\ W + 42\% - 0.66 \Delta W)$
for single loop operation.

where:

S = Rod block setting in percent of thermal power (2346 MWt).

W = Loop recirculation flow rate in percent of rated.

ΔW = Difference between two loop and single loop effective drive flow at the same core flow.

In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$S \leq (0.66\ W + 42\%)(FRP/MFLPD)$
for two loop operation or,

$S \leq (0.66\ W + 42\% - 0.66\Delta W)(FRP/MFLPD)$
for single loop operation.

where:

FRP = fraction of rated thermal power (2436 MWt)

2.1 (cont'd)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

1.1 BASES

1.1 FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR $>$ 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

A. Reactor Pressure $>$ 785 psig and Core Flow $>$ 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variable, i.e., the operating domain. The current load line limit analysis contains the current operating domain map. The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the MCPR operating conditions in specification 3.1.B, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The MCPR fuel cladding safety limit is increased by 0.01 for single-loop operation as discussed in Reference 2. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including the uncertainty in the boiling transition correlation as described in Reference 1. The uncertanties employed in deriving the Safety Limit are

1.1 BASES (Cont'd)

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

D. Reactor Water Level (Hot or Cold Shutdown Condition)

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the Safety Limit at 18 in. above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

E. References

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10958.

2. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO 24281, August 1980.

3. Generic Reload Fuel Application, NEDE 24011 - P-A and Appendices.

BASES

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the FitzPatrick Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition 2535 MWt. The analyses were based upon plant operation in accordance with the operating map given in the current load line limit analysis. In addition, 2436 is the licensed maximum power level of FitzPatrick, and this represents the maximum steady-state power which shall not knowingly be exceeded.

The transient analyses performed for each reload are given in Reference 2. Models and model conservatism are also described in this reference. As discussed in Reference 4, the core wide transient analysis for one recirculation pump operation is conservatively bounded by two-loop operation analysis, and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

Fuel cladding integrity is assured by the operating limit MCPR's for steady state conditions given in Specification 3.1.B. These operating limit MCPR's are derived from the established fuel cladding integrity Safety Limit, 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.

The most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO. The type of transients evaluated were 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, the required operating limit MCPR of Specification 3.1.B is obtained.

The evaluation of a given transient begins with the system initial parameters shown in the current reload analysis and Reference 2 that are input to a core dynamic behavior transient computer program described in References 1 and 3. The output of these programs along with the initial MCPR form the input for the further analyses of the thermally limited bundle with a single channel transient thermal hydraulic code. The principal result of the evaluation is the reduction in MCPR caused by the transient.

2.1 BASES (Cont'd)

C. References

1. Linford, R.B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO-10802, Feb., 1973

2. "General Electric Fuel Application", NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).

3. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", NEDO-24154, October, 1978.

4. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO-24281, August, 1980.

3.1 (CONTINUED)

MCPR Operating Limit for Incremental Cycle Core Average Exposure

At RBM Hi-trip level setting	BOC to EOC-2GWD/t	EOC-2GWD/t to EOC-1GWD/t	EOC-1GWD/t to EOC
S = .66W + 39%	1.24	1.29	1.31
S = .66W + 40%	1.27	1.29	1.31
S = .66W + 41%	1.27	1.29	1.31
S = .66W + 42%	1.29	1.29	1.31
S = .66W + 43%	1.30	1.30	1.31
S = .66W + 44%	1.34	1.34	1.34

During single loop operation, the operating limit MCPR shall be increased by 0.01 from that in the table above to reflect the increase in safety limit MCPR. (See Specification 1.1.A)

C. MCPR shall be determined daily during reactor power operation at $\geq$ 25% of rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

D. When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

E. Verification of the limits set forth in specification 3.1.B shall be performed as follows:

1. The average scram time to notch position 38 shall be:

$$\tau_{AVE} \leq \tau_B$$

2. The average scram time to notch position 38 is determined as follows:

$$\tau_{AVE} = \sum_{i=1}^{n} Ni\,\tau_i \Big/ \sum_{i=1}^{n} Ni$$

where: n = number of surveillance tests performed to date in the cycle, Ni = number of active rods measured in

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6) (16)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X		2 Instrument Channels	A
3	IRM High Flux	$\leq$120/125 of full scale	X	X		8 Instrument Channels	A
3	IRM Inoperative		X	X		8 Instrument Channels	A
2	APRM Neutron Flux-Startup(15)	$\leq$15% Power	X	X		6 Instrument Channels	A
2	APRM Flow Referenced Neutron Flux (Not to exceed 117%) (12)(13)(14)(17)	S$\leq$(0.66W+54%)(FRP/MFLPD)			X	6 Instrument Channels	A or B
2	APRM Fixed High Neutron Flux(14)	$\leq$120% Power			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B

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REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (cont'd)

14. The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux singal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is place in the Run position.

