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

October 3, 1986

Posted
Amdt. 31
to NPF-22
(See correction letter
of 11-28-86)

Docket No. 50-388

Mr. Harold W. Keiser
Vice President
Nuclear Operations
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, Pennsylvania 18101

Dear Mr. Keiser:

Subject: Issuance of Amendment No. 31 to Facility Operating License No. NPF-22 - Susquehanna Steam Electric Station, Unit 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Unit 2. This amendment is in response to your letter dated April 30, 1986, as supplemented on June 19, July 25, September 16 and 25, 1986.

This amendment revises the Susquehanna Unit 2 Technical Specifications to include operational control for Cycle 2 operation.

A copy of the related safety evaluation supporting Amendment No. 31 to Facility Operating License No. NPF-22 is enclosed.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosures:
1. Amendment No. 31 to NPF-22
2. Safety Evaluation

cc w/enclosures:
See next page

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

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
License No. NPF-22

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

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated April 30, 1986, as supplemented on June 19, July 25, September 16 and 25, 1986, 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 the Facility Operating License No. NPF-22 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. 31 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon startup following the Unit 2 first refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
BWR Project Directorate No. 3
Division of BWR Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: October 3, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 31

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

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 area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
i	i
ii	ii (overleaf)
iii	iii (overleaf)
iv	iv
xxi	xxi (overleaf)
xxii	xxii
xxiii	xxiii
xxiv	xxiv
xxv	xxv
xxvi	xxvi
xxvii	xxvii
1-1	1-1
1-2	1-2 (overleaf)
1-3	1-3
1-4	1-4 (overleaf)
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3 (overleaf)
B 2-4	B 2-4
3/4 1-1	3/4 1-1 (overleaf)
3/4 1-2	3/4 1-2
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3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
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REMOVE	INSERT
3/4 2-7	3/4 2-7
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3/4 2-8a	
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
------	3/4 2-10a
------	3/4 2-10b
3/4 3-39	3/4 3-39 (overleaf)
3/4 3-40	3/4 3-40
3/4 3-53	3/4 3-53 (overleaf)
3/4 3-54	3/4 3-54
3/4 4-1b	3/4 4-1b (overleaf)
3/4 4-1c	3/4 4-1c
3/4 7-29	3/4 7-29 (overleaf)
3/4 7-30	3/4 7-30
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 1-4	B 3/4 1-4
-----	B 3/4 1-5
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	------
B 3/4 2-5	------
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2 (overleaf)
B 3/4 7-3	B 3/4 7-3 (overleaf)
B 3/4 7-4	B 3/4 7-4
5-5	5-5 (overleaf)
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1.0 DEFINITIONS

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

ACTION

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

AVERAGE EXPOSURE

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

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

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

'HANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

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

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

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Correction letter of 11-28-86

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CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation, or movement of fuel, sources, or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, TIPs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

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

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.12 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete supression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

This total system response time consists of two components, the instrumentation response time and the breaker arc suppression time. These times may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.13 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the LHGR specified in Section 3.2.2 for that bundle type.

FRACTION OF RATED THERMAL POWER

1.14 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.16 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.17 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.18 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.19 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.20 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.21 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, ie., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.22 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

1.23 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.24 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core for each class of fuel.

OFFSITE DOSE CALCULATION MANUAL

1.25 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.26 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.27 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

2.1 SAFETY LIMITS

BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specification 2.1.2 for both GE and Exxon fuel. MCPR greater than the specified limit 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 the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling (ref. XN-NF-524(A)).

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the XN-3 correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

BASES

2.1.2 THERMAL POWER, High Pressure and High Flow

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).

The Safety Limit MCPR assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR = 1.00) and the Safety Limit MCPR is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. XN-NF-524 describes the methodology used in determining the Safety Limit MCPR.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. The assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that during sustained operation at the Safety Limit MCPR there would be no transition boiling in the core. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicates that LWR fuel can survive for an extended period of time in an environment of boiling transition.

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity difference greater than 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

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

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity difference between the monitored core k_{eff} and the predicted core k_{eff} shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 700 MWD/MT of core exposure during POWER OPERATION.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

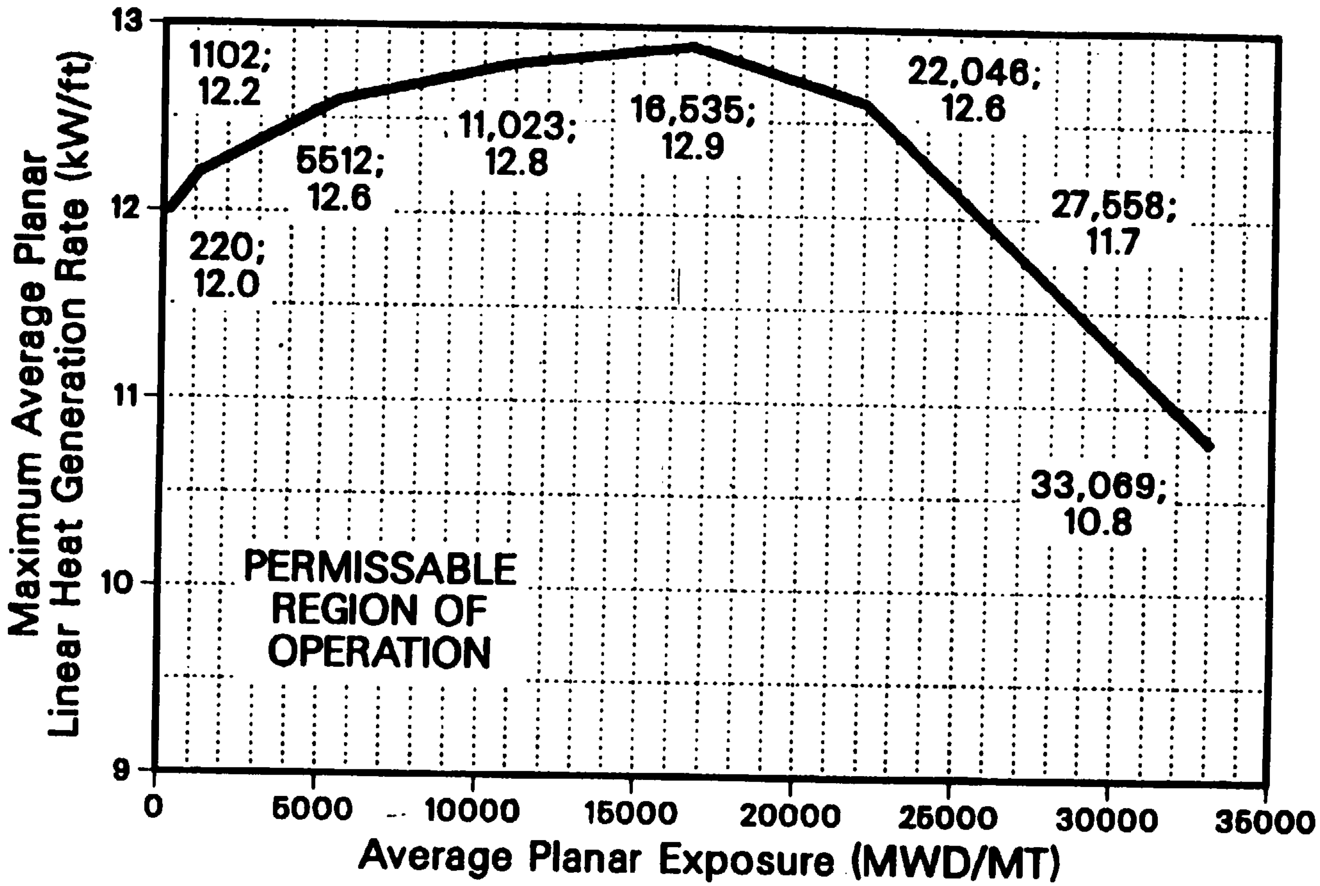
a. At least once per 24 hours,

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

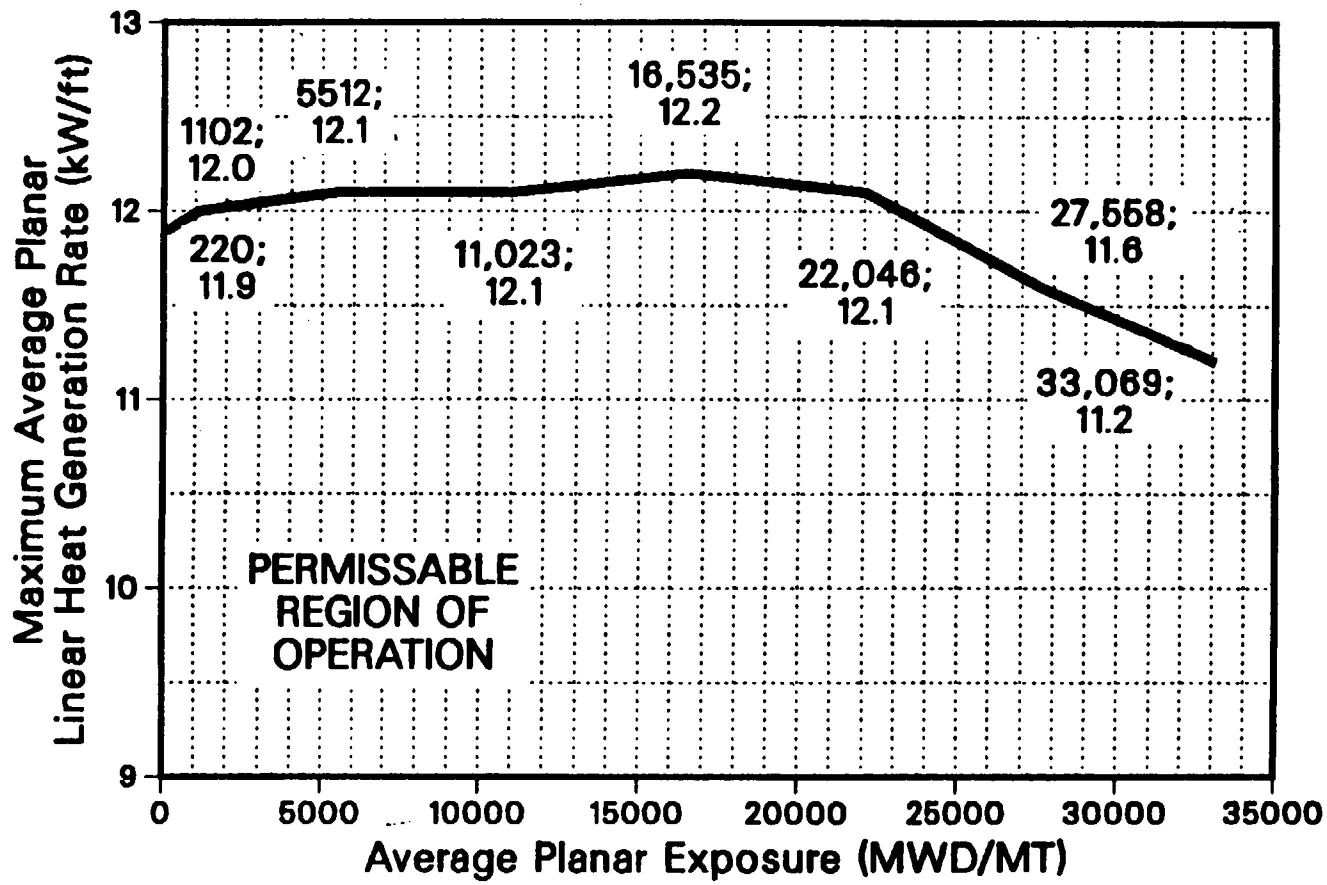
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

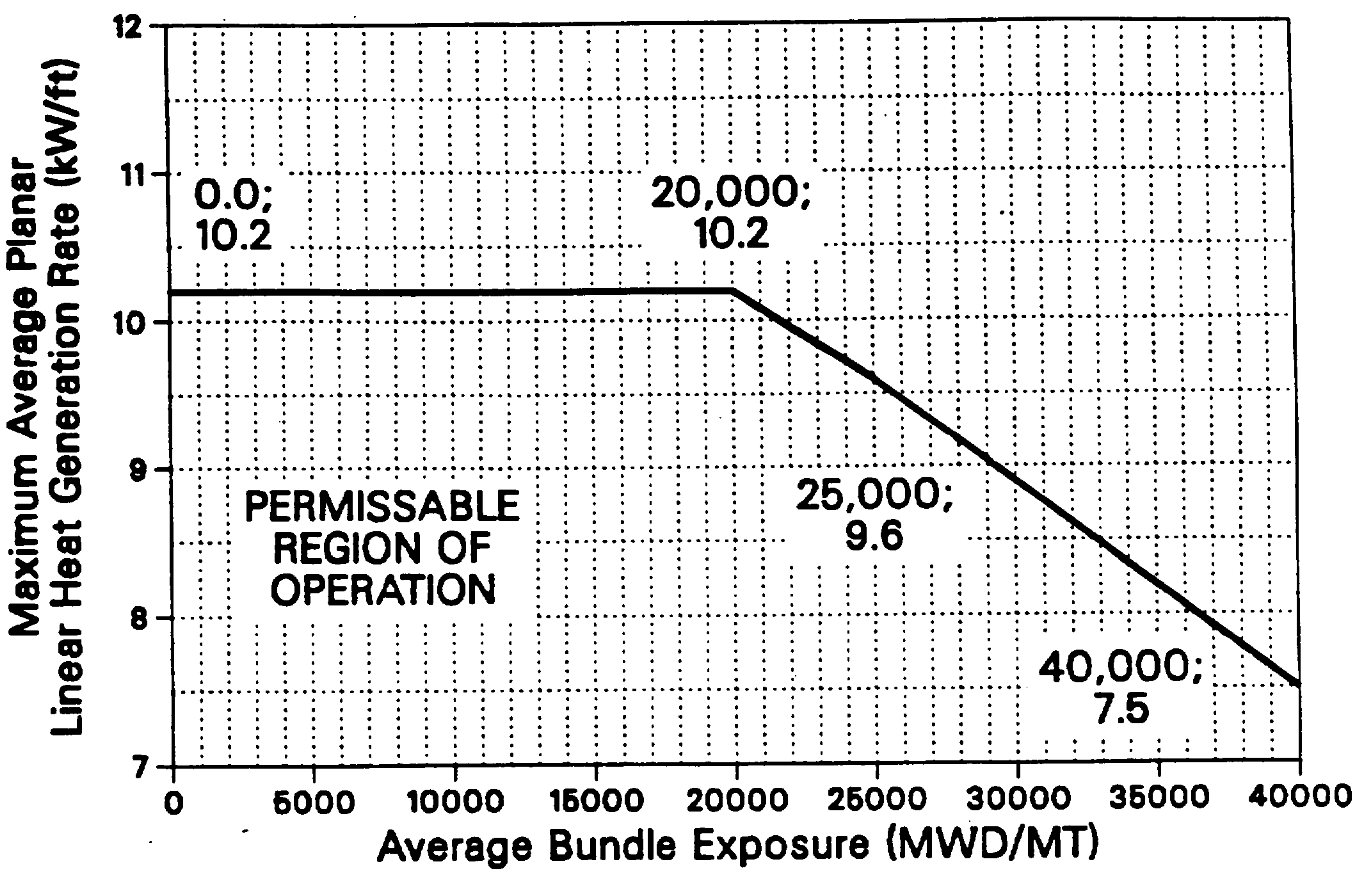
*See Specification 3.4.1.1.2.a for single loop operation requirements.



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE
GE FUEL TYPES 8CR183 (1.83% ENRICHED)
FIGURE 3.2.1-1



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
GE FUEL TYPES 8CR233 (2.33% ENRICHED)
FIGURE 3.2.1-2



MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE BUNDLE EXPOSURE
EXXON 9X9 FUEL
FIGURE 3.2.1-3

POWER DISTRIBUTION LIMITS

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#
$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$

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 100 million lbs/hr,
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. Where:

a. The FRACTION OF LIMITING POWER DENSITY (FLPD) for GE fuel is the actual LINEAR HEAT GENERATION RATE (LHGR) divided by 13.4 per Specification 3.2.4.1, and

b. The FLPD for Exxon fuel is the actual LHGR divided by the LINEAR HEAT GENERATION RATE from Figure 3.2.2-1.

T is always less than or equal to 1.0.

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

ACTION:

With the APRM flow biased simulated thermal power-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 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:

---

*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, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

#See Specification 3.4.1.1.2.a for single loop operation requirements.

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

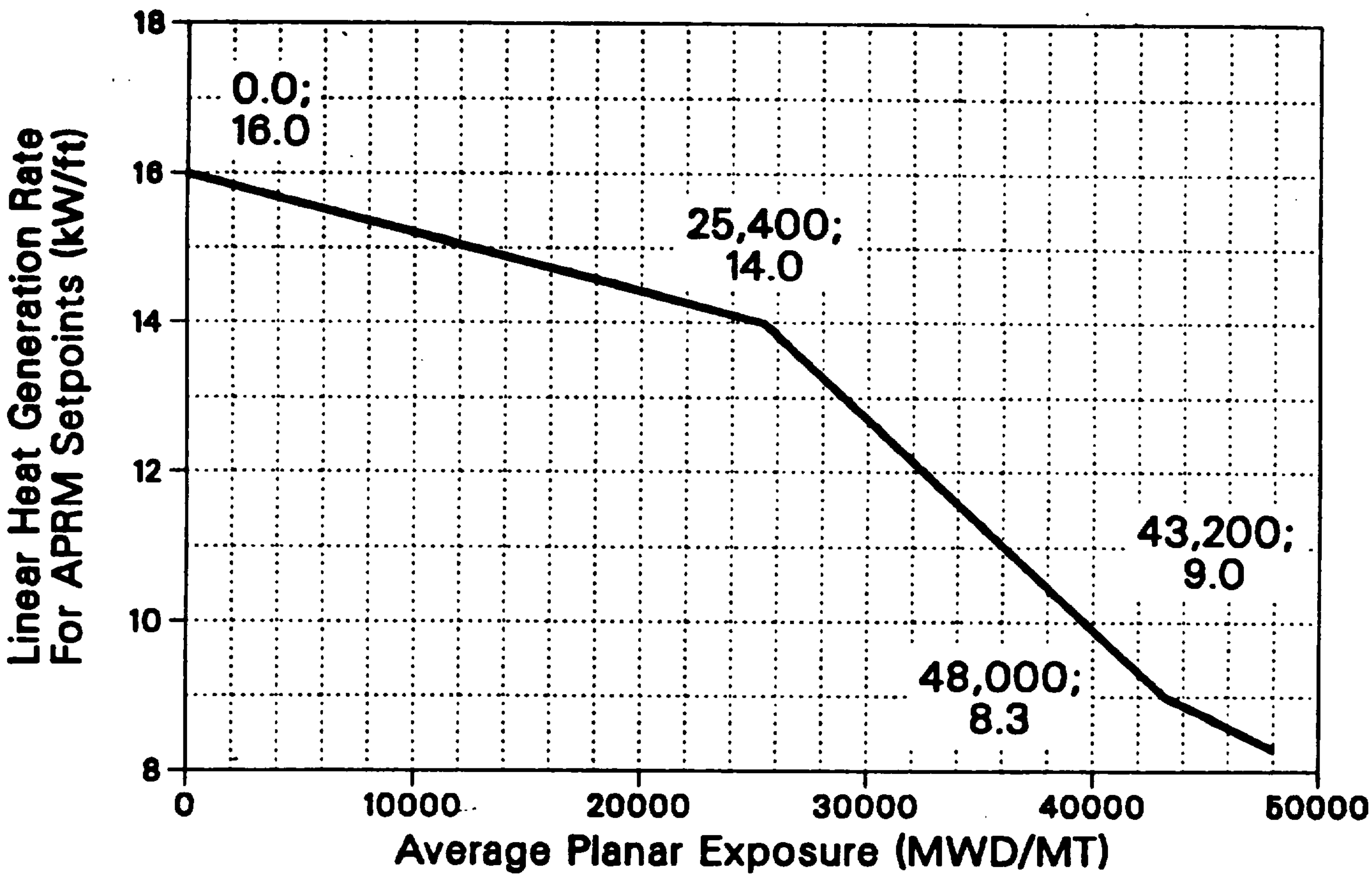
4.2.2 (Continued)

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.

d. The provisions of Specification 4.0.4 are not applicable.