16.* During the proposed Hydrogen Addition Test, the normal background radiation level will increase by approximately a factor of 5 for peak hydrogen concentration. Therefore, prior to performance of the test, the Main Steam Line Radiation Monitor Trip Level Setpoint will be raised to $\leq$ three times the increased radiation levels. The test will be conducted at power levels $>$ 80% of normal rated power. During controlled power reduction, the setpoint will be readjusted prior to going below 20% rated power without the setpoint change, control rod withdrawal will be prohibited until the necessary trip setpoint adjustment is made.

17. This APRM Flow Referenced Scram setting is applicable to two loop operation. For one loop operation this setting becomes $S \leq (0.66W+54\%-0.66\Delta W)(FRP/MFLPD)$ where ΔW = Difference between two-loop and single-loop effective drive flow at the same core flow.

* This specification is in effect only during Operating Cycle 7.

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum no. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action
2	APRM Upscale (Flow Biased)	$s \leq (0.66W+42\%)(FRP/MFLPD)$(11)	6 Inst. Channels	(1)
2	APRM Upscale (Start-up Mode)	$\leq 12\%$	6 Inst. Channels	(1)
2	APRM Downscale	≥ 2.5 indicated on scale	6 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Flow Biased)	$S \leq 0.66W+K$ (8)(12)	2 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Downscale)	≥ 2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (2)	$\geq 2\%$ of full scale	8 Inst. Channels	(1)
3	IRM Detector not in Start-up Position	(7)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq 86.4\%$ of full scale	8 Inst. Channels	(1)
2 (4)	SRM Detector not in Start-up Position	(3)	4 Inst. Channels	(1)
2 (4) (5)	SRM Upscale	$\leq 10^5$ counts/sec	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Water Level	≤ 26.0 gallons per instrument volume	2 Inst. Channels	(9) (10)

NOTES FOR TABLE 3.2-3

1. For the Start-up and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM block need not be operable in run mode, and

TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2-3

11. This is the APRM Rod Block line for two loop operation. For single loop operation this line is $S \leq (0.66W+42\%-0.66\Delta W)(FRP/MFLPD)$.

    ΔW = Difference between two-loop and single loop effective drive flow at the same core flow.

12. This is the RBM Rod Block line for two loop operation. For single loop operation this line is $S \leq (0.66W+K-0.66\Delta W)$ where:

    ΔW = Difference between two-loop and single-loop effective drive flow at the same core flow.

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2-3

11. This is the APRM Rod Block line for two loop operation. For single loop operation this line is $S \leq (0.66W+42\%-0.66\Delta W)(FRP/MFLPD)$.

    ΔW = Difference between two-loop and single loop effective drive flow at the same core flow.

12. This is the RBM Rod Block line for two loop operation. For single loop operation this line is $S \leq (0.66W+K-0.66\Delta W)$ where:

    ΔW = Difference between two-loop and single-loop effective drive flow at the same core flow.

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3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.5-9 through 3.5-11 for two loop operation. For single loop operation these values are reduced by multiplying by 0.84 (see Specification 3.5.K, Reference 1). If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% or rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI, RCIC, or Core Spray System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI, RCIC, and Core Spray shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to insure they are full shall be functionally tested each month.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

3.5 (cont'd)

J. Thermal Hydraulic Stability

1. Whenever the reactor is in the startup or run modes, two Reactor Coolant System recirculation loops shall be in operation, with:

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

   b. Thermal power less than or equal to the limit specified in Figure 3.5-1 (Line A).

   except as specified in Specifications 3.5.J.2 and 3.5.J.3.