LINEAR HEAT GENERATION RATE FOR APRM SETPOINTS
VERSUS AVERAGE PLANAR EXPOSURE
EXXON FUEL
FIGURE 3.2.2-1

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be greater than or equal to the greater of the two values determined from Figure 3.2.3-1 and Figure 3.2.3-2.

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

ACTION:

With MCPR less than the applicable MCPR limit determined above, 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% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

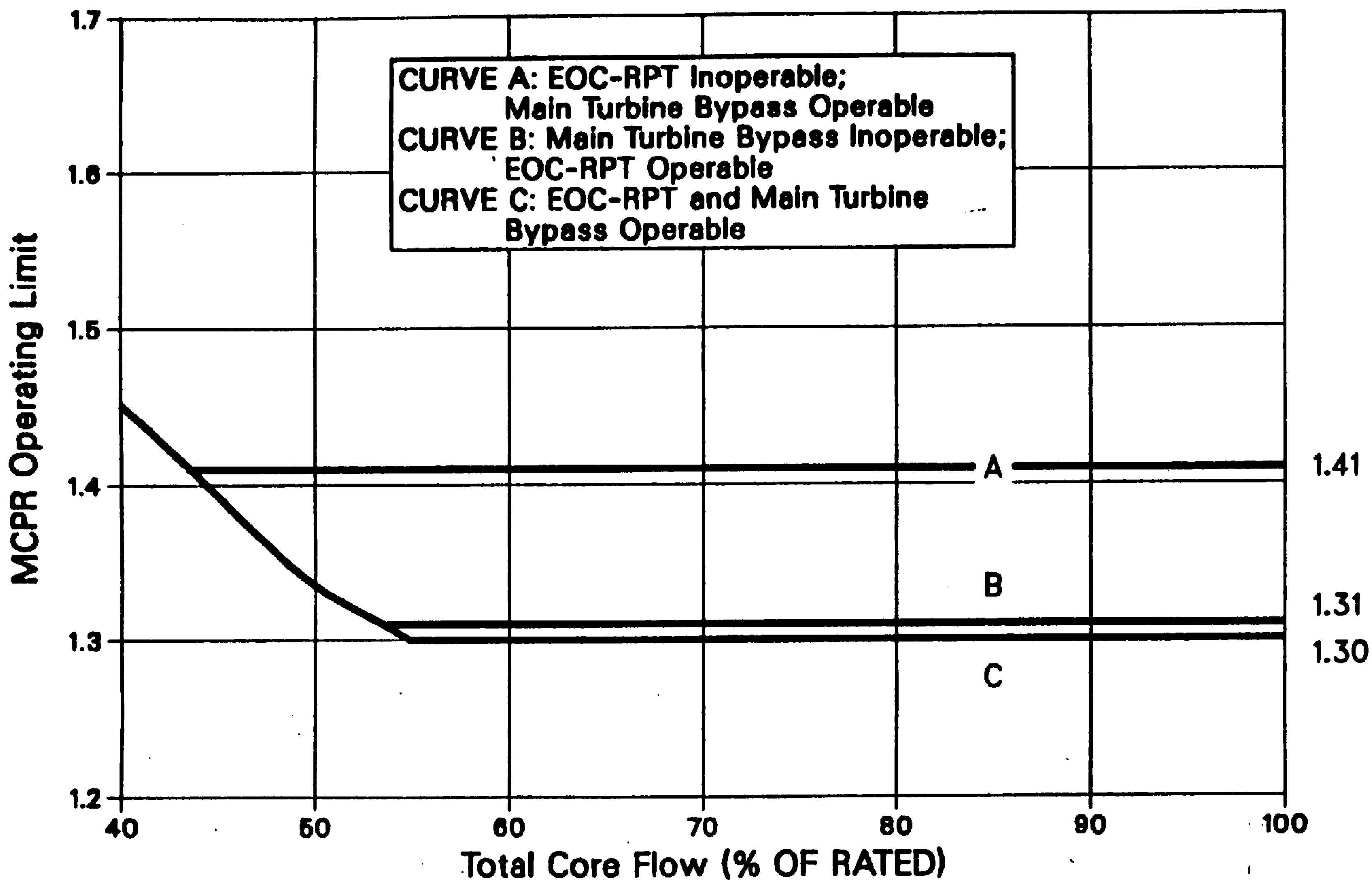
4.2.3.1 MCPR shall be determined to be greater than or equal to the applicable MCPR limit determined from Figure 3.2.3-1 and Figure 3.2.3-2:

a. At least once per 24 hours,

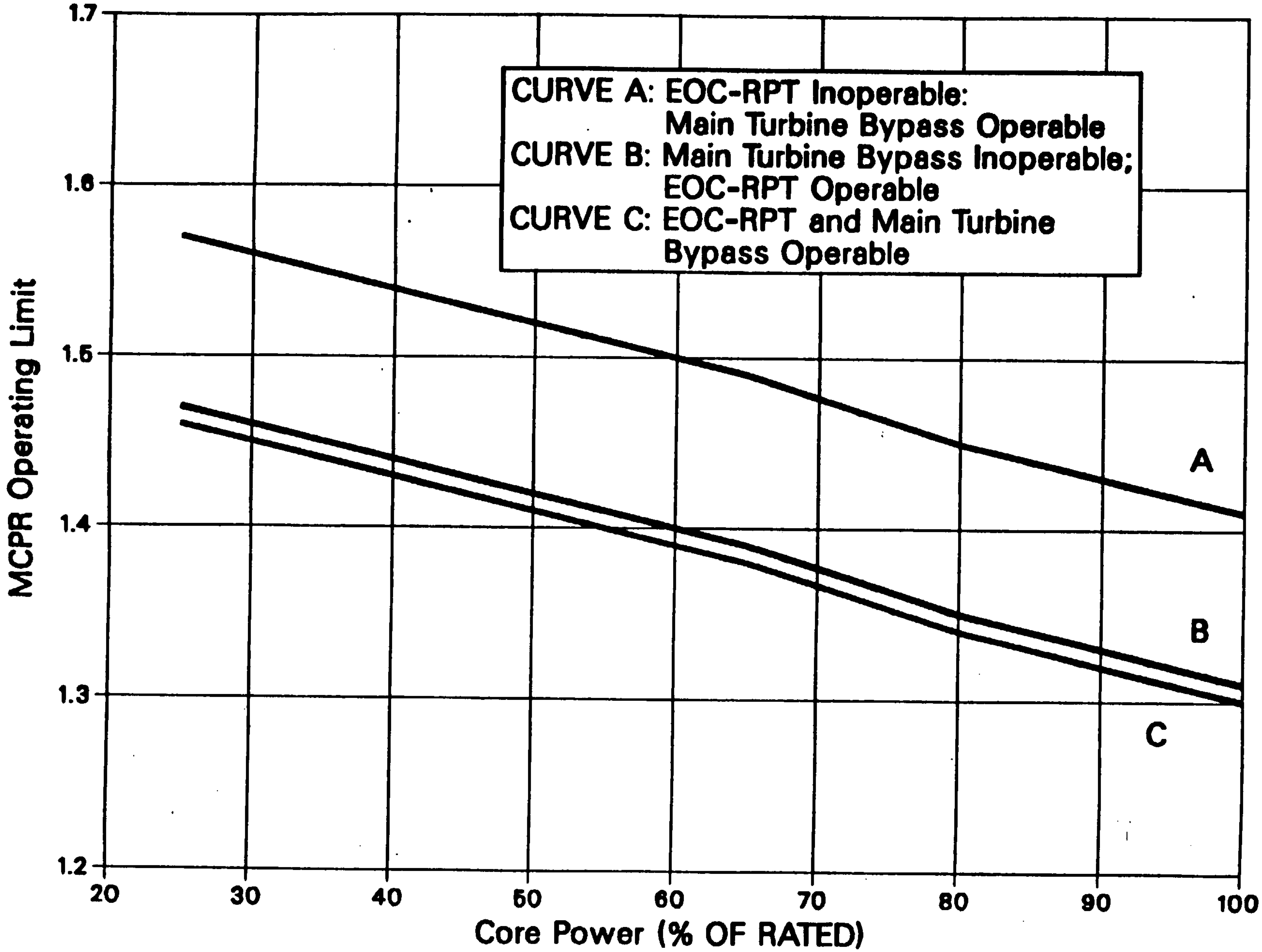
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

d. The provisions of Specification 4.0.4 are not applicable.



FLOW DEPENDENT MCPR OPERATING LIMIT
FIGURE 3.2.3-1



REDUCED POWER MCPR OPERATING LIMIT

Figure 3.2.3-2

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

GE FUEL

LIMITING CONDITION FOR OPERATION

3.2.4.1 The LINEAR HEAT GENERATION RATE (LHGR) for GE fuel shall not exceed 13.4 kw/ft.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 LHGRs for GE fuel shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

3/4.2.4 LINEAR HEAT GENERATION RATE

ENC FUEL

LIMITING CONDITION FOR OPERATION

3.2.4.2 The LINEAR HEAT GENERATION RATE (LHGR) for ENC fuel shall not exceed the LHGR limit determined from Figure 3.2.4.2-1.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

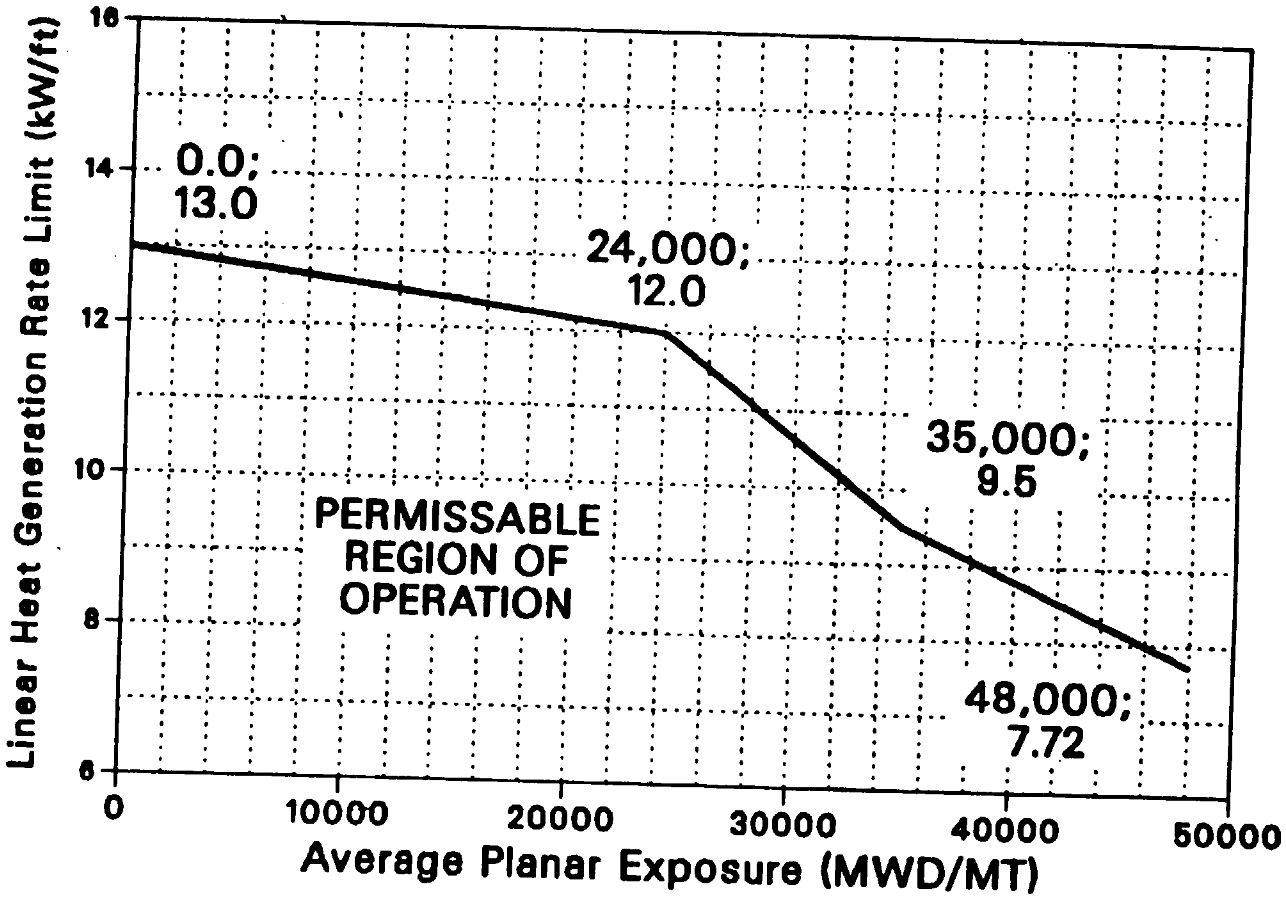
4.2.4.2 LHGRs for ENC fuel shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.



LINEAR HEAT GENERATION RATE (LHGR) LIMIT
VERSUS AVERAGE PLANAR EXPOSURE
EXXON 9X9 FUEL
FIGURE 3.2.4.2-1

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	M	R
2. Reactor Vessel Steam Dome Pressure - High	NA	M	Q

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within one hour.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or evaluate MCPR to be equal to or greater than the applicable MCPR limit without EOC-RPT within 1 hour* or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or evaluate MCPR to be equal to or greater than the applicable MCPR limit without EOC-RPT within 1 hour* or take the ACTION required by Specification 3.2.3.

*If MCPR is evaluated to be equal to or greater than the applicable MCPR limit without EOC-RPT within 1 hour, operation may continue and the provisions of Specification 3.0.4 are not applicable.

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 1 hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 1 hour.

NOTES

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

*** Not required when eight of fewer fuel assemblies (adjacent to the SRMs) are in the core.

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

(b) This function shall be automatically bypassed if detector count rate is $\geq$ 100 cps or the IRM channels are on range 3 or higher.

(c) This function is automatically bypassed when the associated IRM channels are on range 8 or higher.

(d) This function is automatically bypassed when the IRM channels are on range 3 or higher.

(e) This function is automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

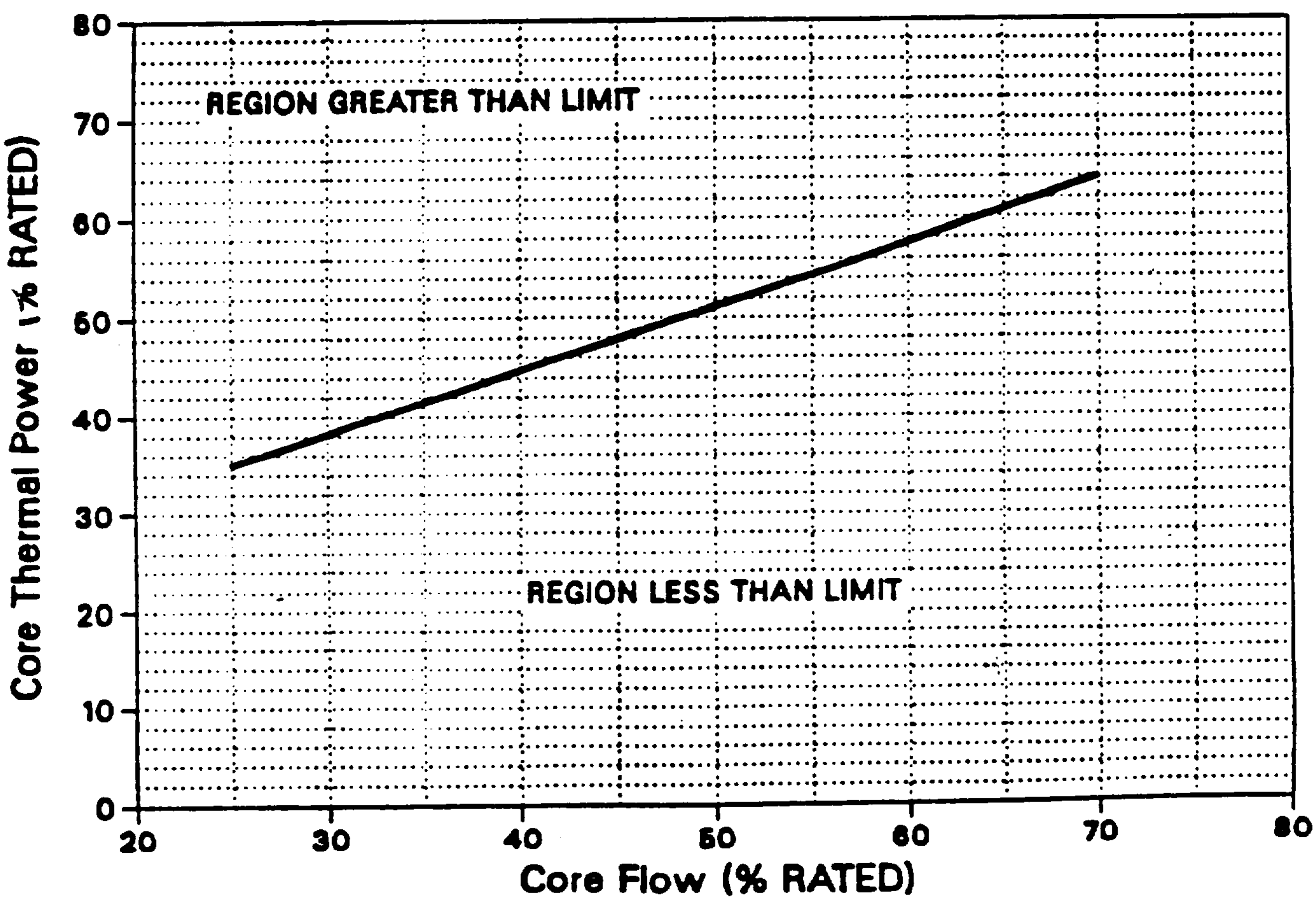
TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale##	$\leq$ 0.66 W + 42%	$\leq$ 0.66 W + 45%
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5/125 divisions of full scale	$\geq$ 3/125 of divisions full scale
2. APRM		
a. Flow Biased Neutron Flux - Upscale	$\leq$ 0.58 W + 50%*	$\leq$ 0.58 W + 53%*
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 2 \times 10^5$ cps	$\leq 4 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 0.7 cps**	$\geq$ 0.5 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 divisions of full scale	$\leq$ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 divisions of full scale	$\geq$ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level - High	$\leq$ 44 gallons	$\leq$ 44 gallons
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 108/125 divisions of full scale	$\leq$ 111/125 divisions of full scale
b. Inoperative	NA	NA
c. Comparator	$\leq$ 10% flow deviation	$\leq$ 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.