2. With two Reactor Coolant System recirculation loops in operation and total core flow less than 45 percent of rated and thermal power greater than the limit specified in Figure 3.5-1 (Line A); or with one Reactor Coolant System loop operating and thermal power greater than the limit specified in Figure 3.5-1 (Line A):

   a. Determine the APRM and LPRM noise levels:

      1. Within 2 hours after reaching steady-state within the regions of Figure 3.5-1 where monitoring is required, and at least once per 8 hours thereafter; and

4.5 (cont'd)

J. Thermal Hydraulic Stability

1. Establish baseline APRM and LPRM neutron flux noise values within 2 hours of entering the region for which monitoring is required unless baselining has been performed since the last refueling outage. Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

3.5 (cont'd)

2. Within 2 hours after completing an increase in thermal power of 5 percent or more of rated thermal power.

b. If the APRM and LPRM neutron flux noise levels are greater than 5 percent and greater than three times their established baseline noise levels, initiate corrective action within 15 minutes to restore the noise levels to within the required limits within 2 hours, by increasing core flow and/or reducing thermal power.

3. If during single-loop operation, core thermal power is greater than the limit defined by line A of Figure 3.5-1, and core flow is less than 39 percent, immediately initiate corrective action to restore core thermal power and/or core flow to within the limits, specified in Figure 3.5-1, by increasing core flow and/or initiating an orderly reduction of core thermal power by inserting control rods.

4. The requirements applicable to single-loop operation in Specifications 1.1.A, 2.1.A, 3.1.A, 3.1.B, 3.2.C and 3.5.H shall be in effect within 8 hours following the removal of one recirculation loop from service, or the reactor shall be placed in the hot shutdown condition.

3.5 (cont'd)

5. During resumption of two-loop operation following a period of single-loop operation, the discharge valve of the low-speed pump shall not be opened unless the speed of the faster pump is less than 50 percent of it's rated speed.

6. With no Reactor Coolant System Recirculation loop in service, the reactor shall be placed in Hot Shutdown within 12 hours.

3.5 BASES (cont'd)

requirements for the emergency diesel generators.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, RCIC, and HPCI are not filled, a water hammer can develop in this piping when the pump(s) are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this technical specification requires the discharge lines to be filled whenever the system is required to be operable. If a discharge pipe is not filled, the pumps the supply that line must be assumed to be inoperable for technical specification purposes. However, if a water hammer were to occur, the system would still perform its design function.

H. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50 Appendix K.

The peak cladding temperature 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 only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.5-9 through 3.5-11. The reduction factor for single loop operation for the above curves is 0.84. The derivation of this factor can be found in Specification 3.5.K, Reference 1.

I. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation.

The LHGR shall be checked daily during reactor operation at $\geq$ 25% rated thermal power to determine if fuel burnup, or control rod movement, has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the ratio of local LHGR to average LHGR would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5 BASES (cont'd)

J. Thermal Hydraulic Stability

Operation in certain regions of the power vs. flow curve have been identified as having a high potential for thermal hydraulic instability (Figure 3.5-1). These regions are located in the high power/low flow area of the curve and can be encountered during startup, shutdown, rod sequence exchange or recirculation pump trip. Operation in these regions is associated with higher than normal neutron flux noise levels. Increased awareness of LPRM and APRM signal noise when operating in these regions will identify instability and allow operator action to correct the problem. The neutron flux noise level, thermal power and core flow limits are prescribed in accordance with the recommendations of General Electric Service Information Letter No. 380, Revision 1, "BWR Core Thermal Hydraulic Stability", dated February 10, 1984.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent of its rated speed provides assurance when going from one to two pump operation that excessive vibration of the jet pump risers will not occur.

K. References

1. "FitzPatrick Nuclear Power Plant Single-Loop Operation", NEDO-24281, August 1980.

Figure 3.5-1
Thermal Power and Core Flow Limits of Specifications 3.5.J.1, 3.5.J.2 and 3.5.J.3

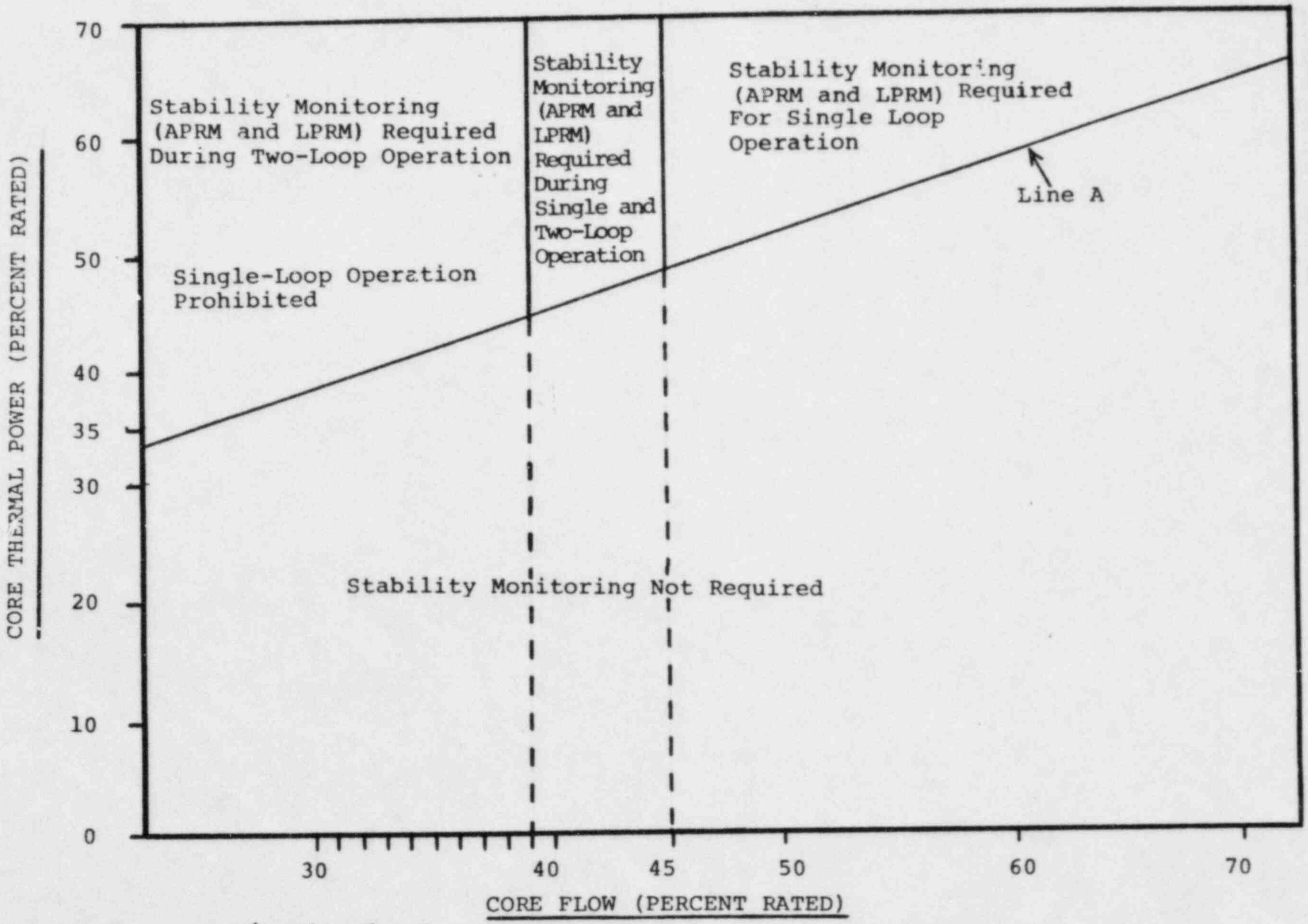

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