**Provided signal-to-noise ratio is $\geq$ 2. Otherwise, 3 cps as trip setpoint and 2.8 cps for allowable value.

##See Specification 3.4.1.1.2.a for single loop operation requirements.

Figure 3.4.1.1.1-1
THERMAL POWER LIMITATIONS



RECIRCULATION LOOPS - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1.2 One reactor coolant recirculation loop shall be in operation with the pump speed $\leq$ 90% of the rated pump speed, and

a. the following revised specification limits shall be followed:

1. Specification 2.1.2: the MCPR Safety Limit shall be increased to 1.07.

2. Table 2.2.1-1: the APRM Flow-Biased Scram Trip Setpoints shall be as follows:

Trip Setpoint	Allowable Value
$\leq$ 0.58W + 55%	$\leq$ 0.58W + 58%.

3. Specification 3.2.1: The MAPLHGR limits shall be the limits specified in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3, multiplied by 0.0.

4. Specification 3.2.2: the APRM Setpoints shall be as follows:

Trip Setpoint	Allowable Value
$S \leq (0.58W + 55\%)T$	$S \leq (0.58W + 58\%)T$
$S_{RB} \leq (0.58W + 46\%)T$	$S_{RB} \leq (0.58W + 49\%)T$

5. Table 3.3.6-2: the RBM/APRM Control Rod Block Setpoints shall be as follows:

a. RBM - Upscale

Trip Setpoint	Allowable Value
$\leq$ 0.66W + 37%	$\leq$ 0.66W + 40%

5.a.1 and 5.a.2 shall be used in conjunction with the MCPR limits specified in Figures 3.2.3-1a and 3.2.3-1b, respectively.

b. APRM-Flow Biased

Trip Setpoint	Allowable Value
$\leq$ 0.58W + 46%	$\leq$ 0.58W + 49%

b. APRM and LPRM*** neutron flux noise levels shall be less than three times their established baseline levels when THERMAL POWER is greater than the limit specified in Figure 3/4.1.1.1-1.

c. Total core flow shall be greater than or equal to 42 million lbs/hr when THERMAL POWER is greater than the limit specified in Figure 3.4.1.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*, except during two loop operation.#

ACTION:

a. With no reactor coolant system recirculation loops in operation, take the ACTION required by Specification 3.4.1.1.1.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.2 Each of the above required fire doors shall be verified OPERABLE by:

a. Verifying the position of each closed fire door at least once per 24 hours.

b. Verifying that doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours.

c. Verifying the position of each locked closed fire door at least once per 7 days.

d. Verifying the OPERABILITY of the fire door supervision system by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

e. Inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months.

3/4.7.8 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 2 hours or evaluate MCPR to be equal to or greater than the applicable MCPR limit without bypass within 1 hour* or take the ACTION required by Specification 3.2.3.

SURVEILLANCE REQUIREMENTS

4.7.8 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. 18 months by:

    1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

    2. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME to be less than or equal to 0.30 second.

*If MCPR is evaluated to be equal to or greater than the applicable MCPR limit without bypass within 1 hour, operation may continue and the provisions of Specification 3.0.4 are not applicable.

BASES

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta k/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the beginning of cycle shutdown margin minus the minimum shutdown margin in the cycle, where shutdown margin is a positive number. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement is small, a careful check on actual reactor conditions compared to the predicted conditions is necessary. Any changes in reactivity from that of the predicted (predicted core k_{eff}) can be determined from the core monitoring system (monitored core k_{eff}). In the absence of any deviation in plant operating conditions or reactivity anomaly, these values should be essentially equal since the calculational methodologies are consistent. The predicted core k_{eff} is calculated by a 3D core simulation code as a function of cycle exposure. This is performed for projected or anticipated reactor operating states/conditions throughout the cycle and is usually done prior to cycle operation. The monitored core k_{eff} is the k_{eff} as calculated by the core monitoring system for actual plant conditions.

REACTIVITY ANOMALIES (Continued)

Since the comparisons are easily done, frequent checks are not an imposition on normal operation. A 1% deviation in reactivity from that of the predicted is larger than expected for normal operation, and therefore should be thoroughly evaluated. A deviation as large as 1% would not exceed the design conditions of the reactor.

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

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

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the limit specified in Specification 2.1.2 during the core wide transient analyzed in the cycle specific transient analysis report. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the limit specified in Specification 2.1.2. The occurrence of scram times longer then those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a

CONTROL RODS (Continued)

reactor scram and will isolate the reactor coolant system from the containment when required.

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

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

CONTROL ROD PROGRAM CONTROLS (Continued)

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

Parametric Control Rod Drop Accident analyses have shown that for a wide range of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient, effective delayed neutron fraction, and maximum four-bundle local peaking factor are compared with the inputs to the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the 280 cal/gm design limit to demonstrate compliance for each operating cycle. If cycle-specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle-specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in XN-NF-80-19 Volume 1.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that none of the withdrawn control rods can be inserted. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 90 to 120 minutes. A minimum quantity of 4587 gallons of sodium pentaborate solution containing a minimum of 5500 lbs. of sodium pentaborate is required to meet this shutdown requirement. There is an additional allowance of 165 ppm in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel. The temperature requirement for the sodium penetrate solution is necessary to ensure that the sodium penetaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

STANDBY LIQUID CONTROL SYSTEM (Continued)

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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 50.46.

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 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 Exxon 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, 3.2.1-2, and 3.2.1-3.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 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 50. These models are described in Reference 1 or XN-NF-80-19, Volumes 2, 2A, 2B and 2C.

3/4.2.2 APRM SETPOINTS

The flow biased simulated thermal power-upscale scram setting and flow biased simulated thermal power-upscale control rod block functions of the APRM instruments limit plant operations to the region covered by the transient and accident analyses. In addition, the APRM setpoints must be adjusted to ensure that $\geq$1% plastic strain and fuel centerline melting do not occur during the worst anticipated operational occurrence (AOO), including transients initiated from partial power operation.

For Exxon fuel the T factor used to adjust the APRM setpoints is based on the FLPD calculated by dividing the actual LHGR by the LHGR obtained from Figure 3.2.2-1. The LHGR versus exposure curve in Figure 3.2.2-1 is based on Exxon's Protection Against Fuel Failure (PAFF) line shown in Figure 3.4 of XN-NF-85-67, Revision 1. Figure 3.2.2-1 corresponds to the ratio of PAFF/1.2 under which cladding and fuel integrity is protected during AOO's.

APRM SETPOINTS (Continued)

For GE fuel the T factor used to adjust the APRM setpoints is based on the FLPD calculated by dividing the actual LHGR by the LHGR limit specified for GE fuel in Specification 3.2.4.1.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

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

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figure 3.2.3-1 and 3.2.3-2.

The evaluation of a given transient begins with the system initial parameters shown in the cycle specific transient analysis report that are input to a Exxon-core dynamic behavior transient computer program. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle. The codes and methodology to evaluate pressurization and non-pressurization events are described in XN-NF-79-71 and XN-NF-84-105. The principal result of this evaluation is the reduction in MCPR caused by the transient.

Figure 3.2.3-1 defines core flow dependent MCPR operating limits which assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The flow dependent MCPR is only calculated for the manual flow control mode. Therefore, automatic flow control operation is not permitted. Figure 3.2.3-2 defines the power dependent MCPR operating limit which assures that the Safety limit MCPR will not be violated in the event of a feedwater controller failure initiated from a reduced power condition.

Cycle specific analyses are performed for the most limiting local core wide transients to determine thermal margin. Additional analyses are performed to determine the MCPR operating limit with either the Main Turbine Bypass inoperable or the EOC-RPT inoperable. Analyses to determine thermal margin with both the EOC-RPT inoperable and Main Turbine Bypass inoperable have not been performed. Therefore, operation in this condition is not permitted.

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.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor recirculation loop inoperable has been evaluated and found acceptable, provided that the unit is operated in accordance with Specification 3.4.1.1.2.

For single loop operation, the MAPLHGR limits are multiplied by a factor of 0.0. This multiplication factor precludes extended operation with one loop out of service.

For single loop operation, the RBM and APRM setpoints are adjusted by a 7% decrease in recirculation drive flow to account for the active loop drive flow that bypasses the core and goes up through the inactive loop jet pumps.

Surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive reactor vessel internals vibration. Surveillance on differential temperatures below the threshold limits of THERMAL POWER or recirculation loop flow mitigates undue thermal stress on vessel nozzles, recirculation pumps and the vessel bottom head during extended operation in the single loop mode. The threshold limits are those values which will sweep up the cold water from the vessel bottom head.

THERMAL POWER, core flow, and neutron flux noise level limitations 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.

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 pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during the loop operation, continued operation is permitted in the single loop 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 function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 10 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

SNUBBERS (Continued)

To provide assurance of snubber functional reliability one of three functional testing methods is used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or
2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7.4-1, or
3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7.4-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.5 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4 7.6 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2 systems, Halon systems and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurances that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the weight and pressure of the tanks.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.7 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.8 MAIN TURBINE BYPASS SYSTEM

The required OPERABILITY of the main turbine bypass system is consistent with the assumptions of the feedwater controller failure analysis in the cycle specific transient analysis.