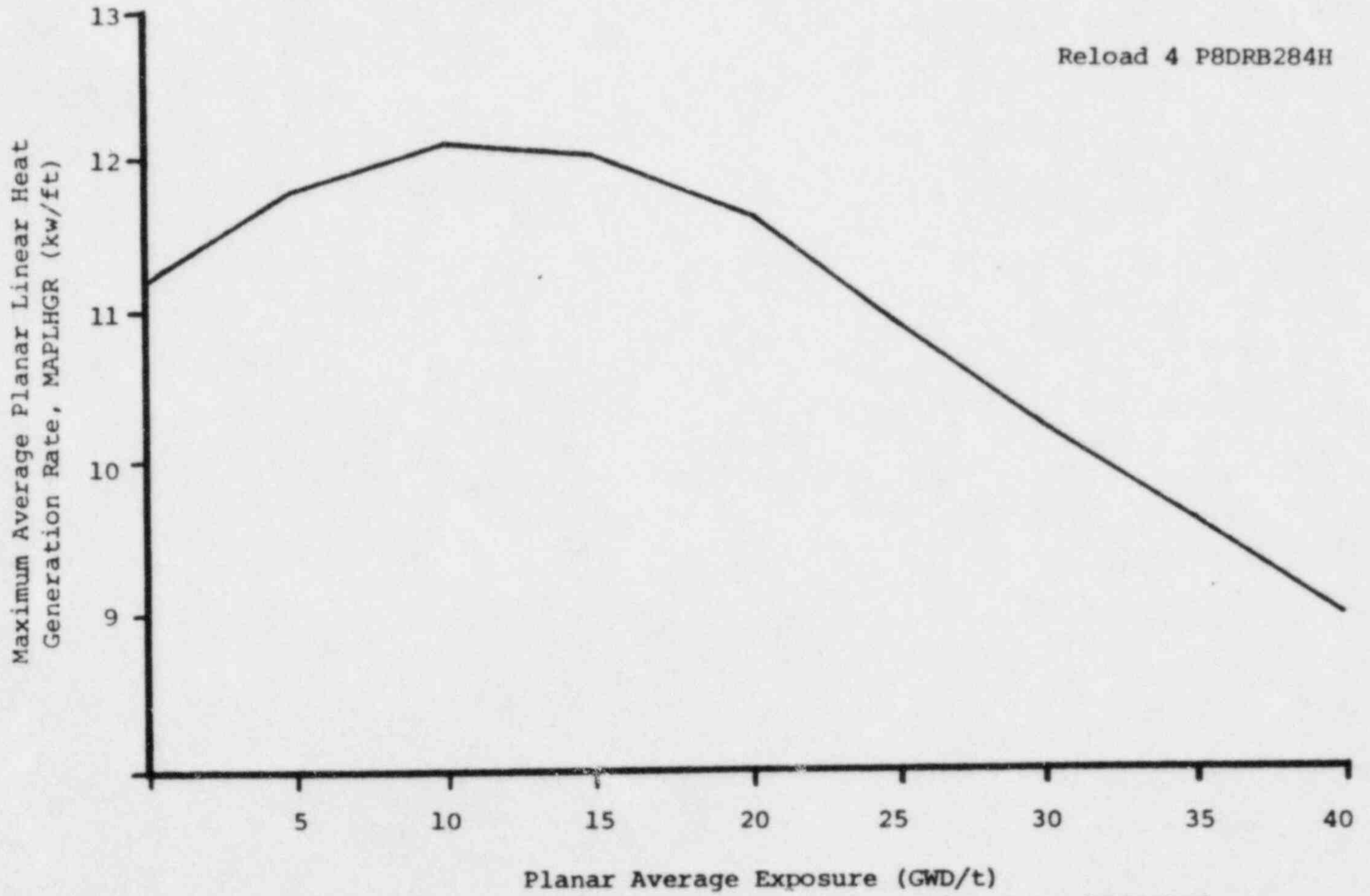
Figure 3.5-9

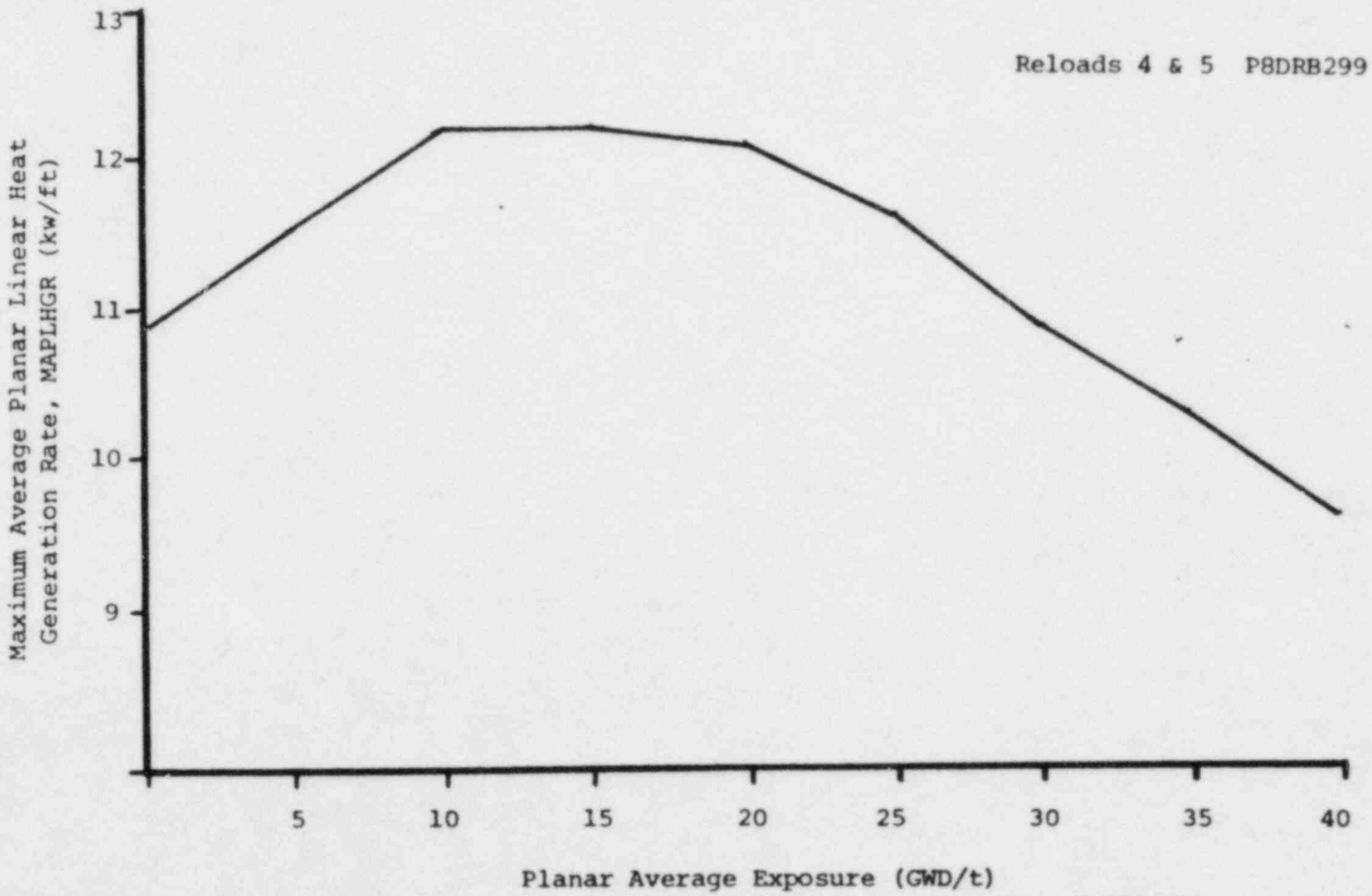


Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Planar Average Exposure

For single-loop operation, these MAPLHGR values are multiplied by 0.84.

Reference: NEDO-21662-2 (As amended August 1981)