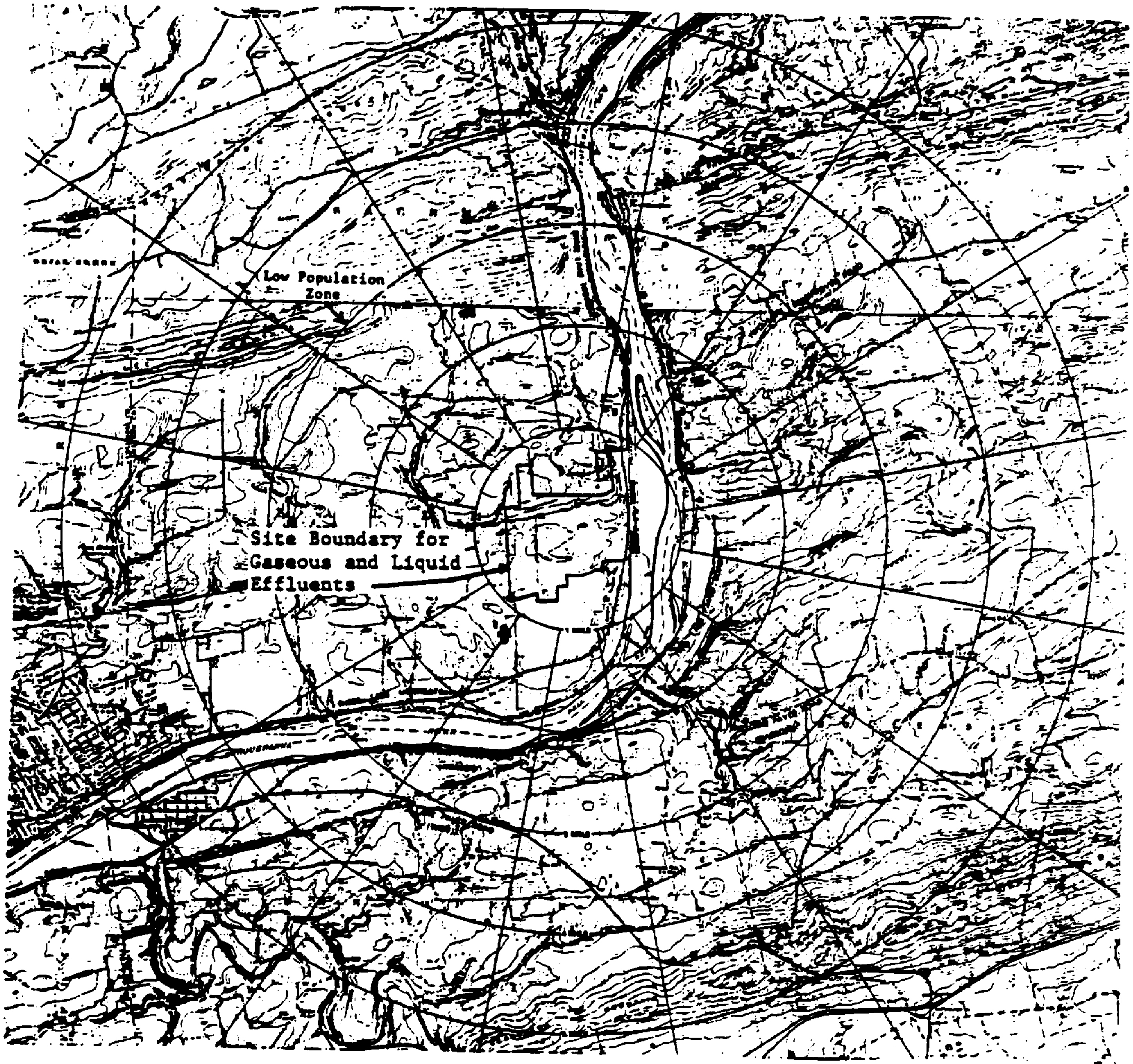


FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED AREAS
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 or 79 fuel rods and two water rods clad with Zircaloy -2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.90 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum average enrichment of 4.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.

2. 1500 psig from the recirculation pump discharge to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,400 cubic feet at a nominal T_{ave} of 528°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-388

1.0 INTRODUCTION

By letter dated June 19, 1986, (Ref. 1) Pennsylvania Power and Light Company (PPLCo or the licensee) proposed to amend Appendix A of Facility Operating License No. NPF-22. The requested amendment furnished information to support authorization for Susquehanna 2 Cycle 2 operation with 9X9 fuel supplied by Exxon Nuclear Company, and revised single loop operation (SLO) provisions in the body of the Technical Specifications.

The Susquehanna 2 Cycle 2 (S2C2) reload will consist of 324 fuel bundles fabricated by Exxon Nuclear Company (ENC). These 9X9 bundles are comprised of 79 active fuel rods and two inert water rods. During Cycle 2 operation, the 9X9 fuel will reside with 440 General Electric P8x8R fuel assemblies presently in the core. In support of the S2C2 reload PPLCo submitted topical reports which describe the design and safety analysis (Ref. 2), the plant transient analysis (Ref. 3), and the LOCA-ECCS analysis (Ref. 4) for the ENC 9X9 fuel. Additional information in response to NRC inquiries was provided by the licensee in References 5 and 15.

To evaluate the single loop operation (SLO) provisions in the Technical Specifications, PPLCo submitted a core stability assessment of ENC 9X9 fuel at Susquehanna 2 in Appendix A of XN-NF-86-60 (Ref. 2). However PPLCo is not requesting SLO approval at this time. The interim TS modifications are discussed in Section 3.3 of this Safety Evaluation.

2.0 EVALUATION OF FUEL DESIGN

2.1 Fuel Mechanical Design

The S2C2 core reload will include 324 Exxon Nuclear Company (ENC) new 9X9 fuel assemblies with the designation XN-1. These reload assemblies contain 79 fuel rods and two water rods. The 324 assemblies will have a bundle average enrichment of 3.31 percent. The fuel design and safety analysis for the 9X9 fuel are described in the Susquehanna 2 specific report XN-NF-86-60 (Ref. 2) and the generic mechanical design report XN-NF-85-67 Revision 1 (Ref. 6). The staff has approved the latter report and issued an SER on July 23, 1986 (Ref. 7).

Table 2.1 of XN-NF-85-67 Revision 1 gives the pertinent data for the XN-1 9X9 fuel. Neutronic values specific to the S2C2 reload are given in Table 4.1 of XN-NF-86-60 (Ref.2). The burnable poison rods contain 4.00 weight percent Gd_2O_3 blended with 3.27 weight percent U-235 to reduce the initial reactivity. The ENC XN-1 fuel is designed to fit into the existing channel boxes. A more detailed description can be found in Table 2.1 of XN-NF-85-67. Based on our review of the information in Table 2.1, we find the mechanical design of the Exxon 9X9 fuel for the S2C2 reload is acceptable. However, approval of extended exposure limits for future operating cycles is contingent on our approval of XN-NF-82-06(P) Supplement 1 (Ref. 8).

2.2 Rod Pressure

For the S2C2 ENC 9X9 reload fuel, calculation of the fuel rod internal pressure was done in accordance with acceptance criteria cited by ENC in XN-NF-85-67, Revision 1 (Ref. 6). The evaluation was performed with RODEX 2A which is a revision of the RODEX2 code (revised fission gas release model) used in the analysis of previous ENC fuel designs. Our review of the RODEX 2A topical report is complete and the staff Safety Evaluation Report has been issued (Ref. 9). The staff has concluded that the acceptance criteria for rod internal pressure can be fully met throughout the entire expected irradiation life of the 9X9 fuel.

2.3 Fuel Rod Bow

Our review of XN-NF-85-67, Revision 1 (Ref. 6) has been completed (Ref. 7) so that we may conclude that Exxon has demonstrated conformance to approved rod bow design limits for minimum gap spacing to a fuel assembly average exposure of 23,000 MWD/MTU for the 9x9 fuel. Projected peak assembly burnups for the S2C2 reload are in the range of 11,000-13,000 MWD/MTU for the 9x9 fuel. We find the S2C2 core acceptable with respect to rod bow considerations. However, since the rod bow criteria are only supported for two cycles of operation, additional justification with regard to fuel rod bowing must be provided for Susquehanna 2 operation beyond an average burnup of 23,000 MWD/MTU for the 9x9 fuel.

2.4 Fuel Centerline Melting

The design basis for the ENC fuel centerline temperature is that no fuel centerline melting should result from normal operation including transient occurrences. The results of an evaluation reported in the S2C2 reload analysis report XN-NF-86-60 (Ref. 2) were based on RODEX2A. RODEX2A has been reviewed and approved (Ref. 9) and the staff has concluded that the generic methodology for the ENC 9X9 fuel is acceptable for the S2C2 reload fuel.

2.5 Cladding Swelling and Rupture

The cladding swelling and rupture models in XN-NF-82-07 (Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model) have been approved for use in the ENC ECCS Evaluation Model and have been incorporated in the approved ENC EXEM/BWR ECCS model. This model was used in the ECCS analysis for the S2C2. The staff has verified that ENC is using the approved model for the 9X9 fuel ECCS analysis, and we find the application to be acceptable.

2.6 Linear Heat Generation Rate (LHGR) Limit for ENC 9X9 Fuel

Pennsylvania Power and Light Company has provided a figure of Linear Heat Generation Rate Limit vs Planar Exposure for the ENC 9X9 fuel type to be incorporated in the Susquehanna 2 Technical Specifications (Ref. 4). This figure was approved in Reference 7 to reflect the design values which have been previously reviewed and approved for the ENC 9X9 fuel in connection with our review of XN-NF-85-67, Revision 1 (Ref. 6).