Figure 3.5-10

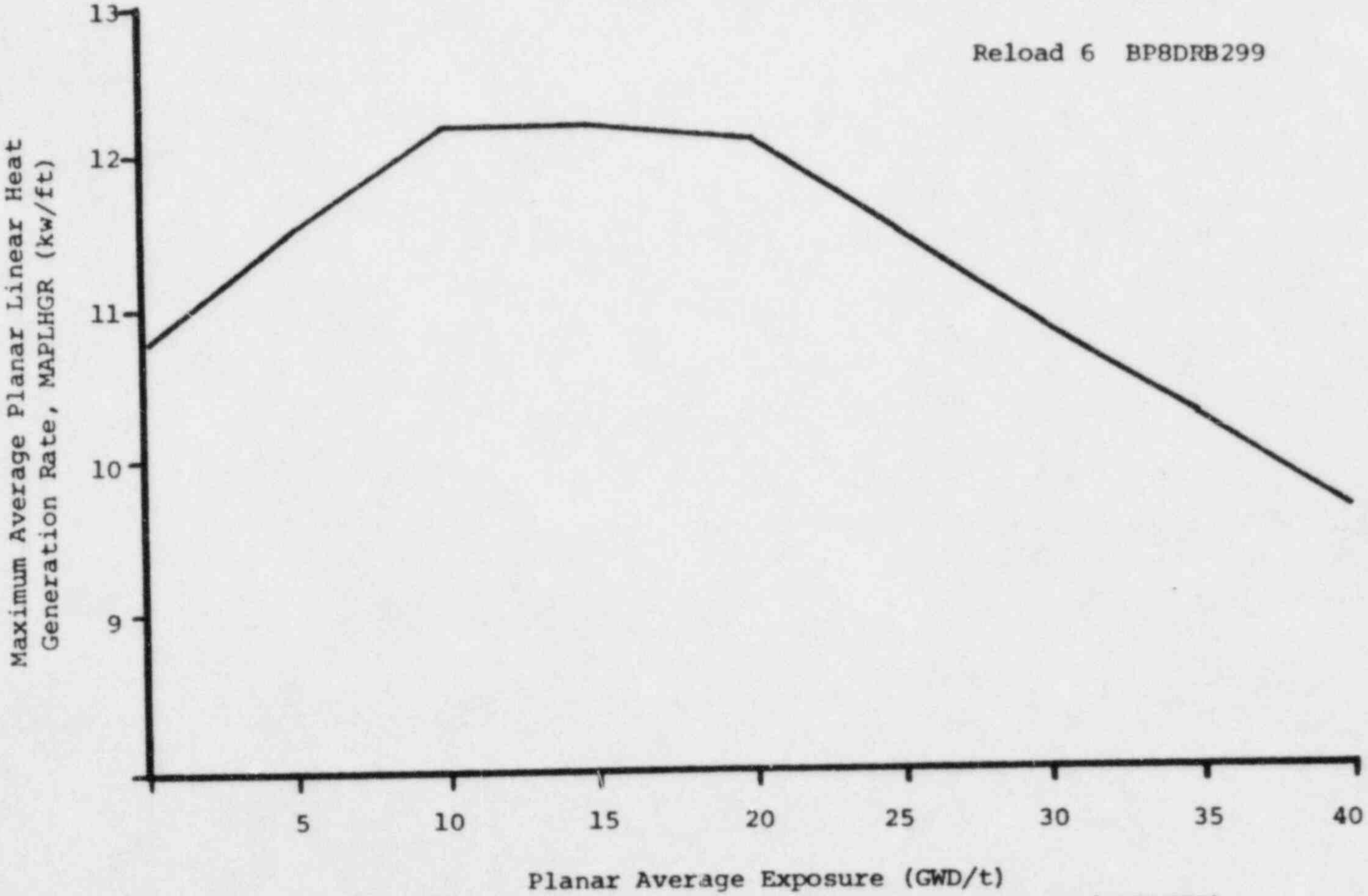


Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)
Versus Planar Average Exposure

For single-loop operation, these MAPLHGR values are multiplied by 0.84.

Reference: NEDO-21662-2
(As amended August 1981)

Figure 3.5-11



Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Planar Average Exposure

For single-loop operation, these MAPLHGR values are multiplied by 0.84.

Reference: NEDO-21662-2
(As amended December 1984)

4.6 (cont'd)

1. The two recirculation loops have a flow imbalance of 15 percent or more when the pumps are operated at the same speed.

2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10 percent.

3. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the average of all jet pump differential pressures by more than 10 percent.

A. Whenever the reactor is in the startup/hot standby or run modes, and there is one loop recirculation flow, jet pump operability shall be verified as follows.

a. Baseline readings will be taken and operating characteristics for the following parameters established:

1. Jet Pump Loop Flow and Recirculation Pump Speed for the operating loop.

2. Individual Jet Pump percent differential pressures for all jet pumps.

b. Initially, and daily thereafter, jet pump operability will be verified by assuring that the following do not occur simultaneously:

4.6 (cont'd)

1. The ratio of jet pump loop flow to recirculation pump speed for the operating loop does not vary from the initially established value by more than 10 percent.
2. The ratio of individual jet pump percent differential pressure to the loop's average jet pump percent differential pressure does not vary from the initially established value by more than 20 percent.

LIMITING CONDITION FOR OPERATION

3.6 (cont'd)

3.6.I Shock Suppressors (Snubbers)

Applicability

Applies to the operational status of the shock suppressors (snubbers).

Objective

To assure the capability of the snubbers to:

Prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, and

Allow normal thermal motion during startup and shutdown.

Specification

1. During all modes of operation except Cold Shutdown and Refueling, all snubbers which are required to protect the primary coolant system or any other safety related system or component shall be operable. During Cold Shutdown or Refueling mode of operation, only those snubbers shall be operable which are on systems that are required to be operable in these modes.

2. With one or more snubbers inoperable, within 72 hours during normal operation, or within 7 days during Cold Shutdown or Refueling mode of operation for systems

SURVEILLANCE REQUIREMENT

4.6 (cont'd)

4.6.I Shock Suppressors (Snubbers)

Applicability

Applies to the periodic testing requirement for the shock suppressors (snubbers).

Objective

To assure the capability of the snubbers to perform their intended functions.

Specification

Each snubber shall be demonstrated operable by performance of the following augmented inservice inspection program.

1. Snubbers shall be visually inspected in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period *#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

* The inspection interval may not be extended more than one step at the time.

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3.6 (cont'd)

which are required to be operable in these modes, complete one of the following:

a. replace or restore the inoperable snubber(s) to operable status or,

b. declare the supported system inoperable and follow the appropriate limiting condition for operation statement for that system or,

c. perform an engineering evaluation to demonstrate the inoperable snubber is unnecessary to assure operability of the system or to meet the design criteria of the system, and remove the snubber from the system.

3. With one or more snubbers found inoperable, within 72 hours perform a visual inspection of the supported component(s) associated with the inoperable snubber(s) and document the results. For all modes of operation except Cold Shutdown and Refueling, within 14 days complete an engineering evaluation as per Specification 4.6.I.6 to ensure that the inoperable snubber(s) has not adversely affected the supported component(s). For Cold Shutdown or Refueling mode, this evaluation shall be completed within 30 days.

4.6 (cont'd)

\# The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

2. Visual inspection shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movements can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.6.I.7 or 4.6.I.8, as applicable. Hydraulic snubbers which have lost sufficient fluid to potentially cause uncovering of the fluid reservoir-to-snubber valve assembly port or bottoming of the fluid reservoir piston with the snubber

3.6 (cont'd)

4.6 (cont'd)

in the fully extended position shall be functionally tested to determine operability.

3. Once each operating cycle, 10% of each type of snubbers shall be functionally tested for operability, either in place or in a bench test. For each unit and subsequent unit that does not meet the requirements of 4.6.I.7 or 4.6.I.8, an additional 10% of that type of snubber shall be functionally tested until no more failures are found, or all units have been tested.

4. The representative sample selected for functionally testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

   a. The first snubber away from reactor vessel nozzle.

   b. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.).

   c. Snubbers within 10 feet of the discharge from a safety relief valve.

3.6 (cont'd)

4.6 (cont'd)

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

5. If any snubber selected for functional testing either fails to lockup or fails to move, i.e. is frozen in place, the cause will be evaluated and if due to manufacturer or design deficiency, snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

6. For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported components remain capable of meeting the designed service requirements.

3.6 (cont'd)

4.6 (cont'd)

7. The hydraulic snubber functional test shall verify that;

   a. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

   b. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

8. The mechanical snubber functional test shall verify that:

   a. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force. Drag force shall not have increased more than 50% since the last functional test.

   b. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3.6 (cont'd)

4.6 (cont'd)

c. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

9. Snubber Service Life Monitoring

A record of the service life of each snubber, whose failure could adversely affect the primary coolant or other safety-related system, the date at which the designated service life commences, and the installation and maintenance records on which the designated service life is based shall be maintained as required by specification 6.10.B.13.

At least once per operating cycle, the installation and maintenance records for each snubber, whose failure could adversely affect the primary coolant or other safety related system, shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next shceduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next schedule service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

3.6 and 4.6 BASES (cont'd)

would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle-riser system failure.

Surveillance tests are performed to verify jet pump operability. Significant changes in either: (1) the relationship between loop flow and recirculation pump speed, or (2) individual jet pump differential pressure compared to average jet pump differential pressure, are used to detect degraded jet pump performance.

3.6 and 4.6 BASES (cont'd)

H. (DELETED)

I. Shock Suppressors

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation. Snubbers excluded from this inspection program are those installed on non-safety related system and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system. Because the snubber protection is required only during low probability events, a period of 72 hours (for normal operation) or 7 days (for cold shutdown or refueling mode of operation) is allowed for repairs or replacement of the snubber prior to taking any other action. Following the 72 hour (or 7 day) period, the supported system must be declared inoperable and the Limiting Condition of Operation statement for the supported system followed. As an alternative to snubber repair or replacement an engineering evaluation may be performed: to demonstrate that the inoperable snubber is unnecessary to assure operability of the system or to meet the design criteria of the system; and, to remove the snubber from the system. With one or more snubbers found inoperable, within 72 hours a visual inspection shall be performed on the supported component(s) associated with the inoperable snubber(s) and the results shall be documented. For all modes of operation except Cold Shutdown and Refueling, within 14 days an engineering evaluation shall be performed to ensure tht the inoperable snubber(s) has not adversely affected the supported component(s). For Cold Shutdown or refueling mode, this evaluation shall be completed within 30 days. A period of 7 days has been selected for repair or replacement of the inoperable snubber during cold shutdown or refueling mode of operation becuase in these modes the relative probability of structural damage to the piping systems would be lower due to lower values of total stresses on the piping systems. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures.

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