2.7 LOCA-Seismic Mechanical Response

The licensee has discussed the mechanical response of the ENC 9x9 fuel assembly design during LOCA-seismic events in Appendix B of Reference 2. The discussion included a comparison of the physical and structural properties of the new 9x9 fuel and the prior GE 8x8 fuel and a reference to an ENC Topical Report XN-NF-84-97 (Ref. 10). The staff SER on Reference 10 has been issued (Ref. 11); the conclusion in the SER stated that conformance to the acceptance criteria of Standard Review Plan Section 4.2, Appendix A can be demonstrated by referencing XN-NF-84-97 (P) and submitting justification that the analyses in the topical report bounds the particular application under review. However, since an analysis specific to Susquehanna 2 has not been performed by ENC, the licensee has chosen to perform comparisons between the ENC 9x9 assembly and the GE 8x8 assembly currently loaded in the Susquehanna Unit 2 reactor to show that the results of the prior GE studies would still apply. The Seismic-LOCA analysis for the GE fuel has been provided in connection with a prior review on Susquehanna Unit 1 (Ref. 28). The licensee has provided a comparison of the fuel assemblies in Table B1 of Reference 2. Results of the comparison and data on the natural frequencies of the GE fuel were provided separately by the licensee in Reference 29, to support the licensing of reload cores for PPLCo owned reactors only. Included in Reference 29 was a discussion of the differences in the determination of the natural frequencies of the ENC and GE fuel types. The comparisons are for ENC 8x8 and 9x9 and GE 8x8 fuel types. The staff has confirmed that the physical and structural characteristics of the ENC 9x9 and GE 8x8 fuel assemblies are sufficiently similar so that the mechanical response to design Seismic-LOCA events is essentially the same. Based on the considerations discussed above, we conclude that the original analysis is still applicable to Susquehanna 2 and the analysis indicating that the design limits are not exceeded is acceptable.

3.0 THERMAL HYDRAULIC DESIGN

The review of the thermal-hydraulic aspects of the S2C2 reload consisted of the following: (a) the compatibility of the ENC 9X9 and prior GE 8X8 fuel bundles; (b) the fuel cladding integrity safety limit; (c) the operating limit minimum critical power ratio (OLMCPR); (d) the amount of bypass flow associated with the different fuel designs; (e) thermal-hydraulic stability, and (f) the proposed technical specifications.

The objective of the review was to confirm that the thermal-hydraulic design of the reload core was accomplished using acceptable analytical methods, provides an acceptable margin of safety from conditions which would lead to fuel damage during normal operation and anticipated operational occurrences and ensures that the core is not susceptible to thermal-hydraulic instability.

3.1 Hydraulic Compatibility

Since a BWR core is a series of parallel flow channels connected to a common lower and upper plenum, the total pressure drop across the bundles will be equal. However, differences in the hydraulic resistances of the fuel designs may cause variations in axial pressure drop profiles across the bundles. Component hydraulic resistances for the proposed constituent fuel types in the S2C2 core have been determined in single phase flow tests of full scale assemblies. Additional analyses of the effects of hydraulic compatibility on thermal margin were presented in the S2C2 reload report (Ref. 2). The results of these analyses showed that the 9X9 hydraulic performance is equivalent to the GE 8X8 fuel. Based on our review of the information provided in the Cycle 2 reload report we conclude that the GE and ENC fuel types are hydraulically compatible.

3.2 Thermal-Hydraulic Stability

The thermal-hydraulic stability of the Susquehanna 2 core was analyzed using the methods identified in References 18 and 19. Reference 19 describes the use of the COTRAN model for use in the analysis of core thermal-hydraulic stability. The NRC has concluded that the use of COTRAN is acceptable in accordance with the restrictions cited in the applicable SER (Ref. 22). For S2C2 operation, the licensee has provided additional stability analyses for the ENC 9X9 fuel using ENC's advanced system stability model COTRANSA 2 documented in XN-NF-84-67(P) (Ref. 23), which is under review by the staff. The results of these analyses and comparison with results from the approved COTRAN code are provided in Reference 5. NRC Generic Letter 86-02 (Ref. 20) provided acceptance criteria to be applied to all core reloads and other design or operating modifications relating to thermal-hydraulic stability. An acceptable margin for ENC analysis of stability is a decay ratio of 0.75, which is a result of the estimated uncertainty of 25 percent in the calculation of the thermal-hydraulic stability decay ratio with the COTRAN code. Permanent approval of the COTRANSA 2 analytical methodology and results for Susquehanna 2 stability analyses is subject to benchmark tests to demonstrate that COTRANSA2 can adequately predict the decay ratio for reactor cores with Exxon 9x9 fuel as it approaches the limit value of 1.0. The licensee has committed to a stability test during startup of Susquehanna 2, with post test analysis performed by Exxon Nuclear Company using the current analytical methodology.

The stability tests are to be performed in conjunction with a cooperative program between the NRC, Oak Ridge National Laboratory (ORNL), Pennsylvania Power and Light Company and Exxon Nuclear Company (ENC). The proposed test program involves the collection of neutron noise measurements at pertinent operating states followed by independent calculations by ORNL using the LAPUR computer code and ENC using the COTRAN and COTRANSA2 stability methodology. The stability test proposal was evaluated by ORNL and is discussed in Reference 24; the evaluation included a review of the relevant documentation and a summary of two meetings between the parties involved. As a result of the discussions, it was agreed that noise level data would be collected at two measurement points on the Power/Flow map for the Susquehanna 2 reactor. The basis for the selection of the two points was the identification of one point close to the baseline noise level used as a reference for all fuel reloads and a second point within the "detect and suppress" region as defined by the General Electric Service Information Letter 380, Revision 1 (Ref. 25). Measurements collected at the second point will be made during single loop operation (SLO) tests instead of startup. The NRC was involved in the determination of the proposed data points, and we concur in their selection.

ENC has provided the results of decay ratio computations using COTRAN and COTRANSA2 in Reference 27. The computations were made for operating points comparable to the points selected for the test measurements; i.e., points within or at the boundaries of the detect and suppress region established by Reference 25 (GE SIL-380, Revision 1). Additional calculational results were provided in Reference 27 for benchmark tests performed at Peach Bottom Atomic Power Station. The results of calculations with COTRAN, COTRANSA2 (original version) and COTRANSA2 (current version) were compared by ENC in Reference 27. The conclusion drawn by ENC was that the refinements incorporated in the version of COTRANSA2 presently under staff review do not alter the basic methodology and the current version of COTRANSA2 is acceptable for stability calculations. For the purpose of judging the adequacy of the proposed Susquehanna 2 stability test program, the staff has reviewed the reference data and comparative calculations with COTRAN and COTRANSA2. Since the scope of the comparisons includes the test range and encompasses the detect and suppress region of the Power/Flow map, we conclude that the commitment by PPLCo to the proposed stability test program is acceptable. Based on the favorable predictions (0.59 using COTRAN and 0.70 using current ENC stability methodology), we also conclude that the proposed one-third core reload with the ENC 9x9 fuel in S2C2 is acceptable. However, we require that additional analysis and evaluation of the test results be performed for subsequent Susquehanna 2 reloads up to and including a full core loading with 9x9 fuel.

As part of its review of the relevant documentation, ORNL considered the reference data and discussion provided by ENC in Reference 26. In Reference 24, ORNL had two comments on the parameters affecting thermal-hydraulic stability. These comments are noted here for the record. The first comment dealt with the effect of burnup on stability. ENC drew the conclusion that the results of stability demonstration tests at two KRB-II reactors in Germany showed that the effect of the 9x9 fuel assemblies is a trend toward more stable conditions. In drawing the conclusion, the effect of burnup should be given more consideration. The second comment dealt with the need to differentiate between fuel time constant and attenuation factor in their effect on stability. These comments do not change the conclusions in this Safety Evaluation.

3.3 Single Loop Operation

The Pennsylvania Power and Light Company presently has Technical Specifications to permit extended reactor operation of Susquehanna Unit 2 with one recirculation loop out of service. Prior staff approval was based on a single loop operation (SLO) analysis performed by the General Electric Company (GE) to determine SLO operating limits with GE fuel in the core. The introduction of the ENC 9x9 reload fuel requires a separate analysis for S2C2. At this time, sufficient analysis has not been completed to support extended SLO of Susquehanna 2 with ENC 9x9 fuel. As an interim measure, the licensee has proposed a modification to the present Technical Specification Limiting Condition of Operation (LCO) for the SLO mode. The proposed change consists of setting the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit multiplier to 0.0 for extended SLO. The effect of this change is to preclude SLO for more than 12 hours. The staff finds this proposal acceptable. Revised analyses with current approved methodology applicable to the 9X9 fuel are to be provided in a future submittal and should include a specific analysis of the one-pump seizure accident.

4.0 TRANSIENT AND ACCIDENT ANALYSIS

4.1 Minimum and Operating Limit Critical Power Ratio

The minimum critical power ratio (MCPR) safety limit for the Cycle 2 reload was determined by the licensee to be 1.06 for all fuel types. A safety limit of 1.06 for GE fuel types was approved for the previous Susquehanna 2 operating cycle. The methodology for Cycle 2 is based on ENC's revised critical power methodology in XN-NF-524, Revision 1 (Ref. 12) which incorporates a constant flow MCPR formulation for BWR applications. The staff has completed its generic review of XN-NF-524 (Ref. 13) and has concluded that the methodology for arriving at an MCPR safety limit is acceptable. The XN-3 correlation used to develop the MCPR safety limit has been approved for the new 9X9 fuel type (Ref. 14). The methodology of XN-NF-524, Revision 1 was applied generically for the upcoming Cycle 2 and is considered applicable to the resident GE 8x8 fuel types as well as the ENC fuel. The staff has verified through its review of the S2C2 transient analysis report XN-NF-86-55 (Ref. 3) that the methodology for determining uncertainties and the application in determining the MCPR safety limit is in accordance with NRC approved methodology and is acceptable.

4.2 Operational Transients

Various operational transients could reduce the MCPR below the intended safety limit. The most limiting transients have been analyzed to determine which event could potentially induce the largest reduction (delta-CPR) in the initial critical power ratio. The transients which resulted in the largest delta-CPR are the Load Rejection Without Bypass and Feedwater Controller Failure. The results of an updated analysis using current methodology are provided in Reference 15. The staff review is discussed below.

The original Susquehanna 2 proposed Amendment 39 contained analysis results for the core-wide transients Load Rejection Without Bypass and Feedwater Controller Failure which were based on methodology described in XN-NF-79-71(A) (Ref. 16). This analysis was revised using an updated methodology based on XCOBRA-T (Ref. 17) and documented in Reference 15 for S2C2. (The XCOBRA-T Topical Report is currently under NRC review and the staff SER is in final processing.) The evaluation for S2C2 is based on the information provided by the licensee in Reference 15. This information includes the calculated delta-CPRs for the overpressurization transients and the MCPR operating limits for S2C2 operation. The staff review of the XCOBRA-T methodology has been completed to the point that we may conclude the approach to the calculation of delta-CPRs is acceptable. The calculated delta-CPRs for the Feedwater Controller Failure and Load Rejection Without Bypass are both equal to 0.24. The resulting MCPR operating limit of 1.30 is acceptable for incorporation into the S2C2 Technical Specifications for all fuel types.

4.3 Reactivity Insertion Transients

The control rod withdrawal error, the fuel loading error and the rod drop accident were evaluated for Cycle 2. The licensee used methods described in XN-NF-80-19, Volume 4 (Ref. 21 with staff SER included). The use of the Single Sequence Control strategy (in which rods inserted during power operation have low worth) assures that the control rod withdrawal error will not be limiting. Using a Rod Block Monitor setting of 108 percent of full power results in a delta-CPR of 0.21 for the control rod withdrawal error transient for 9X9 fuel. The change in CPR due to a fuel loading error is 0.19. These values are comparable to previous reloads and are not limiting.

The control rod drop accident evaluation yields a value of 109 cal/gm for the maximum deposited fuel enthalpy. This is well below the staff's criterion of 280 cal/gm, and is therefore acceptable.

5.0 LOSS OF COOLANT ANALYSIS (MAPLHGR LIMIT)

The MAPLHGR limits for the GE 8X8 fuel as given in the Susquehanna 2 plant Technical Specification remain applicable for Cycle 2. The licensee has proposed additional MAPLHGR limits for the ENC 9X9 fuel based on the analysis results provided in XN-NF-86-65 (Ref. 4). The limiting LOCA break calculations were performed for the Susquehanna 2 reactor with a full core of ENC 9X9 fuel. The approved EXEM/BWR ECCS Evaluation Model codes were used for the LOCA calculations with array dimensions increased to accomodate the 9X9 array. The resulting Peak Cladding Temperature (PCT) was 2147° F at a burnup of 20 GWD/MTU allowing a 53° F margin to the 10 CFR 50.46 limit. Metal-water reaction also peaks at 5.14 percent at a burnup of 20 GWD/MTU remaining well below the 17 percent limit required by 10 CFR 50.46. The MAPLHGR limits from this analysis are proposed for the Susquehanna 2 Technical Specifications for the ENC 9X9 fuel design. Since analysis of the LOCA was performed with reviewed and accepted codes, and the results are well within the limits of 10 CFR 50.46, the staff finds the proposed MAPLHGR limits for S2C2 acceptable.

6.0 TECHNICAL SPECIFICATION CHANGES

The Technical Specification Changes for S2C2 involve three general areas and are summarized below:

(1) Incorporation of Linear Heat Generation Rate (LHGR) limits for ENC 9X9 fuel as a Limiting Condition for Operation (LCO).

The additional information on LHGR limits discussed in Section 2.6 of this SER is provided in the addition of Figures 3.2.2-1 and 3.2.4.2-1 in the Susquehanna 2 Technical Specifications and the identification of the LCO in TS Section 3.2.4.2 (page 3/4-10a).

(2) Addition of ENC 9X9 fuel type.

MAPLHGR values for the new ENC 9X9 fuel type were added and burnup limits were adjusted for the earlier GE fuel types. MCPR safety limits were added for the new fuel and retained for the previous 8X8 fuel types. Additional MCPR operating limits were specified for Manual Flow Control and Automatic Flow Control for all fuel types. The previous TS Figures 3.2.3-1 and 3.2.3-2 were updated to reflect the new information.

(3) Restriction on Single Loop Operation (SLO) Provisions

The Limiting Conditions of Operation for Single Loop Operation in TS Section 3.4.1.1.2 (page 3/4 4-1c) were revised to set the MAPLHGR limit multiplier equal to 0.0 and to delete the RBM/APRM Control Rod Block Setpoints for one loop operation.

Administrative changes were also made to relevant definitions and core design information to reflect the addition of the new 9x9 fuel.

7.0 BASES FOR CONCLUSIONS

We have reviewed the information furnished by Pennsylvania Power and Light in References 1, 5, 15, and 29 and Supplementary ENC reports (Ref. 2, 3 and 4) relative to the proposed License Amendment to allow operation of Cycle 2 of Susquehanna 2. Based on the results of our review, we find that sufficient basis has been provided to allow the addition of 324 ENC 9X9 fuel bundles in the Susquehanna 2 core and interim restrictions on operation in the single loop operation (SLO) mode. The proposed TS changes are therefore approved for S2C2.

Our review as discussed in the Evaluation Sections above has identified certain restrictions relating to our incomplete review of the ENC 9x9 fuel and thermal-hydraulic stability considerations which limit approval to the upcoming Cycle 2 only. Specifically:

(1) SE Section 2.0: Approval of extended exposure limits for the ENC 9x9 fuel for future operating cycles is contingent upon our approval of XN-NF-82-06(P), Supplement 1 (Ref. 2). In addition, justification with regard to rod bow is required for operation beyond the projected S2C2 exposure levels.

(2) SE Section 3.2: The staff will reevaluate the thermal-hydraulic stability (THS) for Susquehanna 2 at the next reload cycle. The evaluation will consider permanent approval of the COTRANSA 2 analytical methodology including the results of benchmarking tests in the high decay ratio area.

8.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

9.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (51 FR 29009) on August 13, 1986. It should be noted that additional information for the purpose of clarification was provided to the staff after noticing of the proposed amendment. The staff consulted with the state of Pennsylvania. No public comments were received, and the state of Pennsylvania did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

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Dated: October 3, 1986

REFERENCES

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3. XN-NF-86-55, "Susquehanna Unit 2 Cycle 2 Plant Transient Analysis", Exxon Nuclear Company, May 1986.

4. XN-NF-86-65, "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for 9X9 Fuel", Exxon Nuclear Company, May 1986.

5. Letter, B. D. Kenyon (PPLC) to Director (NRR), "Susquehanna Steam Electric Station - Response to Request for Additional Information Regarding 9X9 Fuel Stability", dated July 10, 1986.

6. XN-NF-85-67, Revision 1, Generic Mechanical Design Report for Exxon Nuclear Jet Pump BWR Reload Fuel, April 1986.

7. Letter G. C. Lainas (NRC) to G. N. Ward (ENC), Acceptance for Referencing of Licensing Topical Report XN-NF-85-67(P), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel", dated July 23, 1986.

8. XN-NF-82-06(P) "Qualification of Exxon Nuclear Fuel for Extended Burnup", March 2, 1982, and Supplements 1, 2, 4 and 5.

9. Letter, G. C. Lainas (NRC) to G. N. Ward (ENC), Acceptance for Referencing of Licensing Topical Report XN-NF-85-74(P), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model", dated June 24, 1986.

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12. XN-NF-524(P)(A), Revision 1, "Exxon Nuclear Critical Power Methodology for BWRs", November, 1983.

13. Letter, C. O. Thomas (NRC) to J. C. Chandler (ENC) October 31, 1983, "Acceptance for Referencing of Licensing Topical Report XN-NF-524(P)".

14. Letter, C. O. Thomas (NRC) to J. C. Chandler (ENC), February 1, 1985, "Acceptance for Referencing of Licensing Topical Report XN-NF-734, Confirmation of the XN-3 Critical Power Correlation for 9X9 Fuel Assemblies."

15. Letter, B. D. Kenyon (SSES) to Director, NRR (NRC) dated July 25, 1986, "Supplement to Proposed Amendment 39 to License No. NPF-22".

16. XN-NF-79-71(A), Rev. 2, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors", November 1981.

17. XN-NF-84-105(P) Rev. 1, Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis", March 1986.

18. XN-NF-80-19(A) Volume 1, Supplements 1 and 2 "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronics Methods for Design and Analysis", March 1983.

19. XN-NF-691(A), and Supplement 1, "Stability Evaluation of Boiling Water Reactor Cores", Exxon Nuclear Company, August 1984.

20. Technical Resolution of Generic Issue B-19 - Thermal Hydraulic Stability (Generic Letter No. 86-02), January 23, 1986.

21. XN-NF-80-19(P)(A) Volume 4, "Exxon Nuclear Methodology for Boiling Waters: Application of the ENC Methodology to BWR Reloads", dated September 1983.

22. Letter, G. Lainas (NRC) to G. N. Ward (ENC) dated April 30, 1986, "Acceptance for Referencing of Licensing Topical Report XN-NF-80-19(P), Volume 4, Revision 1, Exxon Nuclear Methodology for Boiling Water Reactors, Application of the ENC Methodology for BWR Reloads."

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25. General Electric Service Information Letter No. 380, Revision 1, February 10, 1984.

26. XN-NF-86-90(P), Rev. 1, "Boiling Water Reactor 9x9 Fuel Operating Experience" dated July, 1986.

27. "Supplemental COTRAN and COTRANSA2 Stability Calculations," attachment to July 10, 1986 letter, B. D. Kenyon (PP&L) to E. Adensam (NRC).